SAFETY EVALUATION

BY THE

DIRECTORATE OF LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

NUCLEAR POWER PLANT

POPE COUNTY, ARKANSAS

DOCKET NO. 50-313

JUN 6 1973

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APPENDICES

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
AP&L	Arkansas Power & Light Company
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Material
Btu/hr-ft^2	British Thermal Units per hour per square foot
BWST	Borated Water Storage Tank
B&W	Babcock & Wilcox
cfs	cubic feet per second
CFR	Code of Federal Regulations
CRA	Control Rod Assembly
DBA	Design Basis Accident
DBE	Design Basis Earthquake (currently called Safe Shutdown Earthquake
d-c	direct current

ABBREVIATIONS (cont'd)

Δk	reactivity change
Δt	temperature change or difference
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	feet
ft^2	square feet
ft^3	cubic feet
fps	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 Adsorbing (filter)
I-131	Iodine 131
IEEE	Institute of Electrical and Electronics Engineers

ABBREVIATIONS (cont'd)

in	inch
km	kilometer
kV	kilovolt
kW	kilowatt
kW/ft	kilowatts per foot
lb	pound
LOCA	Loss-of-Coolant Accident
LPZ	Low Population Zone
m	meter
m^2	square meters
mph	miles per hour
m/s	meters per seond
MSL	mean sea level
MWe	megawatts electrical
MWt	megawatts thermal
mrem	one thousandth of a rad equivalent man
NaOH	sodium hydroxide
NDT	nil ductility transition
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head

ABBREVIATIONS (cont'd)

NSSS	nuclear steam supply system
OBE	Operating Basis Earthquake
PMF	probable maximum flood
ppm	parts per million
PSAR	Preliminary Safety Analysis Report
psf	pounds per square foot
psi	pounds per square inch
psig	pounds per square inch gauge
psia	pounds per square inch absolute
PWR	pressurized water reactor
QA	quality assurance
QC	quality control
RCPB	reactor coolant pressure boundary
R&D	Research and Development
rem	rad equivalent man
sec/m^3	seconds per cubic meter
Sr-89	srontium 89
Sr-90	srontium 90
U-235	uranium 235
UO_2	uranium dioxide
USGS	United States Geological Survey

1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

The Arkansas Power & Light Company (hereinafter referred to as AP&L or the applicant) by application dated November 29, 1967, and as subsequently amended, requested a license to construct and operate a pressurized water reactor, identified as the Russellville Nuclear Unit (later as Arkansas Nuclear One - Unit 1 and hereinafter referred to as ANO-1) at a site on the Dardanelle Reservoir in Pope County, Arkansas. The Atomic Energy Commission's Regulatory staff reported the results of its review prior to construction in a Safety Evaluation Report dated October 1, 1968. Following a public hearing before an Atomic Safety and Licensing Board in Russellville, Arkansas on October 30, 1968, the Commission issued Provisional Construction Permit CPPR-57 on December 6, 1968.

On April 23, 1971, the applicant filed, as Amendment No. 19, the Final Safety Analysis Report (FSAR) required by 10 CFR 50.34(b) as a prerequisite to obtaining an operating license for the facility. The operating license application is for a core power level of 2568 megawatts thermal (MWt), the same thermal power considered by the Regulatory staff in the construction permit review. Our evaluation of the design characteristics, the engineered safety features, the containment, and the accident analyses has been based on operation at the

2568 MWt core power level as described in the applicant's Final Safety Analysis Report (Amendment No. 19) and subsequent Amendments 20 through 37 inclusive, all of which are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C., and at the Arkansas River Valley Regional Library, Dardanelle, Arkansas. In the course of our safety review of the material submitted, we held a number of meetings with representatives of the applicant, the nuclear steam supply system (NSSS) manufacturer, the Babcock & Wilcox Company (B&W), and the applicant's architect-engineer, Bechtel Corporation, to discuss the plant design, construction, proposed operation and performance under postulated accident conditions. A chronology of our review is attached as Appendix A to this evaluation.

1.2 General Plant Description

The ANO-1 power plant is one of two pressurized water nuclear plants proposed to be operated at the Arkansas Nuclear One site. Unit 2 (ANO-2), for which construction permit No. CPPR-89 was granted on December 6, 1972, will have a different NSSS (designed by Combustion Engineering, Inc.) but the balance of plant and overall arrangement will be essentially the same. These two plants will share the same control room but little other equipment. The only shared engineered safety feature will be the emergency cooling pond, an ultimate heat sink for both units.

The ANO-1 NSSS uses a pressurized water reactor in a 2-loop reactor coolant system. The reactor will be fueled with slightly enriched uranium dioxide in the form of ceramic pellets enclosed in Zircaloy fuel tubes with welded end plugs. The fuel rods are grouped and supported in assemblies. Initially, the reactor core will be loaded in three regions each utilizing fuel of a slightly different enrichment of U-235. Water which serves as both moderator and coolant is circulated through the reactor coolant system by four pumps. The circulated water, heated by the reactor, flows through the two steam generators (one in each loop) where heat is transferred to the secondary (steam) system. The water then flows to the two parallel pumps in each loop for return to the reactor core to complete the cycle. An electrically heated and spray-cooled pressurizer attached to one of the coolant loops establishes and maintains the reactor coolant pressure, and provides a surge chamber and water reservoir to accommodate reactor coolant volume changes during operation.

The steam produced in the steam generators is used to drive the turbine generator which converts the heat energy to electrical energy. After passing through the turbine, the steam is condensed and the condensate returned to the steam generators to repeat the cycle. The condensers are cooled by water drawn from, and recirculated to, the Dardanelle Reservoir on the Arkansas River.

The reactor coolant system is a closed piping system. It consists of the reactor enclosed in its pressure vessel, steam generators, reactor coolant pumps, reactor coolant piping and the pressurizer. This system is in turn housed inside the reactor (or containment) building, a cylindrical, prestressed concrete structure with a shallow dome roof and a flat reinforced concrete base slab. The inside surface of the reactor building is sealed with a welded steel liner. The reactor building provides a barrier to the escape of radioactive products that might be released from the reactor coolant system in the event of an accident. In addition, the reactor building is equipped with a spray system designed to reduce rapidly both the pressure and the fission product concentration within the containment after a postulated accident.

Auxiliary systems, including the chemical and volume control system, the waste handling system, auxiliary coolant systems, spent fuel storage facility, and components of the engineered safety features are located in an auxiliary building, adjacent to and abutting the reactor building. The regions of this auxiliary building adjacent to penetrations from the reactor building are designated as the penetration rooms and are specially ventilated to filter any leakage from the penetrations. The penetration rooms are equipped with redundant fan-filter systems, either of which is capable of maintaining a negative pressure in the room relative to the environment. This is to assure that a major portion of any leakage of fission products from the primary containment after a release inside is subjected to single pass filtration through high efficiency particulate adsorbing (HEPA) and charcoal filters prior to release to the environment.

Rapid reactivity control of the reactor will be achieved by control rods (neutron absorbers) that will be moved vertically within the core by individual control rod drives. Boric acid dissolved in the coolant will be used as a neutron absorber to provide steady state reactivity control.

A reactor protection system is provided that will automatically initiate appropriate corrective actions whenever plant conditions monitored by the system reach preestablished safety limits.

Appropriate instrumentation circuitry is provided to initiate closure of isolation valves and operation of the engineered safety features should these actions be required. These engineered safety features include the containment systems with their supporting heat removal systems, isolation systems, a filtered purge system for combustible gas control, an emergency core cooling system (ECCS) that will prevent the reactor core from overheating for a broad spectrum of postulated loss of coolant accidents, an emergency feedwater system, and an emergency electrical supply system.

1.3 Comparison with Similar Facility Designs

Many features of the design of the ANO-1 plant are similar to those we have evaluated and approved previously for other nuclear power plants now under construction or in operation especially the Oconee Nuclear Station - Unit 1 (Oconee-1) which is the lead plant for this type of B&W NSSS. To the extent feasible and appropriate, we have made use of these previous evaluations in conducting our review of ANO-1. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our Safety Evaluation Reports for these other facilities also have been published and are available for public inspection at the Atomic Energy Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C.

1.4 Identification of Agents and Contractors

AP&L has arranged for the purchase of equipment and consulting, engineering, and construction services for the design and construction of ANO-1. As sole owner, AP&L is responsible for the design, construction and operation of ANO-1.

The Bechtel Corporation has been retained for architectural, engineering, procurement and construction services. They are also providing assistance in employee training, acceptance testing, quality control, and initial start-up of the plant.

B&W manufactured and delivered the complete nuclear steam supply system and supplied the initial reactor core fuel. In addition, B&W is supplying technical consultation for erection, fuel loading, testing, and initial start-up of the complete nuclear steam supply system. B&W is also participating in the training of the initial plant operating staff personnel.

Dames and Moore, Inc., is the principal meteorology consultant for the ANO-1 site.

1.5 Summary of Principal Review Matters

Our evaluation included a technical review of the information submitted by the applicant particularly with regard to the following principal matters:

1.5.1 Site

We evaluated the population density and land use characteristics of the site environs and the physical characteristics of the site, including seismology, meteorology, geology, and hydrology. Our purpose was to determine that these characteristics have been adequately established and appropriately considered in the final design of the plant, and to further determine that the site characteristics in conjunction with the design features of the facility are consistent with the Commission's siting criteria provided in 10 CFR Part 100.

1.5.2 Criteria

We evaluated the design, fabrication, construction, and testing and performance characteristics of the plant structures, systems, and components important to safety to determine that they are in accord with the Commission's General Design Criteria, Quality Assurance Criteria, Regulatory Guides and appropriate industrial codes and standards, and to determine that any departures from these criteria, codes, and standards have been identified and justified.

1.5.3 Design Basis Accidents

We evaluated the expected response of the facility to various anticipated operating transients and to a broad spectrum of accidents and selected a few highly unlikely postulated accidents (design basis accidents) the potential consequences of which would exceed those of all other accidents considered. We then performed conservative analyses of these design basis accidents to determine that the calculated potential offsite doses that might result from their postulated occurrence would be within the guidelines of 10 CFR Part 100.

1.5.4 Radioactive Releases

We also evaluated the design of the systems provided for control of the radioactive effluents from normal plant operation to determine that these systems are capable of controlling the release of such radioactive wastes from the facility within the limits of the Commission's regulations (10 CFR Part 20). We further evaluated these systems to determine that the equipment provided can be operated in such a manner as to reduce radioactive releases to levels that are as low as practicable.

1.5.5 Organization

We evaluated the applicant's engineering and construction organizations, plans for the conduct of plant operation, including the proposed organization, staffing and training program, the plans for industrial security, and the scope of 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 is technically qualified, staffed and organized to safely operate the plant.

1.5.6 Financial Qualifications

We evaluated the financial position of AP&L to determine that AP&L has adequate financial resources to operate the ANO-1 plant in accordance with the activities that would be permitted and required by an operating license.

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

2.1.1 Site Location

ANO-1 is located on an 1100 acre tract of land adjacent to the Dardanelle Reservoir in Pope County, Arkansas, approximately five miles from the town of Russelville and two miles from the village of London.

2.1.2 Site Description

Figures 1, 2 and 3 depict the location of the site and its relationship to its surroundings. The minimum exclusion distance from the plant vent to the nearest property line is 0.65 miles. The low population zone as specified by the applicant includes the area within a four mile radius of the plant. All land within the exclusion radius, except for the bed and banks of the Dardanelle Reservoir, is owned by AP&L. The bed and banks of the reservoir are controlled by the U. S. Army Corps of Engineers. The Corps has granted AP&L easements in all three areas where the bed and banks of the reservoir are within the exclusion radius. These easements include the right to prohibit human habitation and to exclude all persons from said areas in the event of an emergency situation. There are no residences within the exclusion radius.

2.1.3 Population and Population Distribution

The site of ANO-1 is at a considerable distance from any major population concentrations. AP&L reports a population center distance

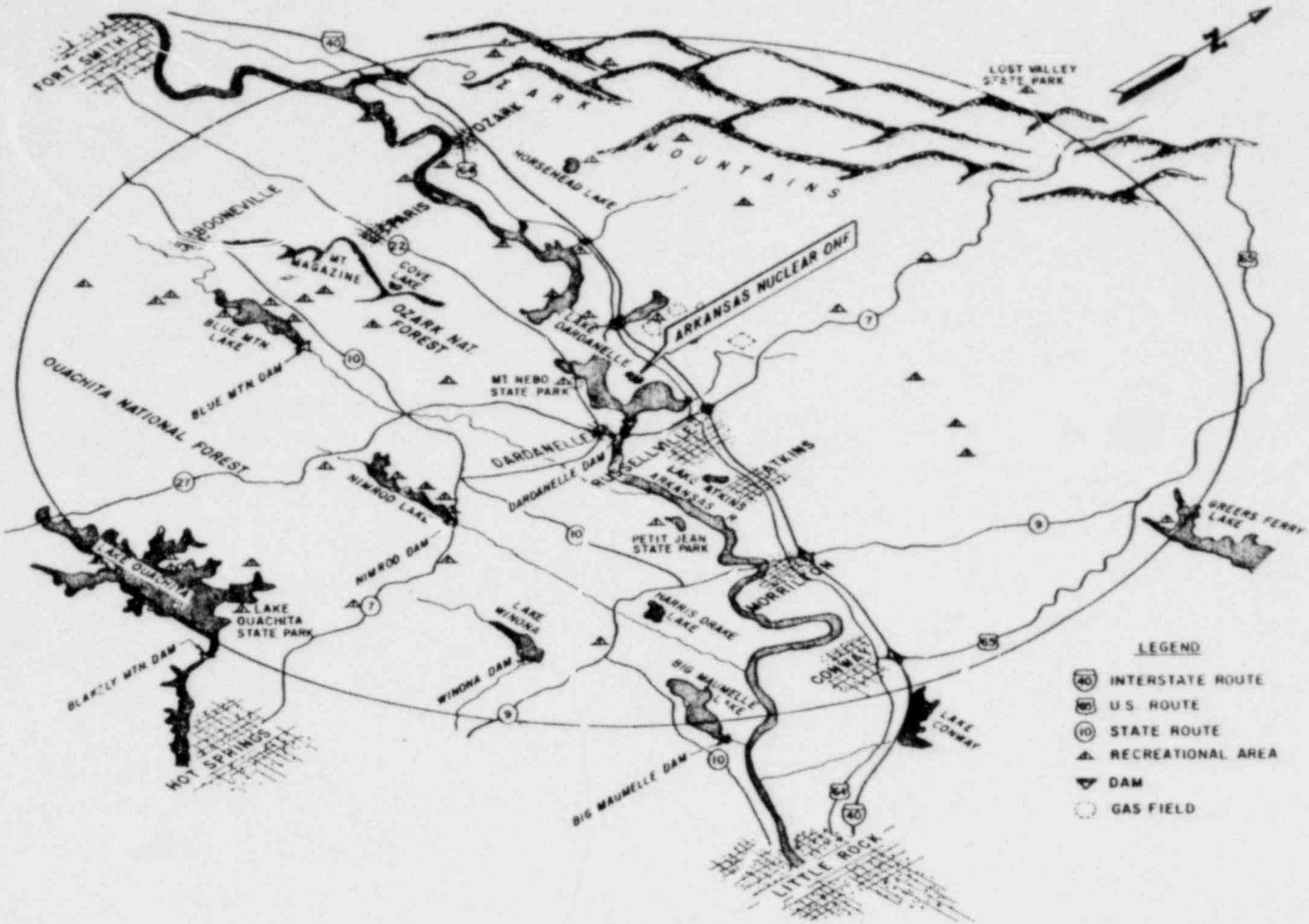

Fig. 1. Area within 50 miles of Arkansas Nuclear One.

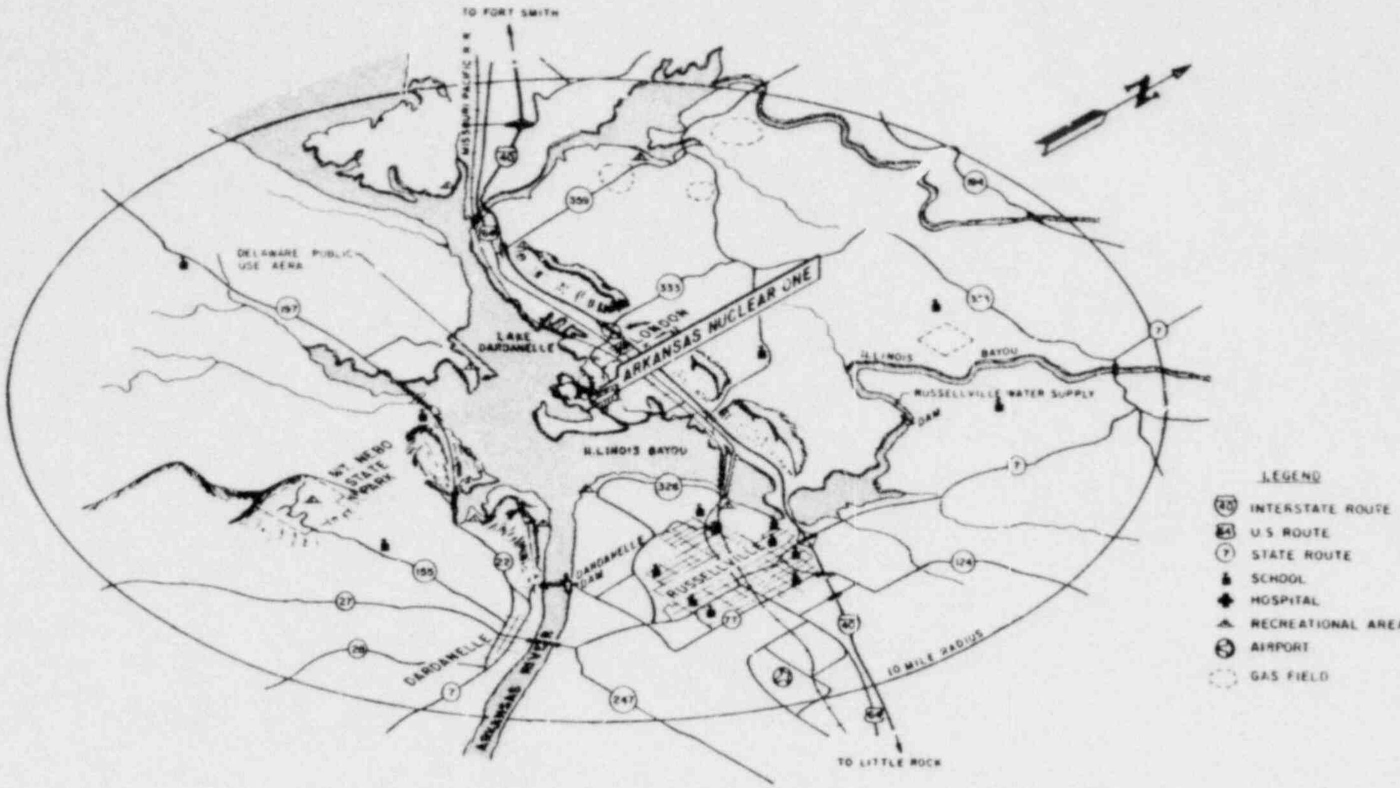

Fig. 2. Area within 10 miles of Arkansas Nuclear One.

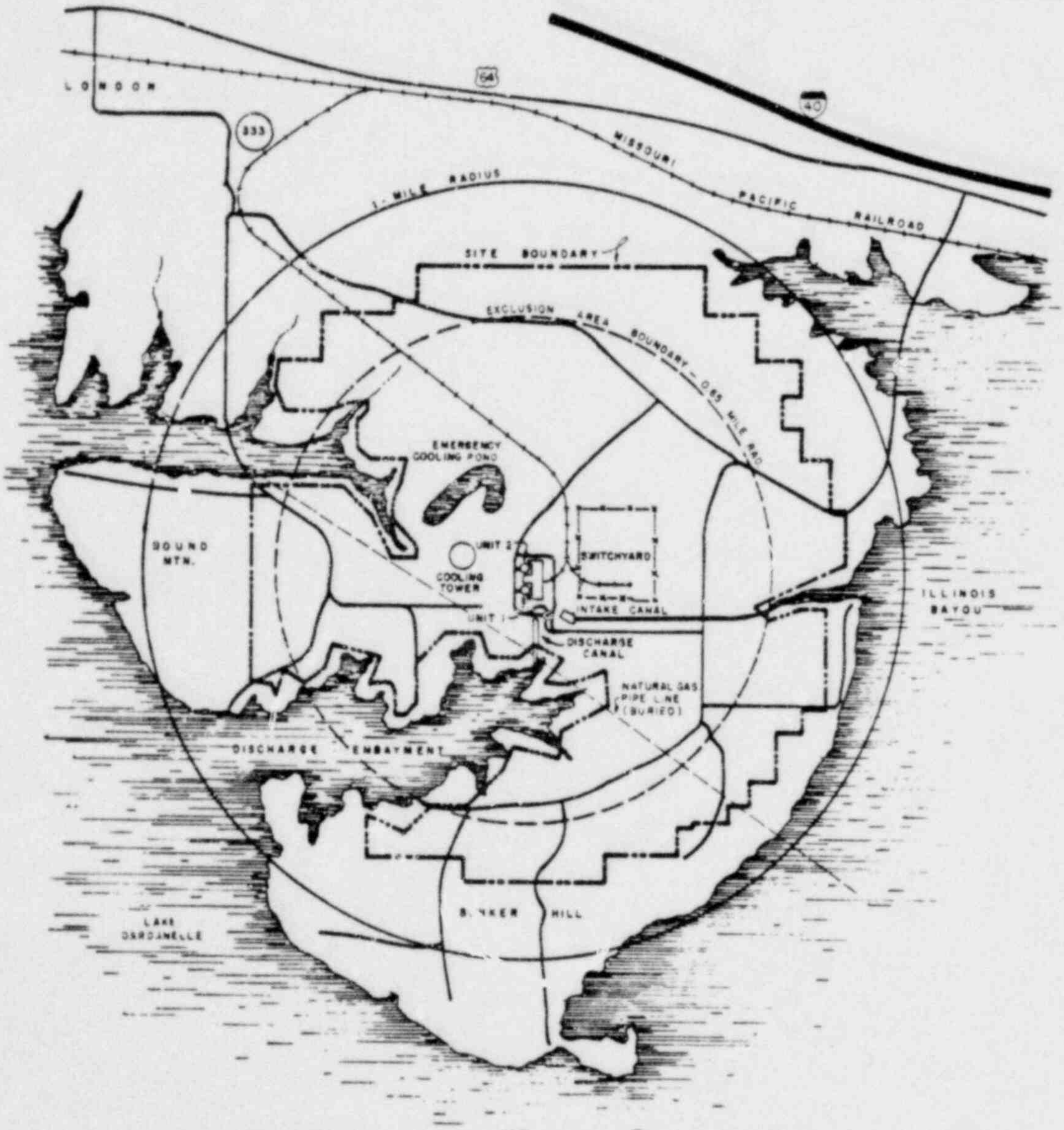

Fig. 3. Area within one mile of Arkansas Nuclear One.

of 55 miles. This is the distance to the nearest population center containing more than 25,000 persons, Hot Springs, Arkansas, which had a 1970 population of approximately 40,000. The resident 1970 population was 600 within a two mile radius of the plant and 1000 within a four mile radius. The applicant estimates these populations will increase to about 1200 and 3900 respectively in the year 2015. The 1970 population within 50 miles was approximately 155,000. Because of the recreational activities offered by the Dardanelle Reservoir, the applicant estimated that a maximum transient population of 1000 and 2000 will exist within five miles of the plant in the years of 1970 and 2015, respectively.

2.1.4 Uses of Adjacent Lands and Waters

The land in the immediate vicinity of the site is largely undeveloped land. There is little farming and dairying is limited to small, sparsely scattered herds. The closest dairy herd is pastured five miles from the site at the Arkansas Polytechnic College in Russellville. However, a small number of cows are pastured about two miles from the site; these animals were used as the basis of iodine-milk calculations.

2.1.5 Conclusions

On the basis of our evaluation of the present population data and the calculated potential offsite doses that might result in the event of a design basis accident (presented in Section 15 of this report), we have concluded that the size of the exclusion area, the low population zone distance, and the population center distance conform to guidelines set forth in 10 CFR Part 100 and are acceptable.

2.2 Nearby Industrial, Transportation and Military Facilities

The site is about six miles upstream from the Dardanelle Dam. A Missouri Pacific Railroad line, U. S. Highway 64, and Interstate Highway 40 pass north of the site at distances of 1.1, 1.2, and 1.3 miles, respectively. The Arkansas River shipping canal is about 1.4 miles south of the reactor buildings.

The closest airport is the Russelville Municipal Airport eight miles from the site, a small airport, without control tower, which handles light planes. There is no major airport within 50 miles of the plant site.

The Bunker Hill Road runs north and south through the exclusion area (see Figure 3), and would serve as one evacuation route for the Bunker Hill residents. Because the transit time by automobile along the road during an emergency was found to be less than six minutes, and because any radioactive plume would not be expected to follow a vehicle traveling the road, the Bunker Hill evacuation route was found to be acceptable.

Stone quarries exist at Midway and Altus and near the Dardanelle dam. The nearest quarry, near Dardanelle, is approximately five miles to the south, a sufficient distance so that blasting will not affect the site.

Sand and gravel deposits of commercial value are near Seranton and the Arkansas River at Dardanelle. Natural gas is produced in a number of locations within 10 miles of the site.

A natural gas transmission line owned by the Arkansas-Louisiana Gas Company crosses the site. The safety aspects of this 10.75-inch line which operates at 500 psig pressure were analyzed during the construction permit review of ANO-1. To meet our requirements the line was rebuilt with ASA Code B31.8 pipe for 1200 feet of its length nearest the reactor building and rerouted under the discharge canal with four feet of earth cover. In its present path the line comes no closer than 600 feet from the reactor building. The applicant has drawn up the Emergency Plan to arrange for prompt closure of nearby isolation valves (south of London and on the west side of Russellville) if the line should leak. With these precautions implemented we conclude that the location of the gas line is acceptable.

Because of the remoteness and natural state of the site, and the easements granted by the U. S. Army Corps of Engineers, public access to the water and banks of the embayment area will likely be limited to occasional fishing. Therefore, large numbers of people will not have to be evacuated from the embayment areas in the event of an accident. A dose calculation as a function of time was made at the embayment area 0.2 miles from the containment, the closest point where fishing could take place. This calculation indicates that at least 30 minutes are available for evacuation before the calculated doses would exceed 10 CFR Part 100 guidelines. Therefore, in the event of an accident, a sufficient time is available to undertake an evacuation of any persons fishing in the embayment areas.

2.3 Meteorology

2.3.1 Regional Meteorology

The plant site is in a climatic region that is primarily continental in character. The Boston Mountains, with elevations up to 2700 feet and oriented generally east-west on the north side of the Arkansas River valley, have some influence on the precipitation and airflow over the site. Snowfall in the region is relatively light.

2.3.2 Local Meteorology

The site is situated on a small peninsula which extends into the Dardanelle Reservoir and is almost surrounded by hills rising to about 150 feet above plant grade around the perimeter of the exclusion radius. The airflow over the site varies markedly from season to season. During the winter the winds show a strong preference for flow to the west with a secondary preference for east-southeasterly flow. During the summer the prevailing airflow is toward the northwest quadrant. The intermediate seasons, spring and fall, exhibit airflow which has attributes of both the winter and summer airflow characteristics.

2.3.3 Onsite Meteorological Measurements Program

Meteorologica measurements were obtained from June 1969 to May 1970 at a location 800 meters east of the reactor building. Measurements of wind at elevations of 20 and 150 feet and temperature at elevations of 5, 85 and 190 feet were made during this period. Calculation of atmospheric diffusion conditions based on data from these measurements were considered during the construction review of ANO-2. However, because the data recovery was only 56%, the meteological instruments were relocated by July 29, 1971. The instruments were relocated to a 190-foot tower about 100 meters from the old location and measurements of wind at elevations of 40 and 190 feet and temperature at elevations of 30, 85 and 190 feet have been made since that time. During the period July 29, 1971 through February 7, 1972, the 40-foot level anemometer was malfunctioning. Therefore the applicant established a correlation between the wind at the 40-foot level and the wind at the 190-foot level and corrected the 190-foot level winds to represent 40-foot level winds during that period. Data from February 7, 1972, to July 29, 1972, were used to complete the year of data required to provide atmospheric diffusion estimates. The data utilized to evaluate accident and annual average diffusion conditions at the site were wind at the 40-foot level, measured and extrapolated, and vertical temperatures difference (ΔT) between the 190- and 30-foot levels for the one year period of record. The joint data recovery during this period was 94%.

2.3.4 Short Term (Accident) Diffusion Estimates

In evaluation of diffusion of short term (0-2 hr) accidential releases from the plant, a ground level release with a building wake factor, cA, of 1100 m^2 was assumed. The relative concentration (X/Q) which is exceeded 5% of the time was calculated to be 6.8×10^{-4} sec/m^3 at the exclusion radius of 1046m (0.65 mile). This is equivalent to dispersion conditions produced by Pasquill Type F stability with a wind speed of 0.5 m/sec. Both the applicant and our consultant, the National Oceanic and Atmospheric Administration (NOAA), have calculated X/Q values which are not significantly different from ours (see Appendix B).

Using the diffusion models presented in Regulatory Guide 1.4 and the onsite meteorological data, we have estimated that the relative concentration for design basis accidents at the outer boundary of the low population zone (6436m or 4 miles). The relative concentrations are:

1.1×10^{-4} sec/m^3 for the 0-8 hour period,

1.1×10^{-5} sec/m^3 for the 8-24 hour period,

4.0×10^{-6} sec/m^3 for the 1-4 day period and

1.3×10^{-6} sec/m^3 for the 4-30 day period. These values are in essential agreement with the values presented by the applicant.

2.3.5 Long Term (Routine) Diffusion Estimates

The Regulatory staff calculated the maximum annual average relative concentration to be 1.5×10^{-5} sec/m^3 at the exclusion radius west of the plant. Both the applicant and our consultant have presented values which are in essential agreement with this.

2.3.6 Conclusions

We conclude that the data presented in the FSAR provide an acceptable basis for estimates of atmospheric diffusion conditions during accidental and routine gaseous effluent releases from the plant.

Hydrology

.1 Hydrologic Description

The ANO-1 site is on a northern floodplain peninsula of the Dardanelle Reservoir about six miles upstream of the Dardanelle Dam on the Arkansas River. The Arkansas River is a major waterway whereby 150,000 square miles of drainage area is controlled by more than 24 reservoirs. The site is about 259 miles upstream from the mouth of the Arkansas River, and the furthest upstream reservoirs from the site are more than 700 miles away.

The minimum navigation pool level of the Dardanelle Reservoir is elevation 336 feet mean sea level (MSL), and the reservoir normally varies between elevations 336 and 338 feet MSL to provide two feet of storage for power generation. Plant grade is elevation 353 feet MSL and plant ground floor levels are a foot higher.

The ultimate heat sink for the ANO-1 plant includes a 14-acre man-made emergency cooling water storage pond. This pond is filled by rainwater run-off from surrounding slopes; its bottom is at 341 feet MSL. A spillway at one end limits the static water level in the pond to a maximum of 347 feet MSL.

2.4.2 Floods

The greatest flood of record in the area occurred in 1943 with an estimated maximum runoff rate of 683,000 cubic feet per second (cfs). Dardanelle Dam is designed to discharge up to 900,000 cfs without exceeding a maximum water level of 338 feet MSL, and can safely pass a substantially more severe probable maximum flood (PMF). The levees along the river channel in this area are designed to pass flows of 830,00` cfs.

2.4.3 Probable Maximum Floods

The U. S. Army Corps of Engineers has estimated that the PMF at Dardanelle Dam would have a maximum runoff rate of 1,500,000 cfs, and a corresponding reservoir elevation of 353.0 feet MSL. The Regulatory staff concurs with this estimate. To determine the corresponding water level at the site, AP&L has conservatively assumed a straight line variation in levels between elevation 353 feet MSL at Dardanelle Dam and elevation 389.5 feet MSL on the downstream side of Ozark Dam, 51 miles upstream. The estimated PMF level at the site, using this method, is elevation 358.0 feet MSL.

The effects of wind-wave activity coincident with a PMF on the Arkansas River were evaluated by the applicant. That evaluation indicated that waves on the reservoir near the plant could be as high as 2.5 feet above the 358.0 feet MSL PMF elevation. Also, the evaluation indicated that wave runup could reach an elevation of 368 feet MSL. The staff's independent analysis of potential wave action using computational techniques developed by the Corps of Engineers indicates that the applicant's analysis is conservative.

The emergency pond has also been evaluated to determine its ability to accept and pass abnormal runoff.

The spillway and exit channel of the emergency cooling pond have been designed by the applicant to pass a standard project flood (SPF) for its local drainage area. The diked sections were originally designed to provide about one foot of freeboard between these dikes and the SPF maximum water surface elevation. Based upon standard practice by the Corps of Engineers and others, we will require that this freeboard be increased to three feet or that the dike be erosion protected. The SPF runoff is about half as great as a PMF, and is considered to represent the most severe precipitation conditions reasonably characteristic of the region based upon historical hydometeorology, excluding extremely rare occurrences. Notwithstanding this, the Regulatory staff performed an independent analysis of the pond runoff characteristics for a storm as severe as a PMF. This analysis used standard Weather Bureau (now NOAA) precipitation estimates, a synthetically developed runoff model based on pond drainage area, spillway characteristics, and the standard three feet of freeboard above the SPF routinely assumed for designs of this type. From that analysis the staff determined that the spillway and exit channel could even pass the flow of a local PMF without a loss of pond inventory. Wind-generated wave action coincident with a PMF and the 3 feet of freeboard above the SPF would not, in the opinion of the staff, be serious enough at this site to threaten a loss of pond inventory. Accordingly, the staff concluded that the spillway and exit channel of the emergency cooling pond will be adequate when the dike is modified to provide 3 feet of freeboard (or equivalent); we have informed AP&L that this will be required prior to licensing.

2.4.4 Potential Dam Failures

The effects at the site of arbitrarily assumed upstream dam failures were investigated independently by both the applicant and the staff. In all cases it was determined that the water level at the site would be less than that produced by a PMF, even though failures of downstream structures might occur as the result of upstream failures.

2.4.5 Ice Flooding

Ice flooding can occur under extreme conditions, but the staff considers the controlling flood conditions to be those associated with a PMF.

2.4.6 Cooling Water

During normal operations, 1,700 cfs of cooling water for once-through cooling for ANO-1 is to be taken from Dardanelle Reservoir through the intake canal on one side of the peninsula, and, except for about 20 cfs which will be consumed in plant operation, is to be discharged back into the reservoir through the discharge canal on the opposite side of the peninsula.

The emergency cooling pond will serve as a heat sink for normal plant shutdown of either Unit 1 or Unit 2, as a source of water for simultaneously shutting down both units in the event of a loss of the Dardanelle Reservoir water inventory, or a plant accident. AP&L has stated that the pond is sized to contain sufficient water (84 acre-feet or 27.4 million gallons) for dissipating the total heat transferred to the Unit 1 and 2 service water systems as a result of a design basis accident in one unit, and a normal plant shutdown of the other unit, while limiting the cooling pond temperature to a maximum of 120°F. The worst condition is a design basis accident in Unit 1 concurrent with a normal shutdown of Unit 2. The staff's independent analysis of the emergency cooling pond as an Ultimate

Heat Sink for the worst case, using analytical models developed under research contract AT(11-1)-2224 (with the University of Pennsylvania), indicates that a sufficient water supply for 30 days will be available even under extreme summer environmental conditions. However, for very extreme and rare combinations of hydrometeorological parameters (wind speed, air temperature, dew point, etc.) use of these models indicated that the temperature of cooling pond might exceed 120°F two to three weeks after shutdown. Since this concern exists only for the simultaneous occurrence of a DBA in ANO-1 and a shutdown of ANO-2, the applicant will be required to resolve this concern before ANO-2 is licensed. The staff has concluded that the emergency cooling pond as a primary portion of the Ultimate Heat Sink is adequate subject to the freeboard increase described in Section 2.4.3.

2.4.7 Channel Diversions

ANO-1 is to normally take its cooling water from Dardanelle Reservoir, which is a part of the Arkansas River Navigation System. It is not anticipated that future upstream diversions large enough to affect plant operations will be made since the water is already committed to maintain minimum navigation depths. Natural diversions, such as landslides or flood-caused rerouting, are also considered unlikely. Even so, the storage available in the emergency pond is sufficient for safe shutdown in such remote circumstances.

2.4.8 Flooding Protection

All safety-related structures and equipment are located above elevation 369 feet MSL, the PMF level, or are protected from flooding and wave runup by structures which can be made watertight. The staff has concluded that the design for flood protection is conservative and adequate. The plant will be shut down, with an appropriate emergency plan to protect safety-related facilities, in the event of a severe flood.

2.4.9 Low Water Considerations

Daily streamflow records for the period of January 1923 to September 1957, collected at the Dardanelle gaging station just below the Dardanelle Dam, have been adjusted by the Corps of Engineers to reproduce flows as they would have been regulated by the complete system of upstream dams. The minimum daily average flow as computed in this study was 4,000 cfs during the driest critical month of record; ANO-1 requires only 1,700 cfs of cooling water.

It is possible for the inflow to Dardanelle Reservoir to be zero under very exceptional circumstances involving emergency operation of upstream dams. These conditions would exist for only a few hours, however, during which time there would either be adequate water in storage in the reservoir, or the plant could be shut down and safely maintained in shutdown by using the emergency cooling pond for a period of 30 days or more. Similarly, a decrease of level in Dardanelle Reservoir to below plant pump intake levels, which could result in the highly unlikely event of failure of the Dardanelle Dam, would still not lead to an unacceptable situation since the cooling pond would be available for shutdown and cooldown.

Failure of Dardanelle Dam would be made known to plant operators by an alarm in the control room that is automatically activated when the reservoir level has dropped one foot below the normal minimum operating level of 336.0 feet MSL. The applicant has stated that 30 minutes are required to operate all 6 sluice gates to transfer to the emergency cooling pond, and that a minimum time of approximately 85 minutes would be available before the reservoir level can drop below the minimum required submergence level. The minimum required reservoir submergence level for plant operation is elevation 327.3 feet MSL.

The emergency cooling pond, to be kept at a normal level of 347 feet MSL (84 acre-feet), will provide a shutdown-cooldown source of water. The pond will be replenished by natural runoff, or in the event natural runoff is not sufficient from the Russellville water supply. Pond level will be monitored daily by the applicant.

2.4.10 Environmental Acceptance of Effluents

Ground water in the upper overburden at the site fluctuates with the level in Dardanelle Reservoir, but at the site is generally found about 10 feet below the surface sloping toward the reservoir. The lower bedrock zones are low-yield artesian sources. Domestic wells located down ground-water slope from the plant site extend into this bedrock; therefore, any contaminated water accidently spilled at the plant will migrate very slowly through the relatively impermeable clayey overburden toward the lake and should have no effect on water supplies taken from the artesian bedrock aquifer. The only use of ground water in the vicinity of the site is for local domestic purposes. Shallow domestic wells in the general vicinity are located up ground-water slope from the plant site; therefore, contamination from the plant is not possible.

The possibility of contamination of ground water, and/or migration of such contaminants to the reservoir is very remote because of the affinity of radionuclides for surface clays, and extremely low permeabilities. These factors should negate any significant or long distance travel of contaminated water.

No potable water supply is drawn from Dardanelle Reservoir or from the Arkansas River downstream of Dardanelle Dam because of its salinity.

2.4.11 Technical Specification and Emergency Operation Requirements

Plant grade elevation is 353 feet MSL and ground floor elevation for the buildings is 354 feet MSL. All critical equipment is located above elevation 369 feet MSL or is protected from flooding by structures which can be made watertight. A flood with a magnitude approaching that of a PMF would be forecast about five days prior to its arrival at the plant site. The applicant has stated that the plant will be shut down by the time the flood level reaches elevation 354 feet MSL, which is the elevation where flooding of the turbine building would commence. The plant will be shut down using normal shutdown procedures and, during the flood, operators will maintain the plant in a safe shutdown condition. Access to the plant during this time would be by boat and/or helicopter.

In the event of a loss of water from Dardanelle Reservoir, this fact would be made known to plant operators by an alarm in the control room (see Section 2.4.9). Should the reservoir level continue to drop, plant shutdown would be required using the emergency cooling pond.

2.5 Geology, Seismology and Foundation Engineering

In our reviews and those by our United States Geologic Survey (USGS) and NOAA advisors of the ANO-1 and ANO-2 Preliminary Safety Analysis Reports (PSAR), we concluded that the applicant's analysis constituted an adequate appraisal of the geological and seismological aspects of the site, and the applicant's proposed values of ground acceleration were adequate. These conclusions are still valid and applicable to this evaluation.

Site foundation investigations such as borings, permeability tests, laboratory test, rock anchor pull tests, and geophysical explorations have been sufficient to adequately define foundation conditions. Following is a brief summary of the geologic, seismic, and foundation engineering aspects of the site.

2.5.1 Geology

The site is located in the Arkansas Valley section of the Ouachita Physiographic Province. The Arkansas Valley is a gently undulating, east-west trending plain 25 to 35 miles wide, extending from Searcy to Fort Smith. Many long, sharp, east-west ridges and several broad topped hills rise above the general level of the valley. The Arkansas Valley is a part of an extensive outcrop area of Paleozoic sedimentary rock consisting mainly of Pennsylvania sandstones and shales. A few scattered igneous bodies are present in the region. Except for areas of high erosion, the bedrock is covered by a thin mantle of residual soil.

The Arkansas Valley is both a topographic and structural trough lying between the horizontal strata of the Boston Mountains to the north and the complexly folded strata of the Ouachita Mountains to the south. The northern portion of the trough is characterized by normal faults and gentle folds. The southern border contains mostly thrust faults and more pronounced folding. The central part is a combination of both of these features. The trough is made up of minor east-west oriented synclines and anticlines. The site overlies the Scranton Syncline which is adjacent to the London and Prairie Anticlines to the north and south, respectively.

No evidence of faulting at the site was found. Further, no recent faulting has been mapped in the region of the proposed site. Two prominent faults, and associated branch faults, the Long and Prairie View, are present 5 and 6 miles, respectively, from the site. They are considered tectonically inactive with the last known movements having taken place during the Cretaceous Period.

The USGS reports on ANO-1 and ANO-2 state that "There are no identifiable active faults or other recent geologic structures that could be expected to localize earthquakes in the immediate vicinity of the site." The staff concurs with this conclusion. There have been no significant changes in the geological and seismological situation since the evaluation of the site during the construction permit review for ANO-1.

The site is situated on a broad, flat topographic saddle at elevation 353 (MSL) on a peninsula formed by the creation of Dardanelle Reservoir. Drainage is generally in a southerly direction. Subsurface materials consist of from 8 to 30 feet of stiff silty clay residual soil overlying shale of the Pennsylvania McAlester formation. Below the shale, at a depth of about 106 feet is the Hartshorne formation which is a Pennsylvanian sandstone. The upper 4 to 8 feet of the shale bedrock is badly weathered. Water loss occurred at the weathered shale contact while drilling several of the site exploratory core borings. The unweathered shale is moderately jointed. The joints are tight and the shale is a competent material as demonstrated by the results of the investigations and laboratory tests.

2.5.2 Seismology

The effects of distant earthquakes could be experienced at the site. Significant earthquakes used in the seismic evaluation were the New Madrid earthquakes of 1811-1812, which occurred about 220 miles northeast of the site with a maximum epicentral Modified Mercalli Intensity of XII and an estimated site intensity of VI. In its report on seismicity NOAA states, "the major earthquake activity that could affect this site would most likely originate in the New Madrid, Missouri, region."

Other earthquakes of significance to the site are an 1882 tremor of epicentral intensity VI to VII located 48 miles west of the site; and a 1969 earthquake with an epicentral intensity of V occurring 50 miles from the site.

In its report on site seismicity for ANO-2 NOAA states, "As a result of this review of the seismological and geological characteristics of this proposed site, the Seismological Investigations Group agrees with the applicant that an acceleration of .10g on good foundation material is adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would provide an adequate basis for designing protection against the loss of function of components important to safety. The Group also agrees with the applicant that an acceleration of 0.20g is adequate for representing the ground motion from the maximum earthquake likely to affect the site. It is believed that these values would provide an adequate basis for designing protection against the loss of function of components important to safety." The staff agrees with these conclusions and therefore concurs with AP&L's use of 0.2g to characterize the Design Basis Earthquake (DBE) or Safe Shutdown Earthquake (SSE) and 0.1g to characterize the Operating Basis Earthquake (OBE).

2.5.3 Foundation Engineering

All Category I structures will be founded on unweathered shale, which we conclude will provide adequate support. Analyses revealed that several smaller Category I structures would undergo flotation if the site were flooded. To prevent this the applicant proposed to anchor these structures using rock anchors implanted into unweathered shale bedrock. He demonstrated the capability of the anchors to withstand the uplift forces by performing pull tests on anchors so placed on site.

The emergency cooling pond and its appurtenant structures are designed to withstand the effects of the SSE and retain their function. To prevent excess seepage from the emergency cooling pond through the relatively pervious zone at the weathered shale contact; AP&L sealed off this zone by overexcavating 2 feet into the shale (where it was encountered in the excavation) and backfilled with compacted, impervious clay.

The Dardanelle Reservoir with the Dardanelle Dam, which was signed and constructed under the direction of the U.S. Army Corps of Engineers, is considered an adequately reliable secondary source of cooling water as regards seismic and static stability of foundations and the earth fill embankment. Construction of the dam was completed in October 1964, and full pond was attained in February 1965. Dardanelle Lock was completed in 1970. Although the seismic capability of the dam has not been demonstrated by the applicant, earthquake loading is an important consideration in the design of Corps of Engineers' dams.

The foundations investigations and seismic and geologic analyses performed by the applicant have been adequate. The staff concludes that the geologic conditions at the site are satisfactory for the safe operation of ANO-1 using the foundation design parameters stated in the amended PSAR and FSAR.

3.0 DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.1 Conformance With AEC General Design Criteria (GDC)

The ANO-1 plant was designed and constructed to meet the intent of the AEC's GDC, as originally proposed in July 1967. The Commission published the revised GDC in 1971 just before the FSAR was filed. The applicant assessed the plant design against the revised criteria and presented this assessment in Amendment No. 25 to the FSAR. We conducted our technical review against the present version of the GDC and we conclude that the plant design conforms to the intent of the current criteria.

3.2 Classification of Structures, Components, and Systems

The applicant has classified structures, components, and systems into two basic classes, as stated in the FSAR:

"Class 1 structures, systems and equipment are those whose failure could cause uncontrolled release of radioactivity or those essential for safe reactor shutdown and the immediate and long-term operation following a loss of coolant accident. When a system as a whole is referred to as Class 1, portions not associated with loss of function of the system may be designated as Class 2."

"Class 2 structures, systems and equipment are those whose failure would not result in the uncontrolled release of radioactivity and would not prevent a safe reactor shutdown or the immediate and long term operation following a loss of coolant accident. The failure of Class 2 structures, systems and equipment may interrupt power generation."

Class 1 items were designed to withstand the Safe Shutdown Earthquake without loss of function and, using our current terminology, are seismic Category I. Class 1 items are housed in seismic Category I structures. All Class 1 items were designated by the applicant as Q List items, that is, within the scope of the Nuclear Quality Assurance Program for ANO-1.

We concluded that this method of classification meets our requirements for the seismic and quality classification of safety-related structures, components and systems.

3.3 Wind and Tornado Design

The sustained design wind speed used for the design of essential plant structures was 67 mph. Wind pressure, shape factors, gust factors, and variation of winds with height were determined in accordance with the American Society of Civil Engineers paper ASCE 3269, "Wind Forces on Structures."

Tornado design loadings consisted of a differential pressure equal to 3 psi occurring in three seconds, followed by a calm for two seconds and a repressurization, and a lateral force caused by a funnel of wind having a peripheral tangential velocity of 300 mph and a forward progression of 60 mph.

We conclude that the design of the facility to the above wind and tornado loads is acceptable.

3.4 Water Level (Flood) Design

All seismic Category I structures have been designed to accommodate the PMF level established for the site at 369 feet MSL. (Factors that contribute to this level are discussed in Section 2.4 of this report). All essential seismic Category I systems and components have been either located on floors above this elevation or protected from flooding by providing adequate wall thickness, waterstops in all construction joints, sump and sump pumps to control local seepage, and by minimizing the number of openings in walls and slabs. All exterior openings and penetrations located below the PMF level have been provided with watertight doors or seals. We concluded, therefore, that protection provided for essential structures and systems against flooding is acceptable.

3.5 Missile Protection

The design of essential structures and vital components considered the effects of a spectrum of tornado-borne missiles and internally generated missiles associated with component overspeed failures and missiles that could originate from high energy system ruptures. There will be no loss of function of seismic Category I structures or components as a result of missile action.

All seismic Category I structures are designed to withstand the effects of the following spectrum of tornado-borne missiles:

a 4-inch x 12-inch by 12-foot long wooden plank travelling end-on at 300 mph; an automobile weighing two tons travelling through the air 25 feet above the ground at 50 mph and striking the structure with a contact area of 20 square feet; and a missile equivalent to a 3-inch diameter schedule 40 pipe, 10 feet long, travelling end-on at 100 mph and striking the structure anywhere over its full height.

Essential components contained in seismic Category I structures are inherently protected from tornado-borne missiles by virtue of being in a tornado resistant structure. The main steam and reactor building purge line penetrations are not located within a tornado resistant structure or protected by missile shielding, but have been designed to withstand tornado winds and pressure drop loadings. We have reviewed the applicant's missile impact analysis for these penetrations (main steam line and purge lines) and conclude that missile impacts on these penetrations would not cause a LOCA or prevent safe plant shutdown.

We conclude that the missile protection provided for ANO-1 is acceptable.

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

The design criteria used by AP&L for determining the break locations and break orientations for the reactor coolant pressure boundary are somewhat different but not inconsistent with the AEC staff position. Piping systems operating at 300 psig or greater were considered as having the potential energy for dynamic effects upon postulated rupture. Both longitudinal and circumferential pipe breaks were assumed to occur at any location. Static analyses instead of dynamic analyses were performed. The stresses in the pipe restraints were limited to 0.9 times the yield stress. In addition, to provide more complete protection against pipe whip consistent with our requirements, the applicant has agreed to perform a dynamic analysis where the loading conditions are the most critical to assure that the restraint stresses do not exceed the limits of 0.5 times the uniform ultimate strain, and to confirm that the equivalent static analyses used for the design are adequately conservative. In locations where this analysis shows additional piping protection would be required by the AEC staff position but is not practicable in the present design, a supplemental in-service inspection program will be required, which consists of 100 percent inspection of the cir-cumferential welds at these locations during each inspection period as specified in Section XI of the ASME Code.

We find these approaches for protection against pipe whip acceptable with adequate implementation of the analysis and inspection indicated above.

3.7 Seismic Design

3.7.1 Seismic Input

The applicant's seismic design response spectra curves were reviewed and approved by the staff prior to the issuance of the construction permit for ANO-1. The modified earthquake time histories used for component equipment design have been conservatively adjusted in amplitude and frequency to envelope these response spectra. We and our seismic consultants conclude that the seismic input criteria used by the applicant are acceptable.

3.7.2 Seismic Analysis

Modal response spectra multi-degree-of-freedom analysis (response spectra) and normal mode-time history analysis (time history) methods are used for the analysis of all Category I structures, systems and components. The vibratory motions and the associated mathematical models account for the soil-structure interaction and the coupling of all coupled Category I structures and components. Governing response parameters have been combined by the square root of the sum of the squares to obtain the modal maximums when the response spectra method is used. The absolute sum of responses is used for closely spaced frequencies. The horizontal and vertical floor spectra of seismically induced vibratory motions used for design and test verification of structures, systems and components were generated by the time history method. Torsional loads have been adequately accounted for in the seismic analysis of the Category I structures. Vertical ground accelerations were assumed to be 2/3 of the horizontal ground accelerations and the horizontal and vertical effects were combined simultaneously. Constant vertical load factors were employed only where analysis showed sufficient vertical rigidity to preclude significant vertical amplifications in the seismic system being analyzed.

We and our consultant, Nathan M. Newmark Consulting Engineering Services, have reviewed the FSAR and applicable amendments and conclude that the applicant has used acceptable seismic system and subsystem dynamic analysis methods and procedures.

3.7.3 Seismic Instrumentation Program

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure correspond to the recommendations of Regulatory Guide 1.12.

In the event of an earthquake supporting instrumentation installed on Category I structures, systems and components will provide data for the verification of the seismic responses which have been determined analytically for these items.

We conclude that the applicant's seismic instrumentation program is acceptable.

3.8 Design of Category I Structures

3.8.1 Foundations

All seismic Category I structures are founded on shale. The staff reviewed and approved the foundation conditions before the construction permit was issued. No new facts have been uncovered during construction which would affect the previous acceptance. We conclude that the foundations are structurally adequate to carry the applied loads.

3.8.2 Seismic Category I Structures

Seismic Category I structures of ANO-1 are similar to seismic Category I structures approved for previously licensed facilities. All seismic Category I structures have been designed in accordance with the ACI-318-63 code for concrete structures, AISC code for steel structures (1963) and the provisions of Regulatory Guides 1.10 and 1.15. In addition to dead, live, and DBA loads. all seismic Category I structures have been designed for the following environmental loads:

We find the applicant's methods of converting the tornado velocities into loadings and the applications of loads to be acceptable. These methods are in accordance with general practice and are similar to methods used on previously approved applications. The seismic loads were based on horizontal ground accelerations of 0.10g for the OBE and 0.20g for the SSE with vertical accelerations equal to two-thirds the horizontal ground accelerations. Other environmental loads such as snow, ice, and floods were also considered in the structural design. These loads have been treated similarly on previously licensed facilities and are acceptable to us.

For all reinforced concrete seismic Category I structures, the principal methods of analysis have been the Ultimate Strength design methods as defined in ACI 318-63 "ACI Standard Building Code Requirements for Reinforced Concrete."

We have reviewed the design criteria and the design methods for the seismic Category I structures as defined and listed by the applicant in the FSAR, and found them to be in accordance with the pertinent codes and sound engineering practice; they are therefore acceptable to us.

The stresses in seismic Category I structures are below the allowables of the codes and are acceptable.

3.8.3 Containment

The nuclear steam supply system is contained in a steel lined prestressed concrete reactor building. This containment structure is post-tensioned by means of BBRV tendons of 186 1/4-inch wires each. Each wire is "button-headed" at both ends of the tendon to anchorage hardware. The design of this structure is substantially similar to previously licensed reactor buildings. Other buildings which have been licensed have six buttresses anchoring horizontal tendons which span about one third of the building circumference. The ANO-1 building has three buttresses which anchor horizontal tendons spanning two thirds of the building circumference. The vertical and dome tendons are essentially the same for the two designs. On the basis of our evaluation of the experimental and analytical information furnished by the applicant, we conclude that this tendon system is acceptable.

The steel liner design is typical for this type of containment. The choice of the materials, the arrangement of the anchors, and the design criteria applied are similar to those evaluated for previously licensed plants. We conclude that the liner design is acceptable.

The containment is designed in accordance with the applicable sections of the ACI 318-63 code for concrete and the pertinent sections of the ASME Pressure Vessel Code, Section III, Division 1, 1965 Edition for the liner.

The containment is designed for dead, live, Design Basis Accident, OBE, SSE, and environmental loads. Its structural design loads and design criteria are very similar to those previously approved for other facilities and are acceptable to us.

Stresses in the shell, penetrations, and foundation resulting from static and dynamic loads were calculated by means of a finite element computer program and were found to be acceptably below the allowables of the codes. We conclude that this method of analysis, which is in accordance with general engineering practice, is acceptable.

3.8.4 Interior Structure

The design basis for the reactor building internal structures is discussed in detail in Appendix 5.A and in Section 5.2.1 of the FSAR. The results of differential pressure computations by the applicant are listed below.

Internal Structure	Differential Pressure on Structure, psid
Reactor Cavity walls below level of refueling canal-reactor vessel seal	230
Steam generator and pressurizer compartment walls	16
Operating floor	0.8

The cavity walls are designed to withstand a jet force coincident with the pressure load resulting from pipe rupture. Loading combinations and allowable material stresses are listed in FSAR Appendix 5A. Local yielding under pipe rupture loads is allowed, with the ductility factor limited to 3.

The reactor building primary concrete shield design considered the effects of radiation generated heat. The design of the reinforcement at this location is based on the results of a computer finite element program and is in accord with the design criteria presented in Appendix 5.A.3.1 of the FSAR.

We conclude that the use of these design methods, which are in accordance with general practice, are acceptable. Moreover, we have performed similar calculations of internal structure differential pressure and our results are in reasonable agreement with those of the applicant. We, therefore, conclude that the design pressures of the internal structures are acceptable.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

The applicant has designated Oconee-1 as the prototype plant from which preoperational vibration test results are applicable in evaluating the design adequacy of the reactor internal structures of ANO-1. Thus, only the confirmatory test in accordance with Regulatory Guide 1.20 will be conducted. The vibration tests of Oconee-1 were completed recently. The results of these tests demonstrated the adequacy of the reactor internals for Oconee-1 and it has been licensed. However, to qualify Oconee-1 as a prototype plant, B&W has submitted Topical Report BAW-10039, an interpretation of the Oconee-1 internals vibration behavior using the test data and vibration theories. This report was reviewed by the Regulatory staff and we concluded that it adequately qualified Oconee-1 as the prototype plant.

We find that the above program of preoperational vibration testing of reactor internals is acceptable.

The reactor internals of ANO-1 were designed to withstand the dynamic effects of a simultaneous occurrence of loss-of-coolant due to coolant pipe rupture near the nozzle and the Safe Shutdown Earthquake. The applicant's analyses in support of these design features are contained in referenced Topical Reports (1) BAW-10008-1 - Rev. 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquakes," and (2) BAW-10035, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation." We have evaluated the analyses in these reports and conclude that the ANO-1 reactor internals are adequately designed to withstand a loss of coolant accident coincident with the occurrence of an SSE.

Vibration Operational Test Program (Piping)

A series of preoperational functional tests will be performed on piping systems both inside and outside the reactor coolant pressure boundary, in accordance with Paragraph I-701.5.4 of ANSI B31.7 Nuclear Power Piping Code. This code requires that piping shall be arranged and supported to minimize vibration and that the designer shall make appropriate observations under startup and initial operating conditions to assure that vibration is within of the codes and standards specified in 10 CFR 50.55a or Regulatory Guide 1.26 as appropriate. All components are designed to sustain normal operating loads, anticipated operational occurrences and the operational basis earthquake (1/2 SSE) within the stress limits of the code specified. In addition, Quality Group A components are designed for a limiting primary stress of two-thirds of ultimate strength for the combination of design loads plus SSE and pipe rupture loading. Quality Groups B and C components are designed to sustain the SSE loading within stress limits comparable to those associated with the emergency operating condition of current component codes. We consider conformance with these codes and standards an acceptable basis for complying with AEC General Design Criterion 1.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

The reactor protection system, engineered safety feature circuits and the emergency power system were designed to meet seismic Category I design criteria. A seismic qualification program was implemented for confirming that all seismic Category I instrumentation and electrical equipment will operate properly during an SSE and post-accident conditions. Also, the support structures for this equipment have been designed to withstand the SSE. The operability of the instrumentation and electrical equipment has been demonstrated by testing. The design adequacy of the supports has been demonstrated by analysis or testing. The applicant has relied upon B&W Topical Report BAW-10003, "Qualification Testing of Protection System Instrumentation" to establish this qualification. On the basis of our review of that report and other information provided by the applicant, we have concluded that the seismic design of this equipment is acceptable.

4.0 REACTOR

4.1 Summary Description

The design of the B&W reactor for ANO-1 is similar to the design of other pressurized water reactors that we have recently approved for operation, and is nearly identical to Duke Power Company's Oconee-1 reactor The core consists of 177 fuel assemblies having 208 fuel rods each; the design heat output of the core is 2568 MWt. A unique feature of the B&W design is internal vent valves which minimize steam binding in the event of a loss-of-coolant accident (LOCA). Full and part length control rods, dissolved boron, and burnable poison rod assemblies (BPRA) are used for reactivity control.

4.2 Mechanical Design

4.2.1 Fuel

The ANO-1 reactor fuel elements, designed and fabricated by B&W, will employ Zircaloy-clad fuel rods containing uranium dioxide pellets. All fuel rods are pre-pressurized with helium gas and are similar to those approved for use in Oconee-1 except for the density of the fuel pellets. The Oconee-1 fuel prior to operation was 93.5% of theoretical density (TD), whereas the fuel for the first cycle of ANO-1 is 92.5% TD for zone 1 of the core and has been resintered to 95-95.5% TD for fuel zones 2 and 3.

Fuel elements designed and fabricated by another manufacturer and used in other power plants have experienced physical changes (due to fuel densification) that could affect core operating conditions. The conditions of operation for these facilities have been restricted where necessary to maintain acceptable safety margins.

The Regulatory staff is continuing its review of the fuel densification phenomenon and the associated effects. Presently, the staff is reviewing the B&W evaluation model for fuel of the type to be used in ANO-1. After development of an acceptable model by B&W, we will determine if any operating restrictions or special inspections will be necessary for ANO-1 and, if required, we will include them in the Technical Specifications. The applicant is aware of this phenomenon and of the possibility that such imposition of operating limitations may be required. The applicant has submitted an interim evaluation of fuel densification and has proposed to change the core power imbalance limits and to reduce the overpower trip setpoint from 114% to 112% of the rated 2568 MWt.

ANO-1 is expected to be ready for fuel loading in November of 1973. We are continuing our review of fuel densification and will report on it in a supplement to this report.

4.2.2 Reactor Vessel Internals

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

For the loads calculated to result from the loss-of-coolant accident (LOCA), the Safe Shutdown Earthquake (SSE) and the combination of these postulated events the reactor internal components were designed to the criteria submitted in B&W Topical Report BAW-10008, "Reactor Internals Stress and Deflection Due to a LOCA and Maximum Hypothetical Earthquake" which was referenced in the FSAR. These criteria are consistent with comparable code emergency and faulted operating condition category limits and the criteria which have been accepted for all recently licensed plants. We find these criteria acceptable. The dynamic analyses of the ANO-1 reactor internals are discussed in Section 3.9.1, "Dynamic System Analysis and Testing."

4.2.3 Reactivity Control System

The mechanical elements of the reactivity control system have been designed to the Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code for the normal design loads of mechanical, hydraulic and thermal origin including anticipated plant transients and the OBE. Tests to determine the operational characteristics of typical prototype control rod drive mechanisms have been satisfactorily completed.

4.3 Nuclear Design

4.3.1 Nuclear Analysis

Our review of the nuclear design of the ANO-1 reactor was based on the information provided by the applicant in the FSAR and revisions thereto, discussions with the applicant and B&W, and the results of independent calculations performed for us by the Brookhaven National Laboratory.

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

The applicant has also performed analyses, using a two-dimensional PDQ computer program in conjunction with fuel cycle calculations obtained with the use of the HARMONY computer program, to provide estimates of core fuel burnups and first and second cycle and equilibrium core enrichments.

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

4.3.2 Power Distribution

Detailed three-dimensional power distribution measurements have been performed at the B&W 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.

The axial distribution of power was calculated for two conditions of reactor operation. The first condition is an inlet peak resulting from partial insertion of a Control Rod Assembly (CRA) group. This condition results in the maximum local heat flux and maximum linear heat rate. The second power shape is a symmetrical cosine which is indicative of the power distribution with xenon override rods (part length rods) withdrawn. Both of these flux shapes have been evaluated for thermal departure from nucleate boiling (DNB) limitations by the applicant. The limiting condition was found to be the cosine power distribution (peak to average power ratio, $P/\bar{P} = 1.5$) although the inlet peak shape has the larger maximum value ($P/\bar{P} = 1.7$). However, the position of the cosine peak farther up the channel results in a less favorable flux to enthalpy relationship and, therefore, the cosine axial shape has been used by the applicant to determine individual channel DNB limits.

We have concluded that the analytical methods used to calculate power distribution are adequate and that core thermal limits are conservatively based on the most restrictive power peaking factors.

4.3.3 Moderator Temperature Coefficient

The moderator temperature coefficient is slightly positive at the beginning of the initial fuel cycle due to the use of soluble boron for reactivity control. Calculations show that above 525°F, the consequences are acceptable. Since the moderator temperature coefficient at lower temperatures will be less negative (or more positive) than at operating temperatures, the applicant has stated that startup and operation of the reactor when the reactor coolant temperature is less than 525°F will be prohibited except where necessary for low power physics tests, when special operating precautions will be taken.

The maximum positive moderator temperature coefficient at full power will not exceed 0.5×10^{-4} $\Delta k/k/°F$ according to the applicant's Technical Specifications. The nominal beginning of life cycle 1 value is substantially less positive than this. The accident analyses including the calculation of clad temperature for the LOCA uses the maximum positive Technical Specification value.

We have concluded that the applicant conservatively accounts for the influence of a positive moderator temperature coefficient on various postulated accidents and adequately demonstrates its acceptability.

4.3.4 Control Requirements

To allow for the typical changes of reactivity due to reactor heatup, operating conditions, fuel burnup and fission product buildup, a significant amount of controllable excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for first and equilibrium cycles for beginning-of-life (BOL) and end-of-life (EOL) and has shown that neutron absorption means have been provided to control excess reactivity at all times. This is done through control of the concentration of soluble boron in the reactor coolant and movement of control rods. Fuel burnup and fission product buildup are partially controlled by fixed B_4C burnable poison rod assemblies (BPRA) for the longer first fuel cycle. These assemblies are used rather than increased concentrations of soluble boron to prevent the BOL moderator temperature coefficient from becoming more positive. The applicant has conservatively shown that the core can be maintained in a subcritical condition by at least 1% $\Delta k/k$ with operating boron concentrations even with the highest worth CRA withdrawn. In addition, under conditions where a cooldown to reactor building ambient temperature is required, concentrated soluble boron can be added to the reactor coolant to produce a shutdown margin of at least 1% $\Delta k/k$ with all the control rod assemblies withdrawn from the core.

On the basis of our review, we have concluded that the applicant's assessment of reactivity control requirements over the core lifetime 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 for all conditions.

4.3.5 Stability

The basic instrumentation for monitoring the nuclear power (neutron flux) level and distribution in the ANO-1 core is the same in principle as for all PWR plants recently licensed for operation. Primary reliance is placed on four axially split, out-of-core neutron detectors that are spaced approximately 90° apart around the reactor pressure vessel. Also, 52 assemblies of self-powered neutron detectors are available for in-core mapping. Each in-core assembly can measure local neutron flux at seven elevations in the core. Normally, the output of these detectors will be read out through the plant computer; however, a backup readout system is provided. The applicant has provided for availability of these detectors for monthly calibration of the out-of-core detector tilt factor. Test results showing that these in-core detectors have a rated lifetime in excess of 5 years and a precision of ± 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 if the power distribution mapping capability provided by the in-core detectors is utilized 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 have been 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 Oscillations - 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 analyses further indicated that axial xenon oscillations (which are slow changes taking place over several hours) can be controlled by operator control of the position of the eight part-length (axial power shaping) rods. In addition, the operator of the prototype plant, Oconee-1, has agreed to perform tests during the initial startup of that plant to demonstrate the as-built stability of this core design against xenon-induced reactivity fluctuations.

As added assurance that power maldistributions will not go undetected should they occur, the Technical Specifications will (1) require appropriate axial and radial power distribution monitoring and control measures to be in effect, and (2) limit the BOL positive moderator coefficient.

On the basis of our review and with the restrictions to be imposed by the Technical Specifications we conclude that the nuclear design is acceptable.

4.4 Thermal Hydraulic Design

The thermal hydraulic design of the ANO-1 is identical to the design of Oconee 1 which was previously reviewed and found acceptable. However, since the applicant does not propose to validate operation of the plant in a single loop configuration (i.e., with both pumps in one loop running while both pumps in the other loop are idle) the Technical Specifications will prohibit single loop operation.

The applicant is evaluating systems and equipment which are available to monitor the reactor coolant system for the presence of loose parts during operation; AP&L will be required to adopt a suitable system before ANO-1 is licensed.

5.0 REACTOR COOLANT SYSTEM

5.1 Summary Description

ANO-1 uses a B&W 2-coolant loop nuclear steam supply system. The two 36-inch reactor outlet lines run to the tops of the 73-feet tall B&W once through steam generators; the reactor coolant passes downward through the vertical tubes of the steam generator. At the bottom of each steam generator there are two 28-inch outlet lines which go to separate reactor coolant pumps, and from the pumps back to the reactor vessel. An electrically heated and spray cooled pressurizer is piped to one of the reactor outlet lines and two nitrogen-pressurized core flooding tanks are piped directly to the reactor vessel. In the reactor vessel the coolant travels down through an annulus at the outermost diameter and then up through the reactor core to the outlet plenum above. The cylindrical encasement of the reactor outlet plenum is fitted with eight 14-inch swing check valves which are closed during normal operation but can open under accident conditions to relieve pressure from the outlet plenum to the loop inlet annulus. The reactor coolant pipe loops have no valves in the main stream. The reactor coolant pumps are fitted with a shaft clutch which prevents rotation in the reverse direction. In addition, the pump motor shaft has a heavy flywheel to prolong coastdown.

In all important aspects, the reactor coolant system of ANO-1 is the same as that previously approved for the Oconee 1 plant. The principal components, physical sizes, materials of construction, basic design codes, and operating conditions are the same for ANO-1 as for Oconee 1 with the following exceptions:

(1) Oconee 1, as a class prototype is more extensively instrumented for additional operational tests.

(2) ANO-1 uses an Allis Chalmers reactor coolant pump motor and flywheel which differs from the Westinghouse pump motor and flywheel used in Oconee 1.

On the basis of our evaluation of the Oconee 1 system and the specific evaluation of the ANO-1 pump flywheel, we conclude that the overall design of the reactor coolant system for ANO-1 is acceptable.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design Criteria, Methods, and Procedures

The reactor coolant system has been designed to appropriate codes to withstand normal design loads including anticipated plant transients and the Operational Basis Earthquake within acceptable stress limits as follows:

The steam generator, pressurizer, and reactor coolant pump casings have been designed to Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, including the Summer 1967 Addenda. Safety and relief valves are in accordance with the requirements of Article 9 of the above edition and addenda of Section III.

The design, fabrication, inspection and testing of the reactor coolant piping including the pressurizer surge line and spray line are in accordance with the USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February 1968, including the June 1968 Errata. Nondestructive examination requirements for reactor coolant system pumps and valves are given in Table 4-12 of the FSAR. These examinations include radiography of castings, ultrasonic testing of forgings, dye penetrant examination of pump and valve body surfaces, and radiography of circumferential welds. The program followed by AP&L upgrades the nondestructive examination of pumps and valves within the reactor coolant pressure boundary to essentially the level now required by the ASME Code for Pumps and Valves for Nuclear Power.

The design, fabrication and inspection criteria discussed above are consistent with those accepted for all recently reviewed plants of this type and we find them acceptable.

Components of the reactor coolant system have also been designed to withstand the loads calculated to result from the Safe Shutdown Earthquake, the Design Basis Accident, and the combination of these postulated events. Strain limits for the reactor coolant system components under these combined loads correspond to an elastically calculated stress limit of not greater than 2/3 of the ultimate tensile strength. We conclude that these design limits are acceptable.

5.2.2 Overpressure Protection

Pressure safety and control are provided by two code (ASME-Section III) safety valves and one electromagnetic relief valve. The three valves are located on separate nozzles on top of the pressurizer. Effluent from the valves is directed into the reactor coolant flash tank located inside containment.

Each safety valve has a relief capacity of 300,000 lb/hr at 2500 psig; the relief valve has a capacity of 100,000 lb/hr at 2255 psig. The combined 600,000 lb/hr capacity of the code safety valves is such that the consequences of a rod withdrawal accident which begins at low power and is terminated by a high pressure trip are acceptable. The applicant has confirmed, by Amendment 23, 27 and 28, that the mounting and support systems for the reactor coolant and main steam safety and relief valves are designed to accept full discharge loads.

5.2.3 Operability of Active Pumps and Valves

The applicant has identified the active valves within the reactor coolant pressure boundary, i.e., valves whose operation is relied upon to shut the plant down safely 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 conducted component test programs, supplemented by analytical predicative methods, that provide additional assurance that the capability of these active valves (a) to withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity and (b) to perform the "active" function (i.e. valve closure or opening) is confirmed under conditions and combinations comparable to those expected when a safe plant shutdown is to be effected or the consequences of an accident are to be mitigated.

Based on the tests conducted on "active" valves, and analyses performed to demonstrate capability of operation under imposed loadings, we believe that these tests and analyses provide an adequate basis for evaluation and a reasonable assurance of valve operability to perform their design safety function. However, the applicant has further agreed to remain cognizant of industry efforts to identify potential valve operability generic problems and to incorporate, if necessary, appropriate modifications that could improve or correct active valve performance under conditions required for their intended design safety function.

5.2.4 Fracture Toughness

To assure that ferritic materials of pressure-retaining components of the reactor coolant pressure boundary will exhibit adequate fracture toughness under normal reactor operating conditions, during system hydrostatic tests, and during transient conditions to which the system may be subjected, we have reviewed the materials testing programs used in plant fabrication and the operating limitations proposed by AP&L.

Acceptance testing for ferritic materials has been performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1968 Edition). Drop-weight NDT data as well as Charpy V-notch energy curves have been obtained for the plates and major forgings in the reactor vessel.

To establish operating pressure and temperature limitations during startup and shutdown, and during hydrostatic testing of the reactor coolant system, AP&L has followed the recommendations of Appendix G, "Protection Against Non-Ductile Failure," of the recently revised ASME Code, Section III, fracture toughness rules (Code Case 1514). The applicant has submitted specific heatup, cooldown and hydrostatic test limitation curves which meet the current fracture toughness requirements; these curves will be made part of the Technical Specifications.

We conclude that the planned operation of the reactor coolant system will assure adequate margins of safety.

5.2.5 Sensitized Stainless Steel

If austenitic stainless steel is sensitized, it has an increased susceptibility to stress corrosion cracking.

The applicant has avoided significant sensitization of all non-stabilized austenitic stainless steel within the reactor coolant pressure boundary through materials selection and control of heat treating processes.

Whenever stainless steel components were welded to ferritic material, inconel "buttering" of the ferritic material followed by a stress-relief treatment preceded the joining of the two components with inconel weld metal.

We conclude that the measures taken to prevent significant sensitization of austenitic stainless steel during the fabrication period were acceptable.

5.2.6 Pump Flywheel Integrity

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

The applicant in the FSAR, Amendment No. 25 and Amendment No. 28 has furnished information on the materials, design, fabrication, inspection and surveillance program for the pump flywheels which is considered to be in accordance with Regulatory Guide 1.14, Reactor Coolant Pump Flywheel Integrity. We conclude that the materials, design, fabrication, inspection and surveillance program for the flywheels are acceptable.

5.2.7 Reactor Coolant Pressure Boundary Leakage Detection System

Coolant leakage within the reactor containment may be an indication of a small through-wall flaw in the reactor coolant pressure boundary. The leakage detection system installed for the reactor coolant pressure boundary is described in Amendment Nos. 23, 25, 26, 27 and 29. The system includes diverse leak detection methods, has sufficient sensitivity to measure small leaks, and is provided with suitable control room alarms and readouts. The major components of the system are the containment atmosphere particulate and gaseous radioactivity monitors, and level indicators on the containment sump. Indirect indication of leakage can be obtained from the containment humidity, pressure and temperature indicators. We conclude that the leakage detection system has the capability to detect leakage from small through-wall flaws in the reactor coolant pressure boundary.

5.2.8 Inservice Inspection Program

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

The applicant has stated, in the FSAR, Amendment No. 26 that the inservice inspection program for the reactor coolant pressure boundary will comply with Section XI of the ASME Boiler and Pressure Vessel Code, "Rules for In-Service Inspection of Reactor Coolant Systems", 1970 edition, to the extent practical.

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

We conclude that the access provisions and AP&L's planning for inservice inspection are acceptable since the provisions of the AEC Guideline, "Inservice Inspection Requirements for Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspection," (January 31, 1969) have been satisfied.

5.2.9 Reactor Vessel Material Surveillance Program

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

The applicant has stated in FSAR, Amendment No. 23 that the material surveillance program will comply with the proposed AEC § 50.55a Appendix H, "Reactor Vessel Material Surveillance Program Requirements," and ASTM E 185-70. The AP&L program is acceptable with respect to the number of capsules, number and type of specimens, withdrawal schedule, and retention of archive material. We conclude that the proposed program will adequately monitor neutron radiation induced changes in the fracture toughness of the reactor vessel beltline material.

5.3 Failed Fuel Detection Instrumentation Systems

The applicant has stated in Section 9.1.2 of the FSAR that the letdown flow from the reactor coolant system will be continuously monitored by a gamma sensitive scintillation detector for gross and iodine gamma activity. We have concluded that this instrumentation, which is of the same type previously approved for other PWR plants, is acceptable for detection of fuel failures in the reactor core.

6.0 ENGINEERED SAFETY FEATURES

6.1 General

The engineered safety features for ANO-1 consist of the reactor building, its associated ventilation and isolation systems, emergency core cooling system, spray system, heat removal system, combustible gas control system, the emergency feedwater system, and the emergency power system. The instruments and controls for these engineered safety features are discussed in Section 7.0 of this report and the emergency power system in Section 8.0

6.2 Containment Systems

6.2.1 Containment Functional Design

The ANO-1 containment structure (reactor building) is a free-standing steel-lined, prestressed concrete structure with a net free volume of approximately 1,800,000 ft^3. The structure houses the reactor coolant system including the reactor, pressurizer, coolant pumps and steam generators, as well as certain components of the plant's engineered safety features systems. The containment structure is designed for an internal pressure of 59 psig and a temperature of 286°F.

The applicant has described the results and methods used to analyze the containment pressure response for a number of design basis loss-of-coolant accidents (LOCA). The applicant has analyzed the containment for a spectrum of both hot leg and cold leg breaks, up to and including the double-ended rupture of the largest reactor coolant line to determine the containment pressure responses. Minimum containment cooling was assumed, i.e., two of the four fan-coolers of the reactor building cooling system, and one of the two spray trains of the reactor building spray system were assumed to operate. The core reflood energy and steam generator stored energy were included, as appropriate, in these analyses. As discussed below, we have reviewed the results of these analyses, and verified by our analyses that the calculational methods used by the applicant were conservative.

The applicant has analyzed the containment pressure response from postulated LOCA's in the following manner. Mass and energy release rates were calculated using the FLASH and CRAFT computer codes. These mass and energy addition rates were then used as inputs to COPATTA, which is a computer program used by the applicant to calculate the containment pressure response.

The CRAFT computer code was used by the applicant to determine the mass and energy release rates to the containment for cold leg breaks during the blowdown phase of the accident, i.e., the phase of the accident during which most of the energy contained in the reactor coolant system, including the stored energy in the coolant metal and the core is released to the containment. The applicant has, however, increased the energy release rate to the containment by extending the time that the core would remain in nucleate boiling beyond the expected time, i.e., the time during which the energy release rate from the core is highest. Based on this method of calculation, the core would transfer more heat to the containment for containment analysis than for emergency core cooling analysis. Since this additional energy release from the core will increase the containment pressure, the calculation is conservative. The CRAFT computer code has been approved by the AEC for calculating energy release during a LOCA.

The applicant has identified the 7.0 ft^2 split break at the pump suction as the cold leg break that results in the highest containment pressure, 51 psig. The largest break (about 8.6 ft^2) results in a peak calculated pressure of 50 psig. We have analyzed the containment pressure response for the 7 ft^2 rupture in the suction leg of the reactor coolant system using the CONTEMPT computer code which includes the energy addition to the containment from the steam generators and have calculated a peak containment pressure about 52 psig as compared to the applicant's calculation of 51 psig using the COPATTA computer code. To determine the mass and energy release to the containment, we used the applicant's mass and energy release rates calculated by CRAFT and the mass and energy release rates during the reflood phase of the accident determined by our computer program FLOOD 2.

Blowdown mass and energy releases for hot leg breaks were calculated by the applicant using the FLASH computer code. The FLASH computer code is an earlier version of the CRAFT computer code with more simplistic modeling of the reactor coolant system. Using FLASH, the applicant calculates more conservative mass and energy release rates than CRAFT during blowdown. On this basis we conclude that the mas, and energy release rates calculated for hot leg breaks is conservative using the FLASH computer code and is acceptable. The applicant's analysis indicates that a 5.0 ft^2 break of the hot leg results in the highest containment pressure of 53 psig. The largest hot leg break (14 ft^2) results in a peak containment pressure of 51 psig.

The applicant has also analyzed the containment pressure response due to postulated failures of the main steam line. The applicant has conservatively assumed that the energy in a steam generator was instantaneously released and has not taken credit for the energy removal capability of the available structural heat sinks. With these assumptions the applicant calculated a peak containment pressure of 36 psig for this accident.

We have evaluated the containment system in comparison to the Commission's General Design Criteria stated in Appendix A to 10 CFR Part 50 of the Commission's Regulations and, in particular, to

Criteria 16 and 50. As a result of our evaluation, we have concluded that the calculated pressure and temperature conditions resulting from any design basis LOCA will not exceed the design conditions of the containment structure. The highest calculated containment pressure and temperature were 53 psig and 280°F, respectively. The containment design pressure of 59 psig provides an 11% margin above the peak calculated pressure. We conclude that the maximum containment pressure is correctly calculated to be below the design pressure and that there is sufficient margin between the maximum containment pressure and the design pressure of the containment structure.

The pressure response within the containment interior compartments, such as the reactor vessel cavity and the steam generator compartments during LOCA are discussed in Section 3.8.4 of this report.

6.2.2 Reactor Building 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 LOCA. Any of the following combinations of equipment will provide adequate heat removal capability:

(a) Both spray trains of the RBSS,

(b) All four fan-cooler units of the RBCS, and

(c) One spray train of the RBSS and two fan-cooler units of the RBCS.

The RBSS serves only as an engineered safety feature and performs no normal operating function. It is a Category I system consisting of redundant piping, valves, pumps and spray headers. All active components of the RBSS are located outside the reactor building. Missile protection is provided by direct shielding or physical separation of equipment. The reactor building sump is covered by a protective grating to keep debris out of the sump. In addition, the recirculation line inlets in the sump are protected by a screen assembly designed to prevent debris that could clog the spray nozzles from entering the spray system.

The RBSS includes a system for injecting sodium thiosulfate and sodium hydroxide solutions with the borated spray water to accelerate removal of fission product iodine from the containment atmosphere in the event of a postulated LOCA. The sodium thiosulfate is the principal removal agent; the sodium hydroxide will raise the pH of the spray water into the alkaline range. Both solutions are added by gravity draining into the spray pump suction piping.

A high reactor building pressure will cause the engineered safety features (ESF) actuation system to automatically place the RBSS in operation (see Section 7 of this report). The spray pumps and valves can also be operated manually from the control room. The spray pumps will initially take suction from the borated water storage tank (BWST). When the water in the BWST reaches a low level, a half hour or more after a LOCA, the spray pump suction is manually transferred to the reactor building sump to initiate the recirculation phase. The applicant's analysis indicates that sufficient water will have been delivered to the containment at that time to provide the required net positive suction head to the spray pumps.

The Reactor Building Cooling System (RBCS) is used during both normal and accident conditions. Four, equal capacity fan-cooler units are provided. Each fan-cooler unit contains separate normal and emergency cooling coils and a single speed fan. During normal plant operation, water from the plant main water chillers is circulated through the normal cooling coils. Under accident conditions, following receipt of an engineered safety features actuation signal, an air bypass damper will open to allow the steam-air mixture to bypass the return air ducts and be diverted to the emergency cooling coils. The bypass damper is designed to fail open. For emergency cooling, heat will be rejected to the service water system. The RBCS can also be operated manually from the control room.

The RBCS is a seismic Category I system. The housings for the fan-cooler units and the supply ducts are designed to withstand an inward pressure differential of 2 psi. They are provided with pressure relief valves that are designed to actuate at a pressure differential of 10 inches of water, and are sized to prevent a differential pressure greater than 2 psi from occurring. The cooling units are located outside the secondary concrete shield for missile protection, at an elevation that precludes flooding. The RBCS equipment is accessible for periodic testing and inspection during normal plant operation.

We have concluded that the reactor building heat removal systems, namely the RBSS and the RCBS, are acceptable because they are designed to provide adequate assurance of operability under accident conditions and satisfy our criteria for redundancy and independence.

6.2.3 Containment Isolation Systems

The Reactor Building Isolation System is designed to isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of closed systems and isolation valves, is provided so that no single valve or piping failure can result in loss of containment integrity. Reactor building penetration piping up to and including the external isolation valve is designed as seismic Category I equipment, and is protected against missiles which could be generated under accident conditions.

Reactor building isolation will automatically occur on receipt of an ESF actuation signal of high reactor building pressure (4 psig). All fluid penetrations not required for operation of the engineered safety features equipment will be isolated. Remotely operated isolation valves have position indication in the control room.

We have reviewed the containment isolation system for conformance to General Design Criteria 55, 56 and 57. We conclude that the system meets the intent of the General Design Criteria.

6.2.4 Combustible Gas Control Systems

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

The applicant's analysis of post-LOCA hydrogen generation, which is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations In Containment Following a Loss-Of-Coolant Accident," indicates that the hydrogen concentration in the containment would not reach the lower flammability limit of 4 volume percent (v/o) until about 17.5 days after postulated LOCA. The concentration limit specified by the applicant for actuating the hydrogen control equipment is 3.5 v/o which is calculated to occur about 11.5 days after a LOCA. We have performed independent calculations to determine that the dose resulting from the hydrogen purge, combined with the calculated LOCA dose, is less than the limits established by 10 CFR Part 100. We calculated thyroid and whole body doses of less than 1 Rem at the LPZ for the course of the accident situation due to the hydrogen purge; we consider this acceptable for ANO-1.

The hydrogen purge system is designed as a seismic Category I system. Two redundant purge trains are provided. Either purge train is capable of maintaining the hydrogen concentration in the containment below the control limit of 3.5 v/o; a purge exhaust fan has a 50 cfm capacity. Each purge train is powered from a separate engineered safety feature electrical bus.

An electric heater, HEPA filter, and activated charcoal filter are provided upstream of each purge exhaust fan to dry and filter the purge air. Provision has been made in the system design for cooling a filter if the temperature approaches the ignition temperature of the charcoal. Each purge exhaust line is equipped with a hydrogen sampler, located between the reactor building isolation valves, to permit monitoring of the hydrogen concentration without operating the hydrogen purge system.

We have reviewed the hydrogen purge system for conformance with the recommendations of the Supplement to Regulatory Guide 1.7 and conclude that the system is acceptable.

6.2.5 Penetration Room Ventilation System

The Penetration Room Ventilation System (PRVS) is designed to collect and process potential reactor building leakage to the penetration room under post-accident conditions to minimize the release of radioactive materials to the environment. The PRVS is designed as a seismic Category I system and is capable of withstanding a single failure without loss of function. The system consists of redundant fan-filter trains. Each train is powered from a separate engineered safety features bus.

The PRVS is not operated during normal plant operation. In the event of an accident an engineered safety feature actuation signal will automatically place a fan-filter system in operation by starting the fan and opening the butterfly valve downstream of the filter assembly. The system can also be remotely operated from the control room. The PRVS equipment is accessible for periodic testing and inspection during normal plant operation. Based on our review we conclude that the PRVS is acceptable.

6.2.6 Leakage Testing Program

Leakage testing of the reactor building and associated systems is intended to provide initial and periodic verification of the leaktight integrity of the containment. The reactor building and its components have been designed so that periodic integrated leakage rate testing can be conducted at a test pressure corresponding to the design pressure of 59 psig. Penetrations, including personnel and equipment hatches and airlocks, and isolation valves, can also be individually leak tested at 59 psig.

We conclude that the design of the reactor building and associated systems will permit leakage rate testing in compliance with the AEC proposed "Reactor Containment Leakage Testing for Water Cooled Power Reactors," § 50.54(o), Appendix J, published in the Federal Register on August 27, 1971.

6.3 Emergency Core Cooling System

6.3.1 General

In 1971 the Regulatory staff reevaluated the theoretical and experimental bases for predicting the performance of emergency core cooling systems (ECCS), including new information obtained from industry and AEC research programs in this field. As a result of this reevaluation, we developed interim acceptance criteria for emergency core cooling systems for light-water power reactors. These criteria are described in an Interim Policy Statement issued on June 25, 1971, and published in the Federal Register on June 29, 1971 (36 F.R. 12247). By letter dated August 11, 1971, the Regulatory staff informed AP&L of the additional information that would be required for our evaluation of the performance of the ANO-1 ECCS in accordance with the Interim Policy Statement. B&W provided a revised analysis of the ANO-1 ECCS performance in Topical Report BAW-10034 titled "Multinode Analysis of B&W's 2568-MWt Nuclear Plants During a Loss-of-Coolant Accident" dated October 1971. The analysis was performed using the B&W ECCS evaluation model in conformance with the Interim Policy Statement, Appendix A, Part 4. In the analysis it was assumed that a LOCA occurs during operation at 102% of rated power (2568 MWt).

Subsequent to the staff's review of the analyses presented in BAW-10034, several additional topics associated with ECCS performance were identified during the staff's review of the operating license application for Duke Power Company's Oconee Unit 1 (Docket 50-269).

These topics included: (1) the reflooding analysis associated with a LOCA; (2) the analysis of small breaks in the primary cooling system; and (3) the analysis of a break in the core flooding tank (CFT) line.

The staff has reviewed the ANO-1 ECCS and, based on the similarity to the Oconee ECCS, find the information and evaluation of the Oconee ECCS performance applicable to ANO-1.

6.3.2 System Description

The ANO-1 ECCS consists of a high pressure injection system, a passive injection system employing core flooding tanks (CFT), and a low pressure injection system with capability of long term recirculation of emergency core coolant to heat exchangers outside containment. Various combinations of these systems are employed to assure core cooling for the complete range of break sizes.

The high pressure injection system includes three pumps, each capable of delivering 500 gpm at 600 psig into the reactor coolant inlet lines. One pump will provide the required minimum flow for the high pressure injection system. The high pressure injection pumps are located in the auxiliary building adjacent to the containment. A boric acid solution from the borated water storage tank (BWST) will be provided to the suction side of the high pressure pumps during ECCS operation. During normal reactor operation, the high pressure injection system is aligned to recirculate reactor coolant for purification and to supply seal water to the reactor coolant circulation pumps. The high pressure injection system would be actuated if the reactor coolant system pressure were to go below 1500 psig, or if the reactor building pressure were to rise above 4 psig. Automatic actuation switches the system from the normal to the emergency operating mode. One of the three high pressure pumps is normally in operation. The high pressure injection system is designed to withstand a single failure of an active component without a loss of function.

The two core flooding tanks (CFT) of the passive injection system are located in the reactor building beyond the secondary shield wall which encloses the reactor coolant loops. Each CFT has a total volume of 1410 ft^3, will contain a nominal stored borated water volume of 1040 ft^3 and will be pressurized to 600 psig with nitrogen. Each CFT is connected to a separate reactor vessel core flooding nozzle by a line incorporating two check valves and a normally-open motor-operated stop valve, the latter being adjacent to the CFT. The CFT's will inject water into the reactor vessel whenever the reactor coolant pressure in the system goes below that of the CFT's (600 psig). The core flooding nozzles on the reactor vessel have been fitted with flow limiters to conserve water in the vessel in the event of a CFT line break.

The low pressure injection (LPI) system would be actuated if the reactor coolant system pressure were to go below 1500 psig or if the reactor building pressure were to rise above 4 psig. To protect the system from excessive heat up, the LPI pumps will operate in the bypass mode until the reactor coolant system pressure decreases below the pump discharge pressure.

The LPI system includes two pumps, each capable of delivering 3000 gpm at 100 psig to the reactor vessel through the CFT nozzles. Each LPI line branches inside the reactor building so that part of its flow goes to each of the two CFT nozzles. Each branch line contains a check valve to prevent back flow; in addition, each of the branch lines contains a flow limiter upstream of the check valve. These flow limiters are sized to pass little more than their rated flow of about 1500 gpm (half of the output of one LPI pump). In the unlikely event of a CFT line break, if one LPI pump fails to function the other pump will have one of its two branches connected to the intact CFT nozzle and approximately half of its flow will be delivered to the pressure vessel.

The LPI system pumps will initially take their suction from the borated water storage tank; later, during recirculation, suction will be taken from the reactor building emergency sump. The recirculation system components are redundant so as to withstand a single failure of an active or passive component without loss of function at the required flow.

All of the ECCS subsystems can accomplish their function whether supplied by emergency (onsite) power or offsite power. If there is a loss of normal (offsite) power sources, the engineered safety features would obtain their power from the emergency diesel generators which have a startup time of 10 seconds or less. The pumps and valves of the injection system will be energized before the emergency generators achieve 100% of rated voltage and frequency so as to achieve the design injection flow rate within 25 seconds.

6.3.3 Performance Evaluation

6.3.3.1 General

To analyze the performance of the ANO-1 ECCS, we have developed a set of conservative assumptions and procedures to be used in conjunction with the B&W developed codes. The B&W assumptions and procedures are described in Appendix A, Part 4 of the Interim Policy Statement (IPS) published in the Federal Register on December 18, 1971 (F. R. Vol. 36, No. 244). Topical Report BAW-10034 "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a Loss-of-Coolant Accident," October 1971, covers the performance of cores with pressurized fuel pins and with a peak linear heat rate of 18.15 kW/ft. As a result of information developed in the ECCS rulemaking hearing (Docket RM50-1), the staff requested a reanalysis of the reflooding transient using a more conservative assumption. From this analysis in BAW-10034 the 8.55 ft^2 cold leg split is determined to be the limiting case accident with a peak clad temperature of 2186°F. For comparison, the peak linear heat rate for ANO-1 is 17.63 kW/ft and the core power is 2568 MWt.

6.3.3.2 Analysis of Blowdown Period

The applicant used the CRAFT and THETA 1-B computer codes for the analysis of the blowdown phase of the transient. Using these codes, and the evaluation model specified in Appendix A, Part 4 of the IPS, the applicant provided the reevaluation of the ECCS performance in compliance with the IPS.

For the blowdown portion of the accident, we have concluded that the applicant's analyses as reported in BAW-10034 conform to the requirements specified in Appendix A, Part 4 of the IPS.

6.3.3.3 Analysis of Refill and Reflood Period

The applicant has considered the thermal behavior of the core during the refill and reflood portion of the LOCA which is explained as follows:

(1) The vessel refill flow is provided initially by the core flooding tanks, and later by the pumping systems, and is assumed to start at the end of the blowdown period. The reactor vessel is assumed to be essentially dry at the end of the blowdown period, as a result of the conservative assumption in Appendix A, Part 4, of the IPS which requires that water injected by the core flooding tanks prior to the end of blowdown is ejected from the reactor coolant system.

(2) No heat transfer in the core is assumed until the level of water rises to the bottom of the core, at which time refill is considered complete and core reflood starts.

For the 8.55 ft^2 cold leg double-ended break, blowdown is considered complete 14.6 seconds after rupture and refill to the bottom of the core is complete about 23 seconds after rupture. For the 8.55 ft^2 cold leg split, end of blowdown is considered complete 18.7 seconds after rupture and refill is complete about 26 seconds after rupture.

(3) The reflood of the core is characterized initially by a rapid liquid level rise both in the core and in the vessel annulus until enough of the core is covered to generate substantial amounts of steam. The core reflooding rate increases and peaks about 10 seconds after the end of blowdown at approximately 11 to 12 inches per second. The rate then decreases rapidly, leveling off at about 2 inches per second about 30 seconds after the end of blowdown.

(4) The steam generated in the core is assumed to flow only through the vent valves within the reactor vessel. No credit is taken for steam flow around the loop. Steam flow resistance acts to limit the rate of liquid rise in the core, but the liquid level in the annulus continues to rise rapidly until the liquid level reaches the inlet nozzle. Water from core flooding tanks and low pressure injection system is piped directly to the reactor vessel with no intervening reactor coolant system piping.

(5) The peak temperature reached in the transient for the limiting 8.55 ft^2 cold leg split occurs about 30 seconds after the break.

The staff has reviewed the applicant's reflooding analysis using a new carryover rate fraction* correlation developed by B&W during the course of the rulemaking hearing (Docket RM50-1) to account for the entrainment of reflooding water. The previous reflood analysis performed by B&W (BAW-10034) used an entrainment assumption of 20% of the inlet core flow rate. The 20% entrainment assumption was based on the FLECHT program. The staff requested a reanalysis of the reflooding transient for Oconee-1, using the new CRF correlation in its letter to Duke Power Company of November 3, 1973. Because the new carryover rate fraction correlation took many FLECHT experimental runs at different conditions into account, the staff views it as a better approach in calculating reflooding rates.

The staff has reviewed the B&W reflood code (REFLOOD) and has compared its results with those of the FLOOD 1 code (an ANC/AEC reflood program). Reflooding rates predicted by both computer programs agree to within 1%, when the REFLOOD code uses the new carryover rate function to predict the entrainment. If the old entrainment assumption of 20% is used, the flooding rates calculated by REFLOOD are higher than those predicted by FLOOD 1.

*The carryover rate fraction (CRF) is defined as the total core flow rate out of the top of the core divided by the total mass flow into the bottom of the core.

6.3.3.4 Results

B&W has recalculated the reflooding rates and heat transfer coefficients for several break locations and sizes using the new carryover rate fraction correlation. The heat transfer coefficients used in determining the peak clad temperature were determined from the FLECHT correlation presented in WCAP-7665,* with the new, lower reflooding rates. Peak cladding temperatures calculated using the new reflooding rates are higher, and remain at elevated temperatures for longer time periods. However, both the maximum clad temperatures and the percent metal water reaction calculated are within the acceptance limits set forth by the IPS on ECCS.

In response to the November 3, 1972 letter, analyses were also provided of the effect of a higher elevation axial flux peak (the previous analyses were done for an inlet flux peak). The higher elevation peak (modified cosine flux peak) resulted in a slightly lower peak cladding temperature, but a greater metal-water reaction (see results for 8.55 ft^2 split in the following table). The greater metal-water reaction is due to the extra time required for the ECCS fluid to rise to the higher elevation.

The following table summarizes the calculated results using the carryover rate fraction entrainment correlation at 102% of rated power (2568 MWt):

*WCAP-7665, PWR FLECHT Final Report, April 1971.

Cold Leg Pipe Breaks		Peak Cladding Temperature, °F	Metal-Water Reaction %	
Area	Type Break		Local	Core
8.55 ft^2	Double-Ended	2082	2.11	0.075
8.55 ft^2	Split	2186*	2.98	0.09
8.55 ft^2	Split (cosine flux peak)**	2135	4.2	0.24
5.13 ft^2	Double-Ended	2029	1.8	0.058
3.0 ft^2	Split	1728	0.046	0.01
0.5 ft^2	Split	1660	0.22	0.004
Hot Leg Pipe Breaks				
14.1 Ft^2	Split	1670	0.14	0.003

*Limiting Case

**All other cases in the table are for an inlet flux peak.

6.3.3.5 Conclusions

The use of the new carryover rate fraction correlation provides a more conservative method of predicting reflood water entrainment than 20% entrainment assumption since the use of this correlation yields lower reflooding rates, higher peak cladding temperatures and greater metal water reactions. The staff has concluded that, based on the present experimental data, the use of this more conservative approach is warranted. The staff further concludes that the ECCS performance analysis, using this more conservative approach, meets the acceptance criteria specified in the Commission's Interim Policy Statement.

6.3.4 Small Break Analysis

6.3.4.1 General

The Interim Policy Statement (IPS) concerning emergency core cooling in the event of a LOCA required LOCA analyses over the entire break spectrum. The B&W evaluation model in Part 4 of Appendix A to the IPS specified an acceptable evaluation model for break sizes from 0.5 ft^2 up to and including the double-ended severance of the largest pipe of the reactor coolant pressure boundary (the large break model). B&W submitted its evaluation model for small breaks in Topical Report BAW-10052, "Multinode Analysis of Small Breaks for 2568 MWt Plants" dated September 22, 1972; the staff has completed the evaluation of this report.

In general, small breaks result in less serious consequences than the larger design basis breaks. Moreover, the specific B&W reactor design used in ANO-1 contains internals vent valves which further mitigate the LOCA consequences, including those caused by small breaks. For cold leg breaks these vent valves prevent a hot leg loop seal from forcing the water level in the core to drop excessively due to steam binding (pressure buildup above the core). A low water level in the core could cause a core heatup transient due to degraded heat transfer. By using these vent valves to vent the reactor upper plenum to the downcomer annulus, the steam generated above the core (by loop depressurization and core heat transfer) has a low resistance path to bypass the hot leg flow path to the postulated cold leg break.

6.3.4.2 Small Break Model

The B&W procedure for analyzing the consequences of small breaks, as reported in BAW-10052, differs somewhat from that given in BAW-10034, Revision 3. These methods are similar to those used for large breaks but differ in some aspects to account for a more tranquil hydrodynamic response of the system for smaller breaks. These differences between the small break model and the large break evaluation model have been reviewed and evaluated.

The CRAFT code described in B&W Topical Report BAW-10030 "CRAFT - Description of Model for Equilibrium LOCA Analysis Program," dated October 8, 1971. CRAFT is used to simulate the hydrodynamic response for the large and small break models. The number of nodes representing the reactor coolant system for the small break model has been reduced to 11, with one node for the secondary system and one node for the reactor building. Additionally, the Redfield variable bubble rise model* described in BAW-10030 and BAW-10034 was used in all nodes for the small break model, whereas the large break model assumed a zero bubble rise model in the lower head, the core, the upper plenum and the pump suction nodes. For a large break this zero bubble rise model would be more appropriate for those nodes where good mixing occurs due to the rapid depressurization and high flow rates.

* A zero bubble rise velocity yields a homogeneous node, while increasing the bubble velocity tends to separate the water phases.

For the associated heat transfer analysis a THETA computer code* model with fewer nodes is used during the flow-controlled heat transfer transient. For one case examined, this change resulted in only a 7°F difference (in the conservative direction) between the small break THETA model and that used for large break analysis. When core flow drops below 1% of its initial value and flow no longer controls heat transfer, another heat transfer code QUENCH is used. QUENCH is a code with one axial node, one clad node, and one fuel node; it assumes heat to be transferred by either pool film-boiling or by forced convection to steam. Multiple QUENCH runs are made at various axial locations to obtain the thermal response of the fuel rod. Morgan's correlation** for pool film boiling is used for that portion of the core covered by a mixture of steam and water. This correlation is the best available for pool film boiling from vertical surfaces. It was derived from a theoretical model of the stable annular flow regime as compared to the dispersed flow film boiling regime, and it is therefore conservative in this regard. The correlation underpredicts (is more conservative than) the available data for pool film boiling from vertical surface for a variety of fluids. The Dittus-Boelter

*"THETA 1-B, A Computer Code for Nuclear Reactor Core Thermal Analysis," Idaho Nuclear Corporation Report IN-1445, February 1971.

**"Charles Davis Morgan, "A Study of Film Boiling from Vertical Surfaces," A dissertation presented to the graduate faculty of Lehigh University in candidacy for the degree of Doctor of Philosophy, Lehigh University (1965) correlation* is used for that part of the core covered by steam. In the steam-flow region the average steam flow is conservatively calculated for the fuel heatup calculation, with the fluid temperature calculated by hand.

A major difference between the small break model and the large break model is the absence in the small break model of any arbitrary core bypass of core flooding tank (CFT) injection water prior to the end-of-blowdown. The CFT bypass assumption would be unduly conservative for the small break analysis since the velocity of fluid in the downcomer (reactor inlet annulus) is too low to entrain CFT injection water and sweep it out the break.

Since the core is never completely uncovered for small breaks, the reflood analysis (which is conducted for larger breaks) is not done. The reflood analyses and the previously discussed CFT bypass assumption are, however, interrelated. A comparison of a 0.5 ft^2 break analyzed by both the large break model (without CFT bypass assumption) and the small break model was conducted by B&W. The two models agree very well yielding a peak clad temperature of only 710°F. However, when the CFT bypass assumption is imposed for the large break model the calculated peak clad temperature would rise to 1660°F.

6.3.4.3 Results and Conclusions

The results of B&W's small break analysis for plants at a core power of 2568 MWt are contained in BAW-10052, and in a December 19,

* Dittus, F. W., Boelter, L. M. K., "Heat Transfer in Automobile Radiators of the Tubular Type," Publ. in Eng., Vol. 2, n. 13, University of California, pp 443-461, 1930.

1972 letter from A. C. Thies of Duke Power Company to the AEC on the Oconee-1 prototype plant. A summary of these results is given below:

Break Size and Location	Peak Clad Temperature, °F	Long Term Cooling Established*, sec
0.5 ft^2 (pump discharge)	710	400
0.3 ft^2 (pump suction)	780	1100
0.1 ft^2 (pump suction)	826	2500
0.1 ft^2 (pump discharge)	720	3400
0.04 ft^2 (pump suction)	978	3000

All conditions of the Interim Acceptance Criteria have been met, the peak cladding temperature is well below 2300°F, there is little or no metal-water reaction at these low temperatures, the core geometry is still coolable and long term cooling can be established. On the basis of our evaluation of these analyses, we have concluded that the emergency core cooling system will provide adequate protection for small breaks in the reactor coolant system.

6.3.5 Core Flooding Tank Line Break Analysis

6.3.5.1 General

In the course of the staff's safety review of several plants with B&W reactors, a potentially serious accident was identified. This postulated accident involves the double-ended break in a short section

* Long term cooling is established in the applicant's opinion when the core is covered with mixture, more water is being supplied than leaked, the pressure is stabilized and the cladding temperature is falling.

of pipe downstream of the last check valve in either of the two lines which connect a core flooding tank (CFT) to the reactor vessel. These lines also connect the low pressure injection (LPI) piping to the reactor vessel. As originally designed, each LPI pump in ANO-1 fed only one CFT nozzle. If, coincident with this postulated accident, loss of all offsite power and a single active failure (such as in one of the buses supplying emergency power) is assumed, the ECCS could be degraded to only one CFT and one high pressure injection (HPI) pump. Although extremely unlikely, this postulated accident would be particularly severe if sufficient water to cool the core were not to remain in the reactor vessel during this accident. On the basis of our review we concluded that the availability of one HPI pump and one CFT would not provide reasonable assurance that a sufficient amount of water would be provided for this purpose.

In order to retain more water in the vessel during this accident, the applicant has installed flow limiting orifices in the nozzles of the CFT lines. This modification has reduced the maximum possible break size from 0.72 ft^2 to 0.44 ft^2 and would, by B&W's calculations, allow several more feet of liquid to remain in the core during this accident. The applicant's original response to the staff's questions on this accident, B&W Topical Report BAW-10034 Supplement 1, is no longer applicable since this report presents an analysis without CFT nozzle flow limiters installed. However, Duke Power Company, in Amendment 39 to its Oconee application dated January 29, 1973 submitted an analysis showing the effects of the flow limiters for this accident. This analysis is also applicable to ANO-1. A summary of the results of the analysis is presented in Section 6.3.5.2.

In evaluating the consequences of this accident, the staff has conducted independent calculations using the RELAP, TOODEE and SWELL computer codes, with assistance from our consultant, Aerojet Nuclear Corporation (ANC). A summary of these independent calculations is included in Section 6.3.5.3 of this report.

6.3.5.2 B&W Analysis

Amendment 39 to the Duke application dated January 29, 1973 provided the results of the B&W analysis of a postulated CFT line break accident for an Oconee reactor applicable also to the ANO-1 reactor. In conducting this analysis, B&W used the small break evaluation model described in Section 6.3.4 of this report. There were several changes to this small break model for the CFT line break analysis due to the unique break location. The most significant changes involved adding two additional nodes in the downcomer annulus and increasing the size of the core node to include most of the upper plenum volume.

One important parameter in this analysis is the amount of water remaining in the vessel during the transient, which determines the height of fluid in the core, and therefore, the heat transfer capability of the core, and the maximum cladding temperature. To determine the quiet water level for the core (no level swell due to heat addition) B&W used three different CRAFT models to determine the sensitivity of the level prediction to noding. The different models provide good agreement, with the lowest quasi-equilibrium liquid level approximately at the six foot elevation. When the liquid swell due to heat addition from the core is considered, the resulting two phase mixture (water and steam) level covers the core for most of the time during the course of the accident. In addition to these CRAFT models which used the Redfield variable bubble rise model, B&W used one with a higher bubble rise velocity which would be more consistent with the two phase mixture height predicted by B&W's FOAM code (described in Section 6.3.5.4). This CRAFT model prevented the two phase mixture from being lost through the vent valves and the break and caused the liquid level to increase from approximately a 6 foot to a 9-10 foot core elevation. B&W's calculations indicate that only the upper part of the core is not covered by mixture during this transient. Sufficient steam is generated by the covered portion, however, to cool this uncovered part. Since the lower portion of the core is covered with a two-phase mixture, pool film boiling will provide sufficient cooling in this region; consequently, the maximum cladding temperature w'' occur in the upper uncovered portion of the core. The upper port when not covered by mixture is cooled by forced convection heat transfer to the steam. To establish the maximum cladding temperature, B&W investigated several axial power peaking shapes. A summary of these results is provided below:

Elevation of power peak from the bottom of core, ft	Elevation of peak cladding temperature from the bottom of core, ft	Peak clad temperature, °F
5.5	5.5	731
7.8	11.4	964
10.6	11.4	1199

These cladding temperatures would produce essentially no metal-water reaction and the core geometry would remain unchanged except possibly for some minor clad swelling in the case of the 10.6 ft power peak.

6.3.5.3 Staff Calculations

Independent analyses of B&W core flood tank line breaks have been performed by the Regulatory staff to aid in the evaluation of this postulated accident. These analyses have considered both the blowdown hydraulics and the heat transfer phenomena resulting from the predicted core water level.

The staff has performed several blowdown analyses using the RELAP computer program. These analyses included both a modeling study and a determination of the sensitivity of the analyses to the bubble rise model. To perform these studies, several system noding models were developed. A summary of these models is presented in Table 6-1. There were three basic models used in the analysis. The first (LARGE MODEL) was a 36 node model previously used to perform large break analyses. This model used excessive computer time for small break analyses, but it was used as the basic comparison model for other small-break models. This model had seven heat transfer nodes in each steam generator and three core nodes. Also, all cold and hot legs were noded separately.

A second model (REDUCED MODEL) was generated to study azimuthal noding in the downcomer region. It consisted of 2 separate reactor coolant loops with the hot legs combined to reduce computer running time. Also the number of heat transfer nodes in the steam generators were reduced from seven to three.

The third model (SMALL MODEL) was developed to perform downcomer axial noding studies. The two hot legs and four cold legs were combined to form a single loop with one steam generator containing two heat transfer nodes. To insure that each model predicted the same blowdown characteristic, the two smaller models were compared to the large 36 node model (standard model used for comparison). The pressure transients calculated by these three models are presented in Table 6-2. This table shows that each model predicts very similar pressure results.

After comparisons had been performed, an axial noding study was made for the downcomer region using the small model. Investigations into the effect of using a bubble rise assumption (compared with a homogeneous assumption) and the number of downcomer nodes was performed. Also, the effect of bubble rise velocity (V_B) on the blowdown characteristics was investigated.

The first effect to be investigated was the assumption of using a bubble rise vs. a homogeneous assumption for the downcomer. Two important differences were noticed when comparing these two models, each having a one-node downcomer, but one having a bubble rise assumption (V_B = 3 ft/sec) and the other using a homogeneous assumption. These differences were in the rate of depressurization and amount of water left in the vessel. Table 6-3 shows a comparison of the downcomer pressure vs. time. The effect of using a bubble rise model is to extend the blowdown time. One other important difference is that the water remaining in the vessel for the homogeneous model during blowdown is reduced. A comparison of the water level in the vessel at 200 seconds showed that the model assuming a homogeneous downcomer predicted 6308 lbs of water would remain in the (corresponding to a level several feet below the core) while the bubble rise model predicted that 83518 lbs ($\sim$ 7 ft core elevation) would remain.

There are considerable differences between the assumptions of the homogeneous and bubble rise models. The bubble rise model inherently assumes that phase separation occurs (separation between the steam and

TABLE 6-1
SUMMARY OF RELAP COMPUTER MODELS

Model Size	No. of Steam Generator Nodes	Number of Core Nodes	Number of Hot Leg Nodes	Number of Cold Leg Nodes	Number of Downcomer Nodes	Description
Large Model, 36 Nodes	7 in each	3	2	4	1	Basic Blowdown Model
Reduced Model, 21 Nodes	2 in each	3	2	2	1	Used to Perform Radial Downcomer Noding Study
Small Model, 15 Nodes	2 (Both Loops combined)	3	1	1	1	Used to Perform Axial Downcomer Noding Study
Small Model	2 "	3	1	1	2	Homogeneous Downcomer
Small Model	2 "	3	1	1	2	Lower Downcomer Node Homogeneous Bubble Rise in Upper V_B = 3 ft/sec
Small Model	2 "	3	1	1	2	Lower Downcomer Node Homogeneous Bubble Rise in Upper V_B = 5 ft/sec
Small Model	2 "	3	1	1	4	All Downcomer Node Homogeneous
Small Model	2 "	3	1	1	1	Downcomer Node Bubble Rise
Small Model	2 "	3	1	1	2	Break Area = 0.44 ft^2
Reduced Model	2 "	3	2	2	4	All Downcomer Node Homogeneous
Reduced Model	2 "	3	2	2	8	All Downcomer Nodes Homogeneous

TABLE 6-2

VESSEL PRESSURE COMPARISON
FOR
THREE STANDARD MODELS

Time, Sec	Calculated Pressure, psig Large Model 36 Nodes	Reduced Model 21 Nodes	Small Model 15 Nodes
0	2250	2250	2250
1	1597	1606	1604
2	1588	1617	1617
5	1583	1637	1637
10	1507	1504	1501
15	1327	1347	1345
20	1177	1153	1153
30	1101	1060	1060
40	1043	993	933
50	966	928	928
60	864	849	849
70	734	745	746
80	587	600	610
90	409	432	429
100	337	262	269

water phases). The homogeneous model assumes that phase separation does not occur. For a large break the homogeneous model may be closer to reality in the early part of the transient. Analyses performed by the staff (as well as B&W) show that the CFT line break leads to a relatively gradual reduction in vessel pressure and low flow rates through the system. This is especially true after the first 20 seconds. From these analyses the staff believes that phase separation occurs and a bubble rise model is appropriate for this accident. This model leads to a prediction of a larger mass of water present during all stages of the transient when compared with the predictions obtained with the homogeneous model. However, the homogeneous model used throughout the transient is not realistic and leads to a nonrealistic low quiet water level calculation. This low level would lead to unacceptable cladding temperatures.

Further support for the use of a bubble rise model was given in an Idaho Nuclear Corporation report (Report IN 1444, December 1970). In this report the RELAP code was used to predict results obtained from a semi-scale blowdown experiment. Figure 10, page 17 of the IN 1444 report shows that the residual water remaining in a vessel after the end-of-blowdown is best predicted by using a bubble rise model. The figure indicates that the density gradient should be between 0.8 and 1.0 with a bubble rise velocity of 3 ft/sec. Based on these results and calculations performed by the staff, we believe that a bubble rise model better predicts the actual system response.

TABLE 6-3

COMPARISON OF VESSEL PRESSURE
FOR
BUBBLE RISE AND HOMOGENEOUS ASSUMPTION

Time, Sec	Pressure for bubble rise model assumption, psig	Pressure for homogeneous model assumption, psig
0	2250	2250
10	1500	1500
20	1150	1150
30	1050	1060
40	970	990
50	890	930
60	800	850
70	720	750
80	620	610
90	510	430
100	410	270
110	340	160
120	280	80

One other conclusion drawn by B&W was that CFT line break could not lead to an end-of-blowdown (as defined in the B&W evaluation model for a large break). In the downcomer noding studies performed by the staff it was concluded that end-of-blowdown could be calculated to be made to occur by selecting 4 axial nodes in the downcomer and using a homogeneous assumption in all nodes. The end-of-blowdown would occur at about 120 seconds. With these assumptions the end-of-blowdown would occur because the cold core flood tank water would enter a node containing steam, which is then condensed, thus reducing the pressure below containment pressure; this node also contained the broken CFT line such that the reduction in pressure caused the break flow to go to zero (the definition of end-of-blowdown). This effect was investigated using the REDUCED MODEL with 2 axial nodes in the downcomer. An end-of-blowdown was not predicted using this model. The staff concluded that the REDUCED MODEL was a better representation of the physical system and that end-of-blowdown probably would not occur.

The model chosen as the analysis tool to analyze the 0.44 ft^2 CFT line break for ANO-1 was the SMALL MODEL using 2 downcomer nodes and a bubble rise assumption. Vessel pressure and quiet water levels predicted by this model were compared with the B&W analysis. Pressure comparisons between the RELAP model and B&W small break model are presented in Table 6-4. Quiet water level comparisons were also made and showed good agreement between the two models. The staff considers

TABLE 6-4

COMPARISON OF VESSEL PRESSURE
FOR
APPLICANT AND STAFF MODEL

Time, Seconds	Vessel Pressure, psig: Applicant's Model	Vessel Pressure, psig: Staff Model
0	2216	2216
50	1050	1020
100	800	800
150	575	530
200	450	412
300	320	255
400	250	210
500	180	170

the "quiet water level" calculated by the B&W model to be a best estimate of residual water left in the vessel.

One assumption used by B&W was that the accumulator (i.e., the CFT) bypass criterion should not apply to the CFT line break. B&W gave two reasons in support of this change. The first was that the system pressure for the CFT line break never reached the end-of-blowdown condition; the second reason was that the fluid velocity in the downcomer was always downward, except for short time periods when the calculated velocities were low (maximum negative velocity was approximately 4 ft/sec). These low velocities should not cause the ECC water to be entrained out the break. In the staff independent evaluation the same velocity effect was seen. For a single node downcomer using a homogeneous assumption, a maximum velocity of about 4 ft/sec for approximately 25 seconds was obtained. Analysis reported by B&W using a three node downcomer and using a bubble rise model (also calculated by RELAP in the independent analysis) showed that the maximum negative velocity was approximately 4 ft/sec for about 100 seconds. Critical velocity for entrainment from an annular film is about 13 ft/sec at 300 psia using the Wallis correlation* given below.

$$j_g = 2.46 \times 10^{-4} \frac{\sigma}{\mu_g} \sqrt{\rho_f/\rho_g}$$

*G. B. Wallis, <u>One-Dimensional Two-Phase Flow</u>, McGraw-Hill Book Company, 1969.

where:

j_g = vapor volumetric flow rate per unit area of pipe (Critical velocity for entrainment)

σ = surface tension

μ_g = vapor viscosity

ρ_g, ρ_f = density of the vapor and liquid

Based on these calculations, the staff has concluded that the accumulator bypass assumption need not be applied to the ANO-1 CFT line break with a break area of 0.44 ft^2.

The boil off rate of about 200 seconds is approximately 5110 lb/min which is approximately matched by the one HPI pump supplying 4078 lb/min for most of the transient. Since the effective supply rate does not meet the AEC's "abundant emergency core cooling" criterion, the staff believes that the applicant should have a method of supplying additional water for this postulated accident. This additional water should be supplied at a rate which would insure that the core could be reflooded at a reasonable rate.

To supply this additional water, the applicant has modified the low pressure injection (LPI) system piping inside containment so that the discharge line from each pump branches to connect to both CFT lines. Each of these branches contains a check valve and a flow limiter. Thus, no matter which CFT line is postulated to break or which LPI supply is assumed to fail approximately one-half the flow rate from the one LPI pump will be injected into the reactor vessel. This amount of additional water would assure the availability of an abundant supply of cooling water to reflood the core and remove stored and decay heat. The staff will review test data for these cross-connect flow limiters to confirm their ability to conserve an acceptable fraction of the LPI flow.

6.3.5.4 Heat Transfer Analysis

B&W's fuel cladding heatup analysis for this accident is basically identical to that described in Section 6.3.4 of this report for small breaks analysis. Since the reactor coolant system never reaches an end-of-blowdown condition, and water remains in the vessel, the reflooding analysis normally done is replaced by a heatup analysis using the THETA and QUENCH codes with input from the blowdown code, CRAFT and the level swell code, FOAM.

There are two major differences between the methods used for the CFT line accident and those used in the small break model. First, the level swell for the CFT line analyses was based on a Wilson bubble rise calculation in the FOAM code, while the small break model used the mixture level calculated in CRAFT. Second, the small break model assumed steam generation due to a mixture level at a core elevation of 8 feet, the minimum level for any transient, while for the CFT line break, the mixture level calculated using FOAM was used for the steam generation calculation. However, the calculation still conservatively assumed the average assembly steam generation rate for the maximum heat generation rate assembly.

B&W has compared results obtained from use of its FOAM code to three sets of experimental data obtained from tests performed by Westinghouse, General Electric (GE) and a Japanese group. The Westinghouse test was contracted for by the Duke Power Company for this explicit purpose. Of the three tests it utilized the largest number of simulated fuel rods (490) and the highest pressure (400 psia). The other tests, by GE and the Japanese, were based on a 49 rod Boiling Water Reactor (BWR) assembly at 100 psia and atmospheric pressure. However, neither the number of rods (49 or 490) nor the configuration (PWR vs BWR geometry) significantly affected the applicability of the data for verification of the FOAM code; in fact, the variations in these two parameters helped to define the insensitivity of the heat transfer/hydraulics phenomena and FOAM's prediction of these phenomena to these parameters. The comparisons of FOAM results to the data were generally within the experimental uncertainty of the data except for several Westinghouse data points at 100 psia. For these data, the FOAM code overpredicted the measure swollen level by about 10%. This may be attributed to nonquantified uncertainty in some of the measured parameters, such as the amount of subcooling in the inlet water.

On the whole, the staff concluded that the FOAM code predicted the swollen levels measured in the three tests reasonably well. These tests were within the range of power levels, pressures and geometric configurations which would exist during the CFT line break accident. The staff concludes that the use of the FOAM code is appropriate in calculating two-phase mixture heights for this accident.

The results obtained by application of B&W's FOAM code to analysis of the CFT line break accident were presented in Section 6.3.5.2. It was predicted that the core would be covered with a two-phase mixture during the accident except for the period between 500 and 700 seconds after the accident. The peak cladding temperature occurred at approximately 700 seconds and reached 1199°F.

In examining the swollen levels predicted by FOAM for this accident, it is necessary to point out a conservatism which may have an exaggerated effect if compared to a more realistic calculation. The lowest liquid levels predicted by CRAFT were used as input to the FOAM code. This is actually a contradiction to fact, since the lowest liquid level CRAFT predicts is the high swollen level (above the top of the core). This swollen level (above the top of the core) would not allow any significant cladding heat up. On the other hand, for the lower swollen level consistant with the FOAM prediction, CRAFT would predict more liquid left in the vessel and this would result in about four more feet of liquid level in the core (9 feet versus 5 feet). This calculation would predict the core to be covered with two-phase mixture and there would also be no significant cladding heat up. Therefore, for a consistant set of predictions (high swollen level and low liquid level or low swollen level and high liquid level) there would be no significant cladding heat up. The analysis which is reported is the worst combination of both situations and results in an increase in cladding temperature.

To independently determine the two-phase mixture height in the core, the staff and its consultant, Aerojet Nuclear Corporation have developed a code (SWELL) using the Wilson bubble-rise model and a calculational procedure developed by GE and described in the Quad-Cities application (Docket 50-254 and 265). The SWELL code uses essentially the same calculational scheme as B&W's FOAM code. Preliminary calculations with this code have shown agreement with B&W's FOAM code for the Westinghouse tests. Since the SWELL code is not presently well indexed against experimental tests, the staff also examined the cladding heat up transient in the 500 to 700 second period where B&W predicts the core may be uncovered. Using the TOODEE* heat transfer code, the sensitivity of the peak cladding temperature to swollen level was examined. The swollen level was reduced by an arbitrary 25%, this resulted in an increase in peak cladding temperature, to 1552°F, at 700 seconds. Although the temperature did increase 300°F over the applicant's calculation, the resultant peak cladding temperatures would be acceptable even for an arbitrary 25% reduction in swollen level.

*J. A. McClure, TOODEE - A Two-dimensional, Time-dependent Heat Conduction Program, IDO-17227, April 1967.

6.3.5.5 CFT Line Break Conclusions

Based on the staff's independent calculations, B&W's analysis, and the design changes incorporated by the applicant the staff has concluded that the ANO-1 emergency core cooling system, as modified, will provide adequate protection for a break of a CFT line.

6.3.6 Conclusions on Adequacy of ECCS

On the basis of our evaluation of the additional B&W analyses, described above, we conclude that our acceptance criteria, as described in the Commission's Interim Policy Statement have been met:

1. The maximum calculated fuel element cladding temperature does not exceed 2300°F.
2. The amount of fuel element cladding that reacts chemically with water or steam does not exceed 1% of the total amount of cladding in the reactor.
3. The calculated 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 thermal activity of long lived fission products remaining in the core.

6.4 High-Energy Line Rupture Outside Containment

The Staff's continuing review of reactor power plant safety indicates that the consequences of postulated pipe failures outside of the reactor building, ircluding the rupture of a main steam or feedwater line, need to be adequately analyzed by licensees and applicants and evaluated by the staff.

On April 6, 1973, in Amendment 36, AP&L provided the results of an initial review which was performed in response to our letter of November 15, 1972 concerning the failure of high-energy lines outside of reactor containment. The basis and evaluation criteria regarding the location and type of failures that were to be considered were developed by the Regulatory staff and forwarded to the applicant in that letter (a copy of these criteria is attached as Appendix C).

The 36-inch diameter main steam lines in ANO-1 come out of the reactor building in the auxiliary building above the fuel handling area, pass outside the building side-by-side and enter the turbine building through a concrete tunnel just below grade level. The 18-inch diameter main feedwater lines both come from the turbine building through the South penetration room to enter the reactor building. AP&L's initial review examined these and many other smaller diameter high-energy lines against the criteria provided. AP&L identified four areas where design modifications are needed:

(1) In the auxiliary building where pipe whip by a failed 36-inch steam line could possible damage both steam supply lines to the emergency feedwater pump turbine. This pump, or its electric-motor driven backup, is required for plant cooldown.

The applicant has proposed a strong pipe restraint as a means to limit pipe whip damage to only one of the pump turbine supply lines.

(2) In the steam line tunnel where the jet force of a steam line failure might damage adjacent safety-related piping or the ambient pressure buildup might cause structural failure of the tunnel. The applicant has proposed a local barrier and pressure vent openings to resolve these concerns.

(3) In the turbine building where jet forces or pipe whip associated with failure of a main feedwater line might break into and cause failure in a safety-related electrical switchgear room. The applicant has proposed reinforcement of the door and walls of this room to resolve this concern.

(4) In the South penetration room where a rupture of main feedwater line can cause overpressurization of the compartment, AP&L has proposed a flow choke pipe encapsulation and controlled venting to cope with this event.

Based on our initial review of these high energy pipe failures outside of the reactor building, we conclude that additional protective measures will be required at ANO-1 to provide for safe shutdown following a postulated main steam or feedwater failure. In its letter of April 23, 1973, AP&L has committed to perform the necessary structural and equipment modifications before exceeding 1% power. Based on our review of modifications proposed recently for other reactor plants with similar problems, such as increasing the vent areas of interior subcompartments or encapsulating the high energy lines or a combination of both, we judge that practical means are available for implementing these protective measures.

Subject to acceptance of the final design of required plant modifications, we conclude that such modifications are feasible and when implemented will assure a safe plant cooldown following the postulated rupture of any pipe carrying a high energy fluid outside the reactor containment. We will perform an acceptance review of the final design and the piping analysis of these modifications prior to plant startup.

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The Commission's General Design Criteria (GDC), IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE-279), IEEE Criteria for Nuclear Power Plant Class IE Electric Systems (IEEE-308), and applicable Regulatory guides for water cooled nuclear power plants have been utilized as the bases for evaluating the adequacy of the protection and control systems.

Also, the results of our logic and electrical schematics review and site visit are reflected in this evaluation.

7.2 Reactor Protection System (RPS)

The RPS is functionally identical to that which was reviewed and accepted for the Oconee-1 plant and is supplied by B&W. In essence, the system consists of four redundant and independent protection channels, each terminating in a trip relay within a reactor trip module. The trip relays de-energize upon detection of any of the abnormal operating conditions listed in Table 7-1. Each reactor trip module combines the four channel trip outputs in a 2/4 logic to operate the control rod power supply breakers. Thus, the coincidence logics in all reactor trip modules trip whenever any two of the protection channels trip, commanding all control rod breakers to open.

Table 7-1
REACTOR TRIP SUMMARY

Trip Variable	No. of Sensors	Steady-State Normal Range	Trip Value or Condition for Trip
Over Power	4 Flux Sensors	2-100%	$\geq$ 107.5% of rated power.
Power-Imbalance-Flow	4 Flux Sensors 8 ΔP Flow Transmitters 2 Flow Nozzles	Variable	1.10 times flow minus reduction due to imbalance function.
Power/Reactor Coolant Pumps	4 Pump Monitors With 16 Contacts 4 Flux Sensors	2-4 Pumps	Loss of one or two operating reactor coolant pumps during two-pump operation. Loss of one operating coolant pump in each loop, and reactor neutron power exceeds 55% rated power. Loss of two operating reactor coolant pumps in same loop.
Reactor Outlet Temperature	4 Temperature Sensors	532-604°F	$\geq$ 619°F.
Pressure/Temperature	4 Pressure Sensors 4 Temperature Sensors	Variable	(13.26T-5989) $\geq$ P.[a]
Reactor Coolant Pressure	4 Pressure Sensors	2,090-2,220 psig	$\geq$2,355 psig (high). $\leq$1,800 psig (low).
Reactor Building Pressure	4 Pressure Sensors	0 psig	$\geq$4 psig.

[a] T is in °F and P is in psig.

We have reviewed all aspects of the RPS, including logic schematics, testing capabilities and control of bypasses, and concluded that this system is acceptable.

7.3 Engineered Safety Features (ESF) Actuation System

The ESF actuation system is functionally identical to that which was reviewed and accepted for the Oconee-1 plant and is supplied by B&W. In essence, the system is comprised of two redundant and independent digital subsystems, each capable of initiating the minimum required ESF through five actuation channels. All five redundant pairs of actuation channels in each digital subsystem receive input intelligence from three redundant and independent analog subsystems. Each analog subsystem includes five distinct trip logic channels, each supplying input information through a trip relay to the corresponding pair of redundant ESF actuation channels. Each of the five actuation channel combines the corresponding logic channel trip signals from the three redundant analog subsystems in a 2/3 logic to initiate a protective action. The trip relays of the analog subsystem logic channels de-energize upon detection of the ESF actuation conditions listed in Table 7-2. Conversely, the digital subsystem actuation trip relays require power to initiate a protective action.

Table 7-2
ENGINEERED SAFETY FEATURES ACTUATION CONDITIONS

Actuation Channel No.	ESF Action	Analog Subsystem Trip Signal	Steady State Normal Value	Analog Channel Trip Points
1, 2	High-Pressure Injection	Low Reactor Coolant Pressure or	2,090-2,220 psig	1,500 psig
		High Reactor Building Pressure	Atmospheric	4 psig
3, 4	Low-Pressure Injection	Low Reactor Coolant Pressure or	2,090-2,220 psig	1,500 psig
		High Reactor Building Pressure	Atmospheric	4 psig
5, 6	Reactor Building Cooling and Reactor Building Isolation	High Reactor Building Pressure	Atmospheric	4 psig
7, 8	Reactor Building Spray System	High Reactor Building Pressure	Atmospheric	30 psig
9, 10	Reactor Building Spray Chemical Addition	High Reactor Building Pressure	Atmospheric	30 psig

We have reviewed all aspects of the ESF actuation system, including logic schematics, testing capabilities and control of bypasses, and concluded that this system is acceptable.

7.4 ESF Actuator Circuits and Related Equipment

We have reviewed the actuator control circuits and related equipment pertaining to the ESF systems, and concluded that the designs conform to our criteria and are acceptable with the resolution of the following items:

7.4.1 Air-Operated Valves

Although ESF air-operated valves do not require air pressure to open or close upon an ESF trip signal, it appeared from reviewing the electrical schematics and functional piping and instrument diagrams (P&IDs) that there are some valves which require air to operate. The applicant verified this to be true for three valves and furnished seismic Category I accumulators and check valves with protective barriers to provide these three valves with an assured air supply. We consider this acceptable.

7.4.2 RHR Overpressure Protection Interlocks

The motor-operated suction valve interlocks used to prevent over-pressurization of the Residual Heat Removal (RHR) System by the

Reactor Coolant System did not initially conform to the criteria the Regulatory staff considers appropriate for high pressure to low pressure interfaces. The following criteria were identified to the applicant:

(1) At least two valves in series shall be provided to isolate the low pressure system.

(2) For systems where both valves are motor-operated, the valves shall have independent and diverse interlocks to prevent valve opening at high pressure. These interlocks shall be designed to comply with all the requirements of IEEE-279.

(3) Automatic closure of the motor-operated valves whenever the primary system pressure exceeds the pressure rating of the low pressure system. The closure devices shall be designed to comply with all the requirements of IEEE-279.

The applicant has submitted a revised design which conforms with the stated criteria in Amendment 37.

7.4.3 Core Flooding Tank Isolation Valves

The applicant has elected to open the breakers supplying power to the CFT motor-operated isolation valves in order to ensure against accident closure of these valves during normal reactor operation. Based on this mode of operation, the applicant was advised that the proposed administrative controls did not provide sufficient assurance that these valves will be open when required. We have informed AP&L that we will require that the valve control circuits be designed to meet IEEE-279 and the following features be incorporated in the design:

(1) Valve position visual indication (open or closed) in the control room for each valve which is not dependent on power being available to the valve actuator.

(2) Valve not open audible alarm in the control room for each valve, actuated when the valve is not in the fully open position and reactor coolant pressure is above a preset value.

(3) Valve position indications both visual and audible to be derived from redundant and independent valve position sensors and circuitry, such as limit switches actuated by the valve motor operator and valve position limit switches activated by stem travel. The reactor coolant pressure signals shall also be redundant and independent.

(4) A Technical Specification requirement that the reactor shall not be made critical or shall be shut down unless each CFT isolation valve is open and the breaker supplying the power to valve operator is locked open and tagged. The applicant agreed to these requirements.

7.4.4 Auxiliary Systems Supporting ESF Systems

Pump-Motor Bearing Cooling Failures

It is not evident that the auxiliary systems providing lubricating oil and cooling water to ESF systems motor and pump bearings are essential to the proper functioning of the ESF systems. The applicant has agreed to review all auxiliary systems supplying lubricating oil and cooling water to ESF systems motor and pump bearings. In any case where the auxiliary system is of lower quality than the main ESF system the auxiliary system will be changed to meet the objectives of IEEE-279. We find this acceptable.

Switchgear Rooms Cooler Failure

The two pairs of redundant and independent ESF switchgear room coolers were found to be powered from the same bus. A failure of this bus would have caused the loss of cooling capability in both switchgear rooms. The applicant modified the design to supply power to each pair of room coolers from independent buses thus providing the necessary independence and redundancy.

7.5 Separation and Identification Criteria for Protection and Emergency Power Systems

We have reviewed the applicant's criteria for separation and identification of cables and examined the design arrangement of these as well as other safety-related systems. We have found that these criteria and design arrangements are acceptable, except for the items listed below and those included in Section 7.9 of this report. We will require final resolution of these outstanding items prior to licensing this plant for operation.

(1) Two of the three redundant coolant pressure sensors associated with the ESF actuation system are mounted on a common instrument rack. The applicant has agreed to provide additional protection in the form of physical barriers to protect these two sensors from common mode failure.

(2) The doors separating adjacent redundant ESF equipment rooms are not of the watertight construction in the diesel-generator and 4160 V switchgear rooms. It was our concern that the break of a service water supply line in either room could cause the flooding of both redundant rooms. The applicant has reviewed these rooms and agrees to modify the design of the door between the diesel-generator rooms and to provide an externally drained guard pipe around the service water lines in the 4160 V switchgear rooms.

(3) The exhaust duct emanating from one of the 125 volt d-c station battery rooms passes through the other redundant battery room. It was our concern that a fire and/or explosion in one room could thus be propagated to the other room resulting in the loss

of both redundant 125 volt d-c systems. The applicant has agreed to relocate this duct to assure the independence of these rooms.

7.6 Environmental, Radiation and Seismic Qualifications

The applicant has stated that all safety-related motors, cables, instruments and other equipment located inside the containment which must operate during and subsequent to an accident, will be capable of functioning under the post-accident temperature, pressure, humidity and radiation conditions for the time periods required. This capability has been demonstrated by testing, as documented in the FSAR, and is acceptable.

The applicant has documented that the seismic testing program meets the requirements of IEEE Standard 344-1971, Seismic Qualifications of Class I Electric Equipment for Nuclear Power Generating Stations. It has also been documented in the FSAR that the protection system instrumentation has been seismically qualified, and we have concluded that it is acceptable.

7.7 Control Systems

The applicant has stated that there are no significant differences between the control systems of this plant and those of the previously approved Oconee-1 Plant except for the controls of the emergency feedwater system. The Oconee design provides for control of emergency feedwater through the normal startup feedwater valves, whereas the ANO-1 design includes regulation of emergency feedwater through separate lines and valves from those used for normal startup. With the exception of the emergency feedwater system controls, we found that minor differences in the other systems have not changed the functional design or degraded the safety of this plant and concluded that these control systems are acceptable. However, the final acceptability of the overall control system scheme is predicated on the resolution of the safety significance of the emergency feedwater system and its controls; this is discussed in Section 7.8 of this evaluation.

7.8 Emergency Feedwater (EF) System

This system is involved in the still current evaluation of high energy line rupture outside containment (see Section 6.4). In addition, we have not completely resolved all our concerns about single failure aspects of the power supply and controls for the electric-motor-driven feed pump. We will complete our evaluation of this system to assure its acceptability prior to licensing this plant.

7.9 Control Room and Rod Drive Control (RDC) Equipment Room

Our review of the control room and RDC equipment room design arrangements revealed the following items of concern:

(1) The RPS equipment cabinets are located in the control room and mounted on a raised floor. Cables entering the RPS cabinets are routed under the raised floor. It appears that the design

arrangement of redundant RPS cables underneath the raised floor lacks the physical independence provided in other areas through which these cables are routed. This cable design arrangement is considered to be vulnerable to common mode failures resulting from design basis events such as fire and flooding. This apparent lack of cable separation and vulnerability to common mode failures is inconsistent with the applicant's own criteria as documented in the FSAR which include compliance with IEEE-279 and IEEE-308. We will require that the applicant either demonstrate the adequacy of this design against all design basis events or modify it to provide the required physical independence of the redundant protection systems prior to issuance of an operating license.

(2) The RDC equipment cabinets, located in a room above the control room, are also mounted on a raised floor. The cable design arrangement underneath the raised floor is of concern for the same reasons stated before for the RPS cables. We will require that the applicant either demonstrate the adequacy of this design against all design basis events or modify it to provide the required physical independence between safety-related cables prior to issuance of an operating license.

(3) Open raceways containing RDC power cables each carrying 47 A are located overhead in the control room. These power cables

are a potential source of fire that could result in not only the loss of the Unit 1 control room, but also the future adjoining Unit 2 control room. The applicant claimed that the cables are derated and only half of these cables will be carrying 47 A at any one time. We have concluded that this cable design does not minimize the probability and effect of fires in the control room as required by GDC 3. We will require that the applicant install a fire barrier separating these open raceways from the control room proper, and provide a separate fire extinguishing system for this space prior to licensing.

7.10 Anticipated Transients Without Scram

In connection with our review of potential common mode failures we have considered the need for means of preventing common mode failures from negating scram action and the possible need for design features to make tolerable the consequences of failure to scram during anticipated transients. This concern is applicable to all light water cooled power reactors.

This problem is being studied on a generic basis and requires further review by the Regulatory staff. If the probability of any of the events considered is determined to be sufficiently high to warrant consideration as a design basis for plants, such as ANO-1, suitable design modifications to reduce the probabilities or to limit the consequences to acceptable levels may be necessary.

8.0 ELECTRICAL POWER

8.1 General

The Commission's GDC 17 and 18, IEEE-308 and Regulatory Guides 1.6 and 1.9 served as the bases for evaluating the adequacy of the electrical power systems.

8.2 Offsite Power

This plant will be interconnected to the electrical grid system through two 500 kV and two 161 kV transmission lines emanating from their respective switchyards. The two types of high voltage transmission lines are located on separate and independent rights-of-way. Both switchyards are arranged in a ring bus configuration and interconnected through an autotransformer bank consisting of three 500/161/22 kV single phase autotransformers. Power from the unit generator will be supplied to the 500 kV switchyard through a bank of three, single phase step-up transformers, and also to the unit auxiliary transformer. Offsite power to the plant is derived from the 161 kV switchyard and the 22 kV tertiary of the autotransformer bank. These power sources are separated by high voltage circuit breakers and are connected to two separate startup transformers. The startup transformer being fed from the 161 kV switchyard will be shared between Unit 1 and the future Unit 2. All of the high voltage circuit breakers in both switchyards are provided with primary and backup relaying circuits powered from independent supplies.

The unit auxiliary transformer and each one of the startup transformers is provided with two redundant feeder breakers, each connected to a separate main 4160 V bus. Each redundant emergency bus is connected to a main 4160 V bus through a single feeder breaker. The emergency buses will be powered from the unit auxiliary transformer during normal operation; upon loss of the normal supply, power will be made available automatically to these buses from either one of the startup transformers. Each of the startup transformers and attendant distribution systems have sufficient capacity to meet startup, shutdown and emergency load requirements. Further, the applicant has stated that the stability of the 500 kV and 161 kV transmission systems will be maintained on tripping of the unit generator.

Our review of the offsite power system revealed that the design of the interlock schemes used to coordinate the connection of available power supplies to the emergency buses was susceptible to single failures. This item is discussed in Section 8.3 of this evaluation. We have concluded that the offsite power system design, with the satisfactory resolution of this item, is acceptable.

8.3 A-C Onsite Power System

The a-c emergency onsite power system is comprised of two redundant and independent distribution systems, each powered by one of the two redundant diesel generators. Each distribution system includes

4160, 480, and 120 volt load centers, and each load center bus in a distribution system can be manually connected to its redundant counterpart in the other distribution system through two serially connected bus tie breakers. The safety loads for the unit are distributed evenly between the two distribution systems with the exception of the third high pressure injection pump and the third service water pump. These pumps can be powered from either distribution system through separate breakers. The selection of the power feed will be accomplished manually through interlock bus-transfer switches which prevent interconnection of the power supplies. In addition, there is a single 480 V motor control center which can be manually connected to either one of the distribution systems through a mechanically interlocked transfer switch. We have determined that the loads connected to this bus have no safety significance and the interlocks provided to prevent the propagation of faults to the redundant emergency buses are considered adequate. We conclude that the design of the manual transfer of this load cent... is acceptable.

Each diesel generator is rated at 4160 V, 2,600 kW continuous, 2,850 kW for 2,000 hours and 3050 kW for 30 minutes. The loading of the diesel generators is within the limits suggested by Regulatory Guide 1.9, and each of the two diesel generators can provide the emergency power needed for minimum required safety loads. Each diesel generator will be automatically started by an undervoltage signal from its respective 4160 V ESF bus, or by either of the two redundant ESF actuation trip signals, or when the main generator trips. If offsite power is not available, the 4160 V ESF buses will be automatically isolated from all supply sources and all outgoing feeder breakers will be tripped. The diesel generators will then be connected to their respective 4160 V ESF buses and, under accident conditions, the ESF actuation trip signals will cause the 4160 V emergency loads to be automatically connected in a predetermined sequence to both diesel generators.

Our review of the electrical schematics revealed that the tie breakers connecting redundant ESF buses at the 4160 and 480 volt levels would not automatically open upon receipt of an ESF actuation trip signal. It was discovered that a single failure of a relay could have prevented both 4160 V bus tie breakers from opening when required. The relay had contacts included in both of these breakers trip circuits. Failure of the tie breakers to open would, in turn, prevent the closure of either of the two feeder breakers connecting onsite power to the ESF buses. In addition, the incoming and outgoing feeder breakers for the two redundant 4160/48 V transformers would not automatically close upon receipt of an ESF actuation trip signal. The applicant has agreed to modify the electrical system design to eliminate this problem. We will review the revised design to determine that the problem is resolved prior to issuance of an operating license.

The diesel generators are located in separate seismic Class I rooms. Each diesel generator has independent auxiliary systems and a separate seismic Class I underground fuel storage tank. The total fuel oil storage capacity in these underground tanks provides for at least 7 days of diesel generator operation at full rated load. The fuel supply for the emergency diesels meets our criteria and is acceptable.

The applicant stated its intention to use the standby power supply diesel generator sets to supply power to the electrical system during peak load demand periods. We questioned and discussed this subject with the applicant and indicated that frequent and prolonged paralleling of the preferred (offsite) and standby power supplies is contrary to providing the independence required by GDC 17 and IEEE-308. GDC 17 requires that provisions be included to minimize the probability of losing electrical power from any of the remaining supplies as a result of, or coincident with, the loss of the main unit generator, the loss of power from the grid (offsite preferred power supplies), or loss of power from the onsite (standby) power supplies. In addition, although IEEE-308 does not prohibit the use of diesel generators for other purposes, it does require that the preferred and standby power supplies not have a common failure mode. Common failure is defined as: "A mechanism by which a single design basis event can cause redundant equipment to be inoperable." On the basis of our review of the intended use of diesel generators for system peaking, involving frequent interconnections of the preferred and standby power supplies, we concluded that the design was not sufficiently immune to potential common mode failures. Therefore, based on our interpretation of GDC 17 and IEEE-308, Section 5, Item 5.2.1(5), we will require that the diesel generator sets be used only for purposes of providing emergency standby power.

Our review of the electrical schematics also revealed the potential for indiscriminate tripping of available offsite power supplies. Also, potential single failures were identified which could result in the loss of both offsite and onsite power to the ESF buses. In our view these problems are a direct result of the complexity of the control circuit design provided to accommodate the proposed system peaking operation with diesel generators. In view of the above deficiencies and our position confining the use of diesel generators, we will require that the applicant perform an overall audit of the present emergency power system design, and modify it as necessary to provide the independence of the power supplies required by GDC 17 and IEEE-308 prior to licensing of this plant for operation.

We conclude that the a-c emergency onsite power system will be acceptable subject to the elimination of design provisions permitting use of diesel generators for system peaking and the correction of associated design deficiencies.

8.4 D-C Onsite Power System

Onsite d-c emergency power is derived from the station and switchyard battery systems. The station battery system is comprised of two redundant and independent 125 volt battery bank-charger units and the attendant distribution systems. Each distribution system will be normally supplied by a battery charger and backed up by a floating battery bank. The battery chargers will be supplied from separate 480 V ESF buses. In addition, there is an installed spare charger which can be manually connected to either d-c distribution system. Each station battery bank is located in a separate seismic Category I room and is sized to carry all its connected safety loads for 2 hours upon the loss of the normal supply.

The d-c safety loads for the unit are distributed evenly between the two distribution systems, except for two of the three ESF actuation analog subsystems which will be powered from one of the distribution systems. Four redundant 120 volt vital a-c distribution panels are provided to supply power to the RPS and the ESF actuation analog subsystems. Each panel will be supplied separately from a static inverter. Each pair of inverters will normally be supplied from separate 480 V ESF buses and backed up from the respective main d-c load center.

The switchyard battery system consists of a single 125 volt battery bank-charger unit and the attendant distribution system.

This power source, in conjunction with a separate d-c supply emanating from one of the two station battery distribution centers, will be used to provide control power for all switchyard breakers.

We conclude that the d-c emergency onsite power system satisfies GDC 17 and 18, IEEE-308, and Safety Guides 6 and 9, and is acceptable.

9.0 AUXILIARY SYSTEMS

9.1 General

The evaluation of safety related auxiliary systems, as set forth in the following subsections, is based on radiological safety requirements. These systems are grouped in the following paragraphs according to their seismic design categories.

The safety-related Category I auxiliary systems consist of the: (1) decay heat removal system; (2) emergency pond and service water system; (3) chemical addition and makeup system; (4) diesel auxiliary systems; (5) fuel storage and handling facilities; and (6) emergency ventilation and air-conditioning systems.

The Category II auxiliary systems consist of the: (1) intermediate and auxiliary cooling water system; (2) process sampling system; (3) fire protection water system; (4) fuel pool cooling and cleanup system; (5) normal heating and ventilation system; (6) compressed air system and (7) communication system.

9.2 Fuel Storage and Handling

9.2.1 New Fuel Storage

New fuel will be stored in a dry vault located in a separate and protected area of the fuel storage and handling portion of the auxiliary building. The vault will accommodate 72 fuel assemblies in storage racks designed with sufficient spacing between the fuel assemblies to assure that the array, when fully loaded with new fuel, will limit the effective multiplication factor of the array (k_{eff}) to less than 0.90 even if flooded with unborated water. We conclude that the design of the new fuel storage facility is acceptable.

9.2.2 Spent Fuel Storage

Irradiated fuel removed from the reactor will be stored in the spent pool. This pool has fuel storage racks to store, shield, and cool spent fuel assemblies prior to shipment. The pool can accommodate 253 assemblies, more than the equivalent of one and one-third full cores. The spent fuel storage racks have been designed with sufficient spacing between assemblies to assure that the effective multiplication factor (k_{eff}) of the array of any fuel stored in this pool will be less than 0.90 even under abnormal (unborated water) storage conditions. Technical Specifications will require use of borated water in the spent fuel pool.

The spent fuel storage pool has been lined with stainless steel to prevent pool leakage through seams and penetrations. No inlets, outlets, or drains have been provided that might allow the pool to be drained lower than 21 feet above the top of the active fuel. External lines extending below this level have been equipped with syphon breakers, check valves, and other suitable devices to prevent inadvertent pool drainage. The pool has been provided with interconnected channel drainage paths behind the liner weld joints to prevent uncontrolled loss of contaminated pool water. A separate spent fuel shipping cask storage area is provided adjacent to the spent fuel pool. An interconnecting canal between the cask storage area and the pool will permit underwater fuel transfer to the shipping cask. The two pools are separated by a watertight barrier, a lined concrete structure to an elevation higher than the stored fuel and a watertight gate, located above the top of the fuel assemblies. The cask storage area, constructed of reinforced-concrete and lined with stainless steel, has been designed to minimize the loss of water due to accidental drop of a storage cask; however, if an accident should breach this area, drainage would not have an adverse effect on the spent fuel pool storage area because of the watertight barrier between the two areas.

The spent fuel storage racks, spent fuel pool and the spent fuel shipping cask storage area have also been designed as seismic Category I structures and the latter two structures afford protection against loss of integrity from postulated tornado missiles as described in Section 3.5. The movement of the crane and a shipping cask over the spent fuel pool is prevented by the use of control interlocks and mechanical stops. We have concluded that the design of the spent fuel storage facility meets the positions set forth in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis", and is acceptable.

9.2.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The system consists of two spent fuel pool circulating pumps, heat exchangers, and filters, a demineralizer, and associated piping, valves, and instrumentation. During refueling operations these pumps will be used to fill the fuel transfer canal inside containment with borated water. In addition, a borated water recirculation pump has been provided to supply water from the BWST (a seismic Category I makeup source) to the spent fuel pool, demineralizer, or filter; this pump may also be used to empty the fuel transfer canal after refueling.

The heat load from the 1/3 of a core stored in the fuel pool following a normal refueling operation will be removed by two pumps and two coolers so as to maintain the pool temperature at 120°F or less. One pump and one cooler, however, can maintain the pool temperature at 135°F. The heat load of an abnormal storage condition

(1-1/3 of a core) could be removed by two pumps and two coolers while maintaining the pool temperature at 150°F. For this case failure of one pump and cooler would result in elevating the pool temperature to 200°F. In the unlikely event that all cooling were lost, the time required to raise the temperature of the pool to 205°F for the specified quantities of stored fuel stated above would be 19 hours (from 120°F with 59 fuel assemblies) and 5 hours (from 150°F with 236 fuel assemblies), respectively. During this period with no outside cooling, ample time would be available to provide alternate cooling through the decay heat removal system utilizing existing piping and valve arrangements.

The cleanup system will maintain the quality of the pool water by recirculating one-half the volume of the spent fuel pool water through the purification loop per day. In addition, this purification loop has the capability of processing water from the fuel transfer canal or the borated water storage tank.

Based on our review, we conclude the spent fuel pool cooling and cleanup system is acceptable.

9.2.4 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 condition until it leaves after post-irradiation cooling. The system consists of the fuel transfer canal, the fuel transfer system, the spent fuel storage pool, the new fuel storage vault, and the fuel cask loading and storage area. The integrated fuel handling operations will basically be performed in two separate buildings, partly inside the reactor building and partly in the spent fuel storage area in the auxiliary building. The reactor building crane and the fuel handling crane including the crane hoist braking have been designed in accordance with Electric Overhead Crane Institute Spejification No. 61. The cranes and major components provided are of essentially standard design and similar to those we have found acceptable previously.

On the basis of our review, we have concluded that the spent fuel handling system is acceptable.

9.3 Water Systems

9.3.1 Service Water System

The service water system (SWS) provides cooling water to all components essential for the plant's safe shutdown and to nonessential water system such as the intermediate cooling system, auxiliary cooling water system, and the condenser circulating water pump bearing lubrication system. The SWS, which acts as an intermediate heat sink for all vital components, receives its water supply from the Dardanelle Reservoir during normal operation and the emergency cooling pond during accident conditions.

The SWS consists of two independent and redundant, full capacity subsystems. Three 100 percent capacity motor driven pumps which are located in a seismic Category I section of the intake structure (the intake structure and associated sluice gate operations are discussed separately under Section 9.3.4, Ultimate Heat Sink) and appropriate valving and cross-connections have been provided to supply service water to either SWS subsystem. In addition, the SWS is designed so that each subsystem and all components connected to it are capable of being isolated on an individual basis. Our independent evaluation indicates that the SWS is capable of providing continuous cooling during all operation conditions in the event of any single active failure or a single passive failure during post-accident long term cooling.

During an accident condition including loss of offsite power, two of the service water pumps will be powered by the emergency diesel generators. Either pump is capable of supplying the minimum essential service water requirements during and following an accident. Our review indicates that for the accident condition, the service water isolation valves will isolate all nonessential cooling systems by an appropriate safety actuation signal or by operator action and the essential components not normally operating will be automatically placed in operation by these same safety signals.

We conclude that the Service Water System is acceptable.

9.3.2 Intermediate and Auxiliary Cooling Water Systems

The intermediate cooling system provides cooling water to both nuclear and non-nuclear components and is a closed loop system. The cooling water pumps and heat exchangers are located in the auxiliary building and are not required for safe plant shutdown. The intermediate closed loop system provides a barrier between the reactor coolant system and the service water system to prevent the accidental release of radioactivity. However, failure of the system would not prevent a safe shutdown. A radiation monitor has been installed upstream of the heat exchangers to monitor for possible in-leakage of radioactive fluids to this system.

The auxiliary cooling water system, which supplies cooling requirements for only non-nuclear related components, also is not required for safe shutdown. Accordingly, this system is designed to be isolated in the event of an accident condition.

Based on our review of the intermediate and auxiliary cooling water systems, we have determined that the failure of any component in these seismic Cagegory II systems will not affect the service water system or any other safety related system.

We conclude that the intermediate and auxiliary cooling water systems are acceptable.

9.3.3 Decay Heat Removal System

The decay heat removal (DHR) system will normally remove reactor heat after a conventional shutdown where the steam generators have cooled the reactor coolant system to a temperature of 280°F. The DHR, in conjunction with the BWST, will also be used to provide low pressure injection in the event of a loss-of-coolant accident as discussed in Section 6.3 of this report. The DHR system consists of two redundant decay heat removal pumps and coolers. We have determined that redundancy of components, valves, and piping provides adequate protection from the effects of a single active or a single passive failure during post-accident long term cooling.

In the decay heat removal mode of operation, the DHR system will take hot reactor coolant from the reactor coolant system outlet line and after removal of the decay heat, discharge the coolant back to the reactor through the core flooding nozzles. The suction line to the DHR system pumps contains three electric motor-operated gate valves in series, the pump discharge line contains an electric motor-operated valve and a check valve in series. The suction line valves are interlocked with reactor coolant system pressure in such a manner that the valves will not open when the reactor coolant pressure exceeds the design pressure of the DHR system. The applicant has agreed to provide additional interlocks to automatically close the valves, if open, when the reactor coolant system pressure reaches or exceeds the decay heat removal system design pressure (see Section 7.4.2 of this report).

On the basis of our review of the system design and functional requirements and the applicant's commitment regarding additional interlocks, we conclude that the DHR system is acceptable.

9.1.4 Ultimate Heat Sink

Two sources of cooling water are available for reactor equipment to use as an ultimate heat sink, the Dardanelle Reservoir and an onsite emergency cooling pond. The emergency cooling pond is a seismic Category I structure which will be used for both normal and emergency operations.

Cooling water flow from the Dardanelle Reservoir will be terminated and flow from the emergency pond will be initiated during normal plant shutdown, accident conditions, and whenever the reservoir drops to an unacceptable low level (a low level alarm in the reservoir is annunciated in the control room) by actuating remote-manual motor-operated sluice gates. The results of our independent failure analysis of sluice gate operations indicate that for all failure modes an adequate supply of essential cooling water will be assured. The sluice gates have controls and indications in the control room and would be powered from the engineered safety feature buses during an accident condition with the loss of all offsite power.

Cooling water will be supplied by gravity flow from the pond through seismic Category I supply lines to the service water pumps located in the intake structure. The intake structure is designed to withstand the effects resulting from the PMF, the SSE and tornadic forces and missiles as described in Sections 3.3 and 3.5 of this report. The cooling water will be returned to the emergency pond after being circulated through the service water system to remove decay heat from the reactor facility. Additional detailed information pertaining to our evaluation of the design of the pond as an ultimate heat sink is contained in Section 2.4 of this report.

Based on our evaluation of the Ultimate Heat Sink, we conclude that the design, which meets the position set forth in Regulatory Guide 1.27, "Ultimate Heat Sink", is acceptable.

9.4 Process Auxiliaries

9.4.1 Compressed Air System

The compressed air system provides air for operation of the air-operated isolation valves. These valves are designed to remain in or revert to a safe position in the event the compressed air supply fails. Three valves are exceptions in that they are required to operate even after air failure; they have been provided with individual air receivers to provide an assured source of air. We conclude that the compressed air system is acceptable.

9.4.2 Sampling System

The sampling system is essentially the same as that used in previously approved reactor facilities. Process sample lines that are connected to a Category I system have two seismic Category I isolation valves in series to assure that failure of the Category II sample line will not effect the integrity of the connecting Category I system. We conclude that the Sampling System is acceptable.

9.4.3 Chemical Addition and Makeup System

The chemical addition and makeup system is designed to: (1) adjust the concentration of boric acid in the reactor coolant for reactivity control; (2) provide the reactor coolant system with fill and operational makeup water; (3) maintain the proper concentration of hydrogen, oxygen, and corrosion inhibiting chemicals in the reactor coolant system; (4) provide seal injection water for the reactor coolant pumps; and (5) in conjunction with the pressurizer, correct for changes in the reactor coolant due to temperature changes. The chemical addition system is used to control the concentration of various chemicals in the reactor coolant system during reactor operations. The makeup system controls the reactor coolant inventory and concentrations of chemical additives through the process of letdown and makeup.

The chemical addition system injects chemicals into the reactor coolant system or the auxiliary systems during normal reactor operations. The addition system consists of the following subsystems: the boric acid addition subsystem to provide concentrated boric acid to the BWST, makeup tank, or the spent fuel pool; the lithium hydroxide subsystem to provide LiOH solutions to the makeup and purification system for pH control of the reactor coolant; and the hydrazine subsystem to provide hydrazine to the reactor coolant system to scavenge dissolved oxygen. These subsystems have been designed as Category II systems.

The makeup system, during normal operation, utilizes part of the pumps, valves, and piping of the high pressure injection system. One pump takes suction from the makeup tank to provide water to the seals of the reactor coolant pumps and to the makeup line. A portion of the seal water supply is also injected into the reactor coolant system. This inleakage necessitates a continuous letdown flow of reactor coolant. Makeup flow to the reactor coolant system is automatically controlled by control valves that operate on signals from the pressurizer level controller. The makeup tank serves as a receiver for letdown flow, chemical additions, and demineralized (unborated) water makeup. The flow of unborated water to the makeup tank is measured by an inline flow integrator (batch controller) and associated instrumentation and is controlled remotely by the makeup control valve. The dilution cycle is initiated by the operator. Therefore, the following measures have been incorporated in the design to prevent excessive dilution of the boron concentration in the reactor coolant by makeup: (1) the dilution valves have been interlocked so that the predetermined dilution batch size must be preset prior to initiating the dilution cycle, (2) the cycle will be automatically terminated when the integrated dilution flow equals the preset batch size, (3) the regulating control rod bank has been interlocked to automatically terminate the dilution cycle, and (4) the operator can manually terminate the cycle at any time.

Based on our review, we conclude that the chemical addition and makeup system is acceptable.

9.5 Air-Conditioning, Heating, Cooling, and Ventilation System

9.5.1 Control Room

The normal air-conditioning system for the control room and computer room consists of two redundant trains, one of which is normally operating, with the other in standby status isolated from the system. The standby unit is available by manual actuation in the event of equipment failure in the operating unit. A small portion of the system's air is supplied to the relay and cable spreading rooms for pressurization to prevent inleakage of air from the turbine building.

With the original design, the normal air-conditioning system was continuously monitored only for high radiation. However, in Amendment

No. 34 the applicant agreed to install smoke detectors in the air supply duct to preclude significant quantities of smoke from entering the control room. A high radiation or smoke detection alarm will automatically deenergize the normal air-conditioning system and isolate the control room. In the isolated condition the control room is air-conditioned by two redundant packaged air-conditioning units located within the control room. These Category I fan-filter units consist of a fan, a roughing filter, a HEPA filter, and a charcoal filter. The control room emergency air-conditioning system is powered by the diesel generators, thus, the isolated control room is capable of operating on only recirculated air through the course of the accident. Based on our evaluation of the failure mode and effects analysis we have determined that the design of the normal and emergency control room ventilation and air-conditioning systems meet our single failure criterion.

In the event of a fire in the control room, provision has been made to preclude the recirculation of smoke-laden air and to supply outside air while exhausting the control room air outside the building.

We conclude that the control room normal and emergency air-conditioning and ventilation systems are acceptable.

9.5.2 Auxiliary Building

The auxiliary building has separate ventilation systems serving auxiliary equipment areas such as the spent fuel pool areas, the non-radioactive area, the radwaste area, other radwaste areas and the control room area. Air flow is designed to maintain movement from clean or low-activity areas to areas of progressively higher potential activity to preclude the spreading of radioactive contamination. The ventilation air from the fuel handling, radwaste, and the other potentially contaminated areas is continuously discharged through roughing filters, HEPA filters, and charcoal filters to the reactor building vents. These ventilation exhaust systems have been provided with redundant automatically started fans to ensure continuous ventilation of the areas.

Equipment in areas that must remain operable during and after a DBA have been provided with redundant, seismic Category I air-conditioning and ventilation systems. The control room, makeup pumps rooms, decay heat removal rooms, switchgear rooms, diesel generator rooms, and reactor building penetration rooms of the auxiliary building all have Category I emergency air-conditioning and ventilation systems.

We conclude that the design of the auxiliary building air-conditioning and ventilation system is acceptable.

9.5.3 Turbine Building

The turbine building ventilation system is a once-through system composed of three subsystems. The two subsystems that provide ventilation to the operating floor are also designed to operate with a recirculated air system, with provision for fresh air makeup. The fresh air intake ducts are provided with power operated dampers to provide isolation capabilities. Exhausted air is discharged directly to the atmosphere through roof ventilators. We find this system to be acceptable.

9.6 Other Auxiliary Systems

9.6.1 Fire Protection System

The fire protection system (FPS) is designed to meet the requirements of the National Fire Protection Association (NFPA) and of the Nuclear Energy Property Insurance Association (NEPIA). This includes inspection and approval of the fire protection system and its equipment by a NEPIA inspector.

The FPS piping located within Category I structures, with the exception of the intake structure, is Category I piping. In non-Category I structures the FPS piping was designed in accordance with standard NFPA requirements. With regard to the exception, we have determined that a failure of the non-Category I piping in the intake structure and the continued operation of the fire pumps will not flood essential Category I equipment and that in all areas, including the intake structure, where the system does not use Category I components, physical separation has been employed to assure that the failure of the Category II system will not have an adverse effect on a Category I system located in the same structure. In the emergency diesel generator rooms, the FPS headers are equipped with fusible heads but kept dry. FPS water to the headers is controlled by remote-manual valves, located outside the room they serve, and operated from the control room.

Expanded fire protection is provided around the property of the plant building complex by yard hydrants spaced at 250 foot intervals and each is capable of being isolated. The internal fire protection for general plant areas is provided by hose stations located such that all building interiors are protected by the various 50-foot hose lengths, and strategically located portable dry chemical and CO_2 fire extinguishers. The fire protection for specific plant areas is provided by: (1) automatic deluge systems in the turbine lubricating oil storage tank area, the turbine lubricating oil reservoir area, the hydrogen seal oil unit area, and the feedwater pumps lubricating oil reservoir; (2) fusible head sprinkler systems are located to protect the floor area under the turbine on the lubricating oil piping side of the turbine pedestal, the oil piping, the fuel oil storage tank, the intermediate floor and grating floors; and (3) remote manual or locally operated sprinkler systems protect the engineered safety features cable areas in the reactor building and in the penetration rooms, the emergency diesel generator rooms, and in the emergency generator diesel fuel oil storage vaults.

The fire detection system consists of alarms in the control room which annunciate upon operation of any of the individual systems. The detectors utilized to actuate systems are the "rate of rise" type for the automatic deluge system, and the smoke and heat ionization type for the manual sprinkler systems.

We conclude that the fire protection system is acceptable.

9.6.2 Communication System

The onsite intraplant communication system consists of two plant telephone and paging systems with redundant power supplies to provide the control room operator with constant communication with all vital areas of the plant during normal plant operations. Acoustic booths have been provided in areas where the potential background noise levels are high. However, it is not certain that necessary communication can be maintained during accident or incident conditions while maximum potential noise levels obtain. The applicant has agreed to perform noise level tests during preoperational testing to demonstrate the adequacy of the system to provide communication between all vital plant areas.

We conclude that the communication system is acceptable.

9.6.3 Diesel Generator Fuel Oil Storage, Transfer and Auxiliary Systems

The diesel generators are housed in separate rooms located in Category I, tornado protected portions of the auxiliary building. The diesel generators are located above the PMF water level calculated for this facility. Each diesel generator room is self-sufficient and protected from one another for fire, flooding, and internally generated missiles. The seismic Category I diesel generator fuel oil storage and transfer system consists of redundant 20,000-gallon emergency storage tanks, day tanks, transfer pumps, and associated cross-connected piping and valves. Each emergency storage tank and transfer pump unit is contained in a fire, tornado, and flood proof seismic Category I underground vault and is capable of supplying sufficient fuel oil to operate one diesel generator for 7 days at full load. Appropriate valving in the fuel oil transfer system is provided to enable either transfer pump to take suction from either fuel tank and to discharge to either diesel generator day tank.

The diesel generators have been provided with independent auxiliary systems, such as cooling water system, starting system, lubrication system, and air intake system. The design and location of these subsystems are such that a single failure in any one system will not disable both diesel generator units.

Based on our review, we conclude that the diesel generator fuel oil storage, transfer, and auxiliary systems are acceptable.

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is of a conventional design, similar to those of previously approved plants. The system is designed to remove heat energy from the reactor coolant by two B&W steam generators and convert it to electrical energy by a Westinghouse turbine driven generator. The steam condenser transfers waste heat in the cycle to the condenser circulating water system. The entire system is designed for the maximum expected energy from the nuclear steam supply system. Upon loss of full load, the system is designed to reduce the rate of power generated and dissipate the energy stored in the system at time of load loss through bypass valves to the condenser and through power operated dump valves to the atmosphere.

10.2 Turbine Generator

The turbine generator is a tandem compound, three element machine consisting of one double-flow high pressure stage followed by two double-flow low pressure stages. The ac generator with its excitation system is connected to the turbine shaft. Steam extraction for feedwater heating is provided, and a combination of moisture separation and live steam reheating is used between the high pressure and the low pressure stages. The moisture from the separators is returned to the feedwater system.

10.3 Main Steam Supply System

The steam from each of the two steam generators penetrates the reactor building in a single steam line. Each steam line is provided with a main steam block valve close to the reactor building between the reactor building penetration and the turbine stop valve. This valve serves as an isolation valve to prevent blowdown of the unaffected steam generator in the event of a steam line break between the steam generator and the turbine stop valve. The steam line from each steam generator up to and including the block valve is Category I design. The applicant has provided these block valves with automatic isolation controls in order to mitigate the consequences of a postulated steam line break.

Overpressure protection for the main steam supply system is provided by eight spring-loaded code safety valves and one power-operated relief valve on each main steam line, which relieve to the atmosphere. These valves are all connected upstream of the block valves. The pressure relieving capacity of each safety valve is approximately 846,000 lb/hr. The total relief capacity of the safety valves only is equal to the energy generated at the reactor's highest power level trip setting.

Based on our review of the Main Steam Supply System we conclude that it is acceptable.

10.4 Steam and Power Conversion System

10.4.1 General

Subsystems of the steam and power conversion system, such as main condenser and evacuation system, turbine gland sealing system, condensate clean-up system, and condensate and feedwater systems, are similar in design to those of previously approved facilities.

We conclude that the above subsystems of the Steam and Power Conversion System are acceptable. Others are discussed below.

10.4.2 Turbine Bypass System

The turbine bypass system is designed to divert a total steam flow equivalent to 15 percent of main feedwater flow directly to the condenser. Each of four 6-inch bypass lines is provided with a pneumatically operated pressure reducing control valve. These bypass lines connect to the steam lines downstream of the block valves and have been provided with manual isolation valves upstream of the control valves for isolation in the event of malfunction of the control valves.

As mentioned in Section 10.1 of this report, the bypass system allows a sudden loss of load from full power without reactor trip, provided the control system functions to reduce reactor power. The safety valves (see Section 10.3 of this report) relieve excess steam until the reactor output is reduced to the point that the steam bypass to the condenser and the atmospheric dump valves can handle all the steam generated.

We conclude that the turbine bypass system design is acceptable.

10.4.3 Circulating Water System (CWS)

The CWS supplies condenser cooling water to four individual condensers through four circulating water pumps. Each pump is designed to supply approximately 192,000 gal/min of water. The CWS piping is cross-connected downstream of the pump discharge to permit any pump to supply any condenser. The CWS pumps take suction from a portion of the intake structure which in turn is supplied with Dardanelle Reservoir water through the intake canal.

Although the CWS supplies cooling water for both normal and turbine bypass operation of the condenser, the system is not essential for safe plant shutdown.

In response to our request, the applicant has stated that a failure of any component in the CWS such as pipe breaks, pump failure, or expansion joint rupture will not result in the loss of components or systems necessary for safe shutdown due to the resultant flooding.

The CWS pumps are located outside the floodtight section of the intake structure which contains safety-related equipment. The CWS lines run to the turbine building in backfilled trenches. In the turbine building the CWS is located at or below grade level. A CWS failure in the turbine building would not adversely affect safety-related equipment. The safety-related euqipment in the turbine building is limited to water-proof cable. The turbine building is not watertight at grade level 353 feet MSL, and has walls only of sheet metal above 361 feet MSL. Therefore, flooding would not reach safety-related equipment in the auxiliary building since the auxiliary building is flood protected to 369 feet MSL.

On the basis of our review, we have concluded that the CWS is acceptable.

10.4.4 Turbine Overspeed Protection

The turbine is provided with overspeed protection by two primary protection systems operating on diverse principles and an independent backup system with redundant trip circuits. The first limit of 103 percent of turbine shaft peed is provided by the speed governor action of an electro-hydraulic control system. A second limit of 111 percent of rated speed is provided by a mechanical overspeed trip device which is actuated by a spring-loaded eccentric ring mounted on the end of the turbine shaft. In addition, an independent and redundant backup electrical overspeed trip circuits have been provided. Each circuit senses the turbine speed by means of a magnetic pick up which monitors the speed of the main turbine shaft. At 111.5 percent of rated speed, the master trip solenoid valve is deenergized which releases the emergency trip system hydraulic pressure and closes all turbine valves, including the turbine stop, control, and reheat intercept valves.

The system described above will limit the turbine to approximately 120 percent of rated speed. To exceed this rate, it would take the simultaneous failure of two independent systems plus a failure of the back-up systems.

We conclude that the provisions for turbine overspeed control are acceptable.

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 General Description

The waste treatment systems have been designed to provide for controlled handling and disposal of radioactive liquid, gaseous, and solid wastes. The liquid and gaseous radwaste systems have been designed to control releases of radioactivity to within 10 CFR Part 20 limits. In addition, the applicant has agreed to maintain and use existing plant equipment to achieve as low as practicable radioactive releases to the environment in accordance with the requirements of 10 CFR Part 50.

The liquid waste treatment system is comprised of several subsystems, which collect wastes from several specific sources and process the waste separately. Cross connections between subsystems allow flexibility for processing these liquid wastes.

The gaseous waste treatment system provides holdup capacity for fission product gases stripped from the reactor coolant to permit decay of short-lived radioactivity before release to the environment.

The solid waste system is designed to package waste in accordance with the regulations set forth in 10 CFR Parts 70-71 and Department of Transportation shipping regulations.

The liquid and gaseous waste treatment systems were evaluated to determine their capability to reduce the effluents that would be released to unrestricted areas such that:

(a) the annual total quantity of liquid releases does not exceed 5 curies;

(b) the annual exposure rate due to noble gases at the site boundary does not exceed 10 mrem;

(c) the annual dose to an organ or whole body of any individual from gaseous radioiodine through the food chain does not exceed 5 mrem.

11.2 Liquid Wastes

To reduce radioisotopic releases from Unit 1 to as low as practicable, the applicant will design the radwaste system of Unit 2 to process the waste liquids from both units. Since the construction and operating schedules indicate that the Unit 2 system will not be available until about two years after Unit 1 begins operating, two estimates of liquid radioactive discharges are presented, one for the initial period and a second for the period after the Unit 2 radwaste system is operational.

The liquid radioactive waste system for Unit 1 consists of collection tanks, piping, pumps, demineralizers, process equipment and instrumentation necessary to collect, process, monitor, store and dispose of potentially radioactive liquid wastes. The Unit 2 radwaste system when operational will provide additional treatment by evaporation and demineralization. The Unit 1 radwaste system is divided into four main subsystems; (1) the reactor coolant treatment system (RCTS) which includes the chemical and volume control system (CVCS), (2) the clean liquid waste treatment system (CWTS), (3) the dirty liquid waste treatment system (DWTS), and (4) the laundry waste system (LWS). Waste is classified as clean or dirty waste on the basis of conductivity and not radioactivity. Treatment of the wastes is dependent on the source, activity, composition, and intended disposal procedure.

Treated wastes will be handled on a batch basis to permit optimum control of release. Prior to release of any treated liquid wastes, samples will be analyzed to determine the type and amount of radioactivity in a batch. Based on the results of the analysis these wastes will either be released under controlled conditions, retained for additional decay or processed further. Radiation monitoring interlocks will automatically terminate liquid waste discharge if radiation levels are above a predetermined level in the discharge line.

The primary function of the reactor coolant treatment system (RCTS) is to process the coolant letdown stream to maintain reactor coolant water quality and boron concentration at the proper levels. Part of the treated reactor coolant will be removed from the RCTS and fed to the clean liquid waste treatment system (CWTS) to permit adjustments in boron concentration in the reactor coolant system. The RCTS will also collect excess reactor coolant that results from startup expansion and startup boron dilution. We estimate that about 720 gpd of excess reactor coolant and about 865 gpd of boration bleed on an annual average basis will enter the CWTS system from the RCTS.

Clean liquid wastes are collected in the auxiliary building equipment drain tank from reactor loop leaks, sample sinks, make-up system, etc. These wastes will also be sent to the CWTS. In our evaluation, we assumed that an additional 515 gpd will be sent to the CWTS from the sources. Liquid wastes in the CWTS will be processed through a filter and two deep-bed demineralizers in series until the Unit 2 radwaste system is operational. After Unit 2 radwaste equipment is operational, these wastes will receive additional treatment by evaporation and demineralization. In both cases, the processed clean liquid wastes will be collected in the clean waste monitoring tank where they will be analyzed for radioactive concentration. Depending on the activity present, the wastes will either be given additional treatment or discharged to the circulating water canal. In our evaluation, we assumed that all effluent from the CWTS would be discharged following the above treatment without being recycled for additional decontamination.

The dirty liquid waste treatment (DWTS) system will collect and treat liquid waste from floor and equipment drains, leakoffs, wastes from laboratory drains and decontamination drains. In our evaluation we assumed that about 330 gpd will be collected from these sources. These wastes will be processed through a filter and collected in the filtered waste monitoring tank until Unit 2 waste equipment is operational. Based on the results of the analysis of the filtered wastes, the wastes will be sent to the CWTS for additional treatment or will be discharged to the circulating water canal. After the Unit 2 waste equipment is operational, these wastes will receive additional treatment by evaporation and demineralization. In our evaluation we assumed that all the effluent from the DWTS will be discharged to the circulating water canal following the above treatment without being recycled for additional decontamination.

The B&W once through steam generators used in this plant do not require steam generator blowdown which is a potential source of liquid radwaste in other PWR's. However, secondary coolant purity is maintained by treating 70% of the feedwater flow by six deep-bed demineralizers upstream of the feedwater train. These demineralizers are regenerated periodically and the regenerant is processed through the dirty liquid waste treatment system. In our evaluation we assumed a 30-day regeneration cycle, a holdup time of 6 days and steam generator tube leakage of 20 gallons/day.

The applicant estimated releases of about 20 curies per year from the liquid radioactive waste system, excluding tritium and dissolved noble gases. This estimate is for the treatment process prior to installation of Unit 2 radwaste equipment. No estimate has been made by the applicant of radioisotopic releases after Unit 2 waste equipment is installed. The applicant's release estimate is based on 0.1% fission product inventory release from the fuel, 371,000 gallons per year of clean wastes with twenty days holdup, and 147,000 gallons per year of dirty wastes with a one day holdup.

Based on our independent evaluation, we estimate releases of about 35 Ci/yr (excluding tritium and dissolved gases) before installation of Unit 2 equipment and 0.1 Ci/yr (excluding tritium and dissolved gases) after installation of the Unit 2 radwaste equipment. These estimates assume 0.25% fission product inventory release from the fuel, 80% operating factor, 28 days holdup for clean wastes, and five days holdup for dirty wastes. During the first two years of reactor operation, the applicant estimates an operating factor of about 65% with the first fuel cycle extending over the two years. On that basis, the releases for that period of time could be lower than our estimates. Based on present operating experience, we estimate 1000 Ci/yr of tritium will be released from Unit 1.

The liquid waste treatment system has been designed to collect, process, and store waste from operation with up to 1% fission product inventory release to the reactor coolant. We conclude from our evaluation that the radioisotopic releases from Unit 1 can be controlled well within the limits of 10 CFR 20 for up to

1% fission product release and consider that such releases will be as low as practicable in accordance with 10 CFR Part 50 for normal operation.

On the basis that the calculated dose from liquid radioactive wastes is less than 5 Ci/yr after the augmented system is installed, and on the basis that the calculated dose from expected releases with the present system is less than 5 mrem/yr, we conclude that the liquid waste treatment system is acceptable. The applicant will install the Unit 2 liquid radwaste equipment during early construction of Unit 2 and has committed to have it operational at the beginning of the second fuel cycle for Unit 1.

11.3 Gaseous Wastes

The waste gas treatment system (WGTS) consists of gas decay tanks, piping, high-efficiency particulate filters, charcoal adsorbers and instrumentation necessary to collect, store, process, monitor and dispose of potentially radioactive gaseous wastes. The purpose of the WGTS is to maintain an inert cover gas of nitrogen in tanks and equipment that contain potentially explosive gas, hold up radioactive gas for decay and release (radioactive gases mixed with nonradioactive gases) to the atmosphere under controlled conditions.

The major source of gaseous waste activity during normal operation will be the waste gases removed by the degassifier from the reactor coolant that is letdown, evolution of gases from the various liquid holding tanks, displacement of nitrogen cover gas from liquid-storage tanks, and pressurizer vents. Additional sources of gaseous waste activity which are not concentrated enough to permit collection and storage include the ventilation air released from the auxiliary building and the turbine building, the exhaust from the condenser air ejectors, and the air purged from the reactor building.

The gaseous waste received by the WGTS (mostly hydrogen with small amounts of entrained noble fission gases) is collected in the waste gas surge tank, compressed by one of two compressors and sent to one of four waste gas decay tanks having a design capacity of 2,500 standard cubic feet at a pressure of 123 psig. As liquid storage tanks are filled, the displaced cover gas is compressed and stored in a waste gas decay tank. When a specified tank pressure is reached, the contents of that tank are sampled and analyzed to determine the permissible release rate or the need for further hold-up for radioactive decay. The contents of the decay tank will be discharged under controlled conditions through the unit vent which is 200 ft. above ground level. Continuous monitoring of the discharged waste is provided by a radiation monitor which, on a high radiation signal, will actuate an interlock to close the valves through which the gas is being discharged.

The applicant has estimated a holdup time capability for the waste decay tanks of 30 days. Our evaluation is based on 30-day holdup for decay even though we calculate that additional holdup time is available. The applicant has estimated the total average annual noble gas release to be 1227 Ci/yr and the I-131 release to be essentially zero. By independent analysis we have estimated that an annual average of about 1430 Ci/yr of noble gases and essentially no I-131 will be released from the gas decay tanks.

Radioactive gases may be released inside the reactor containment building when components of the reactor coolant system are opened to the building atmosphere for operational reasons or when minor leaks occur in the reactor coolant system component seals. Provisions have been made to periodically purge the reactor building air through prefilters, HEPA filters and charcoal adsorbers with release to the environment through the monitored unit vent.

The applicant has estimated the noble gas release to be approximately 80 Ci/yr and the I-131 release to be approximately 0.0014 Ci/yr from the venting of the reactor building four times a year. Based on our independent evaluation, we have estimated an annual release of about 150 Ci of noble gases and 0.04 Ci of I-131 from the venting of the reactor building four times a year.

Radioactive gases may also be released to the auxiliary building through leaks and open equipment. To reduce the subsequent release of radioactive materials to the environment, the auxiliary building will be maintained at negative pressure with respect to the outside pressure. Ventilation air will move from areas of low contamination potential towards areas of higher potential. Gases purged from the auxiliary building will be discharged through pre-filters, HEPA, and charcoal filters through the plant vent where they will be monitored prior to discharge to the environment. The applicant does not consider this source as significant. We estimate an annual release fromtthe auxiliary building of about 855 Ci of noble gases and less than 0.01 Ci of I-131 for Unit 1.

Most of the turbine building exhaust will be released through roof-mounted exhaust fans. About 10% of the turbine building ventilation air flow will be exhausted to other spaces for ventilation; e.g., battery rooms, switchgear areas, etc. All the turbine ventilation air will be released without treatment. We do not expect this to be a contributing source of gaseous radioactive release to the environment.

Radioactive gases which may enter the secondary coolant loop, along with air inleakage, will be removed by the mechanical vacuum pumps. These gases will be discharged through prefilters, HEPA, and charcoal filters to the unit vent. The applicant has estimated the release to be about 2600 Ci/yr of noble gases and 0.00375 Ci/yr of I-131. We estimate an annual release rate of 860 Ci/yr of noble gases and less than 0.01 Ci/yr of I-131.

The applicant estimates total annual release of about 4000 Ci of noble gases and 0.00515 Ci of I-131 from Unit 1. This estimate is based on 0.1% fission product release from the fuel, 100 gpd of reactor coolant leakage into the reactor building, and 100 gpd of reactor coolant leakage into the steam generator. Our evaluation of the gaseous radwaste treatment system is based on 0.25% fission product inventory release from the fuel, 40 gpd of reactor coolant leakage into the reactor building, 20 gpd of reactor coolant leakage into the secondary system, 20 gpd reactor coolant leak into the auxiliary building, and 5 gpd of condensate leakage in the turbine building. From our evaluation, we estimate a total annual release of 3300 Ci of noble gases and 0.06 Ci of I-131. Radiation doses to individuals at the site boundary from gaseous effluents were calculated to be less than 1.0 mrem/year whole body and 1.5 mrem/year to a two-year old child's thyroid. The thyroid dose is based on the child drinking one liter of raw milk per day that is derived from a cow feeding on the nearest pasture which is two miles from the plant.

The gaseous waste treatment system has been designed to collect, process, and store waste from operation with up to 1% fission product inventory release to the reactor coolant. We conclude from our evaluation that the radioisotopic releases from the plant can be controlled to well within the limits of 10 CFR Part 20 for up to 1% fission product inventory release to the reactor coolant and are considered to be as low as practicable in accordance with 10 CFR Part 50 for normal operation. We conclude that the gaseous waste treatment system is acceptable.

11.4 Process and Area Radiation Monitoring System

The process radiation monitoring system is designed to provide information on radioactivity levels in certain systems, leakage from one system to another, and radioactivity released to building spaces and to the environment. The monitoring will include: the reactor building air, the gas waste system tanks, mechanical vacuum pump discharge, the plant vents (gas, iodine, and particulate), the service water system, and liquid discharges. All building spaces that could significantly contribute to the source of airborne radioactive release from the plant are served by the ventilation system described in Section 11.3 of this report. Ventilation exhausts from these spaces are routed to the plant vents where the releases are monitored.

The area radiation monitoring system is designed to provide information on radiation fields in various areas of the plant for personnel protection. Twenty monitors are located throughout ANO-1. Areas protected include the control room, spent fuel pool area, radwaste area, turbine building, and primary containment. Radiation monitor alarms and activity level indications are provided in the control room and on local area radiation monitoring panels.

These monitoring systems will detect, indicate, annunciate and/or record as required the levels of activity to keep radiation levels as low as practicable and to verify compliance with existing regulations.

We conclude that the process and area monitoring equipment satisfies the requirements of Regulatory Guide 1.21 and General Design Criterion 64 for effluent discharge paths and, therefore, is acceptable.

11.5 Solid Wastes

Solid wastes from station operation will be composed primarily of spent resins, air filters, and miscellaneous paper and rags. Radioactive resins from the demineralizers will be collected and stored in the Spent Resin Storage Tank until ready for disposal. The resins are transferred to appropriate containers and solidified when ready for shipping. The applicant makes no estimate of the quantity of solid waste that will be generated. Based on experience at similar facilities, we have estimated that 235 drums of resins and filters will be generated at the station per year. We estimate that each drum will contain about 20 Ci of radioactivity after 180 days decay. Miscellaneous dry wastes will be compacted in drums. We have estimated that 600 drums of this waste will be generated per year with a total activity less than 5 Ci after 180 days decay. All solid wastes will be packaged and shipped to a licensed storage area in accordance with AEC and Department of Transportation regulations.

The storage and packaging facilities described are similar to those previously reviewed and found acceptable for other reactor facilities. We conclude that the ANO-1 solid waste system is acceptable.

11.6 Offsite Radiological Monitoring Program

The applicant has undertaken a radiological environmental monitoring program which has as objectives: (1) the establishment of existing levels of background radioactivity, (2) the identification of potentially significant pathways of radionuclides released from the plant to man, and (3) the determination of the effect of plant operation on the environment. The program is comparable in scope to those of other nuclear facilities currently in operation or being licensed.

Preoperational measurements were started approximately one year prior to the anticipated fuel loading date. The first quarter of the program has been completed but results are not yet available. The Arkansas State Department of Health established a state sampling program in 1956. Results from this program in the vicinity of the site for the years 1969 and 1970 were included in the Applicant's Environmental Report. No significant deviations from expected background values are noted in these results.

Airborne radioactivity is monitored at four locations onsite, at two locations within a 10 mile radius of the Station and at one control location 20 miles from the Station. Radionuclide concentrations in air, in vegetation, and in soil will be measured at these locations as will integrated gamma-ray doses (TID). Precipitation will be collected at four of these locations which are situated in the principle wind directions. Collection frequencies range from weekly for continuous-air-sampling filters to semi-annually for soil samples. Analysis will consist of measurement of gross radioactivity and gamma spectra. Soil will also be analyzed for Sr-89 and Sr-90. Liquid effluents are monitored by taking samples of water, fish, aquatic plants, and bottom sediments at two locations in the discharge embayment and two locations in the main body of the Dardanelle Reservoir. A fifth sampling station downstream of the reservoir was recommended in the Final Environmental Statement and will be provided. Drinking water will be sampled from three wells in the area and from the Russellville city water system intake. For all samples, gross radioactivity and gamma spectra are measured. Fish will be analyzed for Sr89, 90 as well as for gamma-ray emitters. The sampling frequency will be quarterly except for aquatic biota which will be sampled twice yearly.

Preoperational milk sampling has been carried out by the Arkansas State Department of Health which collects milk quarterly from six local herds and analyzes for specific gamma-ray emitters as well as Sr-89, 90. The Technical Sepcification requirements for sampling frequency and sensitivity of analysis for the operational phase will reflect the most recent staff requirements.

We conclude that the applicant's program will be adequate for monitoring the radiological impact of plant operation on the environs and for verifying the adequacy of in-plant monitoring and control of radioactivity with regard to the health and safety aspects of the release of radionuclides to the environment from the proposed operation of ANO-1.

11.7 Design Standards

The radioactive waste treatment system has been designed and fabricated in accordance with the following codes and standards:

Piping	ANSI 31.1
Low Pressure Radwaste Tanks	API 620
High Pressure Radwaste Tanks	ASME III Class C
Radwaste Demineralizer	ASME III Class C
Vacuum Degasifier	ASME III Class C
Gaseous Radwaste System	Nuclear Quality Assurance
Valves and Support in Liquid Radwaste System	Nuclear Quality Assurance
Process Radiation Monitoring System	Nuclear Quality Assurance

The liquid radwaste system has not been designed to withstand the Safe Shutdown Earthquake, but is located in a structure designed to Category I requirements. The gaseous radwaste system is designed to withstand the Safe Shutdown Earthquake. We find that the radwaste treatment systems have been designed in accordance with acceptable codes and standards.

11.8 Conclusions

Based on our model and assumptions, we calculate an expected whole body dose of less than 10 mrem/yr from gases and less than 5 mrem/yr from liquids at the site boundary after the augmented treatment system is installed. We calculate the potential dose to a child's thyroid from the iodine food chain to be less than 5 mrem/yr. Based on our evaluation, we conclude that the liquid, gaseous and solid waste treatment systems are in accordance with 10 CFR Part 50 for normal operation.

We also conclude that the system is designed in accordance with acceptable codes and standards, that the process monitoring system is adequate for monitoring effluent discharge paths as specified in General Design Criterion 64 of 10 CFR Part 50, and that personnel protection systems satisfy the requirements of existing regulations regarding exposure of individuals to radiation.

12.0 RADIATION PROTECTION

12.1 Shielding

Radiation shielding has been designed for normal plant operation according to the objectives of 10 CFR 20. Allowable design dose rates for all controlled access areas of the plant correspond to a maximum whole body exposure of 1.25 rem per calendar quarter.

The reactor vessel and the primary loop components are shielded both by internal structures and the reactor building shell. Radiation levels in occupied areas outside the shell will be below 1 mrem/hr. Portable shielding will be provided on the reactor operating floor for additional personnel protection during periods of refueling and reactor vessel maintenance. Areas of the auxiliary building which contain radioactivity are shielded. Different systems are isolated from each other by individually shielded compartments. As far as practicable tanks, pumps, filters, demineralizers and piping containing contaminated materials are shielded by concrete for the protection of adjacent areas. Access for maintenance purposes is thus provided without unnecessary exposure to adjacent equipment. All areas which are frequently occupied by plant personnel are designed to receive an exposure rate of less than 1.0 mrem/hr during operation.

We conclude that adequate consideration has been given to shielding design to keep exposures within applicable limits and to reduce unnecessary exposures during normal operation of the plant. The effectiveness of the shielding provided will be evaluated by means of complete radiation surveys of the plant during initial low power reactor operation and during full power operations. These surveys will ensure that the radiation levels in all areas are below the maximum designated limits.

12.2 Health Physics Program

Personnel protection will be accomplished through administrative controls and procedures, through the use of protective equipment and verified by personnel monitoring. All work in controlled areas will require an appropriate Radiation Work Permit (RWP) which will require determination and evaluation of the radiological hazards associated with the job before issuance. Exposures in plant will be minimized by rotating personnel assigned to tasks in high exposure areas and by training, prejob planning, and practice runs. Extension tools will be used where feasible and equipment will be moved to lower radiation areas for maintenance if feasible and/or portable shielding will be provided. Permanent shielding is provided for all waste treatment components as described above.

Special protective equipment is provided which includes covering garments, shielding and self-contained air-breathing units. A change room and personnel decontamination facilities are also provided.

Personnel monitoring will normally be accomplished by Thermo Luminescent Dosimeter (TLD) badges or the equivalent. Direct-reading dosimeters, pocket high-radiation alarms, and extremity badges will be available for use when required. Bioassay and medical programs using the equipment available at the University of Arkansas Medical Center at Little Rock will be used to back up work done at the site.

We conclude that the applicant plans to implement a health physics program of sufficient scope to maintain in-plant exposures of personnel within applicable limits. Plant design criteria and health physics related equipment and procedures indicate the applicant's intent to minimize in-plant personnel exposure.

13.0 CONDUCT OF OPERATIONS

13.1 Plant Organization and Staff Qualifications

The ANO-1 staff will consist of approximately 70 full-time employees. The plant is under the supervision of the Plant Superintendent who reports to the Vice President and Chief Engineer through the Director Power Production. The Plant Superintendent will be responsible for the safe operation of the plant. He has an Assistant Plant Superintendent to assist in the execution of his supervisory responsibilities and to assume full responsibility in the Superintendent's absence. The plant staff consists of an operations group, maintenance group, nuclear engineering group and technical support group. In addition, a Quality Control Engineer reports through the Assistant Plant Superintendent to the Plant Superintendent.

The Operation Supervisor directs and coordinates the activities of the shift personnel. The applicant has proposed a five-man shift complement consisting of a Shift Supervisor (licensed as a Senior Reactor Operator) a Plant Operator and an Assistant Plant Operator (both licensed as Reactor Operators), an Auxiliary Operator and a Waste Control Operator. The crew size and license requirements are acceptable.

The Maintenance Supervisor will be responsible for organizing and conducting preventive maintenance and repairs of electrical and mechanical equipment. The Technical Support Engineer will be responsible for the maintenance and proper operation of all instrumentation, control systems, non-nuclear systems, radiation and health physics work, plant chemistry and water control. He will be assisted by a Results Engineer and a Chemistry and Radiation Protection Engineer. The Nuclear Engineer will be responsible for monitoring and evaluating core physics, core performance and for the performance of all nuclear instrumentation, control and protective systems.

The applicant has conducted a training program that included the following courses: Basic Nuclear Training for Supervisors and Management, Basic Fundamental Training for Shift Supervisors, Basic Fundamental and Nuclear Training for Supervisors and Operators, Operator Training at a Comparable Nuclear Power Station, Basic Radiolcgical Health and Reactor Safety and Hazards Evaluation Training, PWR Technological Training, PWR Simulator Training and On-the-job Training and Station Check-out. Members of the technical groups completed formal training specifically oriented to their assigned responsibilities.

The qualifications of key supervisory personnel with regard to educational background, experience and technical specialties have been reviewed and are in general conformance with those defined in ANSI N18.1, "Selection and Training of Nuclear Power Plant Personnel."

Technical support for the plant staff will be provided by the Design and Construction Section of the Power Production Department up to the time of commercial operation (they will still be available afterward to provide technical assistance as required,thereafter) and by the Operations and Maintenance Section of the Power Production Department. In addition Middle South Services, Inc. has established a Nuclear Fuel Management Group to provide technical assistance to the Middle South System Operating Companies, one of which is the Arkansas Power and Light.Company.

We have concluded that the organizational structure, the training and qualification of the staff for ANO-1 are adequate to provide an acceptable operating staff and technical support for the safe operation of the facility. Additional technical support during the startup test program will be required (see Section 14.1).

13.2 Emergency Planning

The applicant has established an organization and plans for coping with emergencies. The plan includes written agreements, liaison and communications with appropriate local, state and federal agencies that have responsibilities for coping with emergencies. The applicant has defined categories of incidents, including criteria for determining when protective measures should be considered and for the notification of offsite support groups. Arrangements have been made by the applicant to provide for medical support in the event of a radiological incident or other emergencies. Provisions for periodic training for both plant personnel and offsite emergency organizations have been include in the emergency plan.

We have reviewed the applicants emergency plan and conclude that it meets the criteria of Appendix E of 10 CFR 50, that adequate arrangements have been made to cope with the possible consequences of accidents at the site, and that there is reasonable assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

13.3 Safety Review and Audit

The safety review and audit function for ANO-1 will be conducted by the Plant Safety Committee and the Safety Review Committee. The Plant Safety Committee is advisory to the Plant Superintendent and will review all proposed tests, changes in plant operating procedures and design modifications. The Safety Review Committee is advisory to the Vice President and Chief Engineer and provides corporate management with a review and audit capability to verify that organizational checks and balances are functioning to assure continued safe operation and design adequacy of the plant. The Safety Review Committee is in general conformance with the review and audit provisions of ANSI N18.7, Standard for Administrative Control for Nuclear Power Plants. We will require that the provisions for the Safety Review Committee be maintained in general conformance with that Standard.

We conclude that the review and audit structure proposed by the applicant is acceptable.

13.4 Plant Procedures

Plant operations are to be performed in accordance with written and approved operating and emergency procedures. Areas covered include normal startup, operation and shutdown, abnormal conditions and emergencies, refueling, maintenance, surveillance and testing, and radiation control. All procedures, and changes thereto will be reviewed by the Operations Committee prior to implementation.

We conclude that the provisions for preparation, review, approval, and use of written procedures are satisfactory.

13.5 Industrial Security

The applicant has submitted an industrial security program that describes its provisions for the protection of ANO-1 from industrial sabotage. The information was submitted as proprietary information pursuant to Section 2.790 of the Commission's regulations. The applicant has agreed to make several alterations in the program. With these alterations we conclude that the program meets the criteria of Regulatory Guide No. 1.17, "Protection Against Industrial Sabotage," and is acceptable.

14.0 INITIAL TESTS AND OPERATIONS

The initial startup, including preoperational checkout of equipment, functional and system tests, fuel loading, initial criticality and power operation will be performed by the regular plant staff. Technical assistance will be provided by the Production Department, B&W and the Bechtel Corporation in the areas of operations management, shift support, nuclear engineering and instrumentation and control.

The applicant has agreed to a preoperational and startup testing program that is in accord with the AEC publications "Guide for the Planning of Preoperational Testing Programs," and "Guide for the Planning of Initial Startup." This program will provide an adequate basis to confirm the safe operation of the plant and is therefore acceptable.

15.0 ACCIDENT ANALYSES

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

In addition to the above accidents, consideration was given to a postulated fuel cask drop accident because of a unique aspect of the plant arrangement. When a cask is being lifted or lowered through the equipment hatch, it could potentially be dropped a distance greater than 30 feet, the free drop distance required by 10 CFR Part 71 to be considered in the design of shipping casks. Therefore, in the unlikely event of such a drop, a cask failure and partial release of radioactivity was assumed to occur. Our final analysis of the radiological consequences of a fuel cask drop was based on Pasquill Type D meteorological conditions with a wind velocity of 2 m/s and no allowance for filtration before release. The potential doses thus calculated (Table 15.1-1) are well below the guidelines of 10 CFR 100. The Pasquill D and 2 m/s conditions are common daytime meteorological conditions at the ANO-1 site. We will impose a Technical Specification requirement that a loaded fuel cask not be carried above or into the shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/s. In addition, the Technical Specifications will require that the outside door of the Turbine Building be closed and that the filtered ventilation system in the fuel handling area be in operation. We have concluded that a separate pneumatic seal at the bottom of the shaft is not necessary with these specific limits on meteorological conditions.

Consideration has also been given to assuring the capability to shut the plant down and cool down safely following the rupture of a high energy line outside containment. This is discussed in Section 6.4 of this report.

On the basis of our experience with the evaluations of the steam line 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 and secondary coolant system radioactivity concentrations so that potential offsite doses are small. We will include limits in the Technical Specifications on these coolant 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.

The applicant has evaluated the loss-of-coolant accident, the fuel-handling accident and the radioactive gas decay tank rupture accident using assumptions that are substantially the same as those used by the Regulatory staff. In calculating the loss-of-coolant accident doses, consideration was given to the processing of leakage into the penetration room by an iodine absorber system prior to release to the environment. Assurance will be obtained, by Technical Specification, that any leakage greater than one-half the design leak rate of the reactor building will be shown to be processed by the penetration room filtering system. The effective iodine removal efficiency was 88% for the fraction of the reactor building leakage assumed to be filtered.

The offsite doses that we calculate for these accidents are presented in Table 15-1 of this report. Our assumptions are listed in Tables 15-2, 15-3 and 15-4 of this report. All of these doses are well within the guideline doses given in 10 CFR Part 100 and are considered acceptable.

TABLE 15-1

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	TWO HOUR DOSE AT EXCLUSION BOUNDARY (1046 Meters)		COURSE OF ACCIDENT DOSE AT LOW POPULATION ZONE (6440 Meters)	
ACCIDENT	THYROID (Rem)	WHOLE BODY (Rem)	THYROID (Rem)	WHOLE BODY (Rem)
Loss-of-Cooling	158	13	62	5
Fuel Handling	23	3	5	<1
Fuel Cask Drop	3	<1	<1	<1

Note: These potential offsite doses are the notable ones for the ANO-1 site. Doses calculated for other design basis accidents are well below these values.

TABLE 15-2

LOSS-OF-COOLANT ACCIDENT ASSUMPTIONS

Regulatory Guide 1.4	
Volume of the Reactor Building	1.865×10^6 cubic feet
Core Power Level	2568 MWt
Number of Fuel Rods in Core	36,816
Operating Time	3 years
Fraction of Noble Gases Released	100%
Fraction of Halogens Airborne	25%
Halogen Composition	85% elemental 10% organic 5% particulate
Reactor Building Leak Rate	0.2%/day 0-24 hours 0.1%/day after 24 hours
Exclusion Radius	1046 meters
Low Population Zone	6440 meters
Atmospheric Dilution Factors	(sec/m^3)
0-2 hours at 1046 meters	6.8×10^{-4}
0-8 hours at 6440 meters	1.1×10^{-4}
8-24 hours at 6440 meters	1.1×10^{-5}
24-96 hours at 6440 meters	4.0×10^{-6}
96-720 hours at 6440 meters	1.3×10^{-6}

TABLE 15-3

REFUELING ACCIDENT ASSUMPTIONS

1. Long term operation at 2568 MWt core power level.
2. Fuel transfer 72 hours after shutdown.
3. A total of 208 rods (one assembly) are damaged.
4. This assembly has operated at 1.8 times the average power density.
5. The rods release 10% of their noble gas inventory and 10% of the iodines to the water.
6. The initial composition of iodine is taken as 99.75% elemental and 0.25% organic.
7. The effective overall reduction for iodines is a factor of 100.
8. The fraction of elemental iodine which is removed by the charcoal filter is 90% and for organics, 70%, giving an overall efficiency of 85%.
9. The release is complete within 2 hours.
10. Meteorological assumptions are the same as for the LOCA.

TABLE 15-4

FUEL CASK DROP ASSUMPTIONS

3 year operating time

18 fuel assemblies are in the dropped cask

100 days decay prior to shipping

10% halogens released

10% noble gases released

meteorology - Pasquill "D" and 2 meters/sec
wind speed at the 1046 meter
exclusion radius, under
controlled conditions

16.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the AEC. We reviewed the proposed Technical Specifications and held a number of meetings with the applicant to discuss their contents. Modifications to the proposed Technical Specifications submitted by the applicant were made to describe more clearly the allowed conditions for plant operation. The finally approved Technical Specifications will be made part of the operating license. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls. On the basis of our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of the 10 CFR Part 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

17.0 QUALITY ASSURANCE

The Quality Assurance (QA) Program proposed for operating, maintaining, repairing, testing, refueling and modifying ANO-1 is described in Section 1.6 of the FSAR, supplemented by Amendments 25, 27, and 28. Our evaluation of AP&L QA Program is based on an overall detailed review of this information with subsequent discussions with the applicant to determine AP&L's ability to comply with the requirements of 10 CFR 50, Appendix B and to assure safe operation of the facility.

17.1 Organization

Responsibility and authority to define and direct the Quality Assurance Program is assigned to the Vice President and Chief Engineer, who reports directly to the President of AP&L. The Chief Quality Assurance Coordinator, under the direction of the Quality Assurance Committee, assists the Vice President and Chief Engineer in defining and implementing the Quality Assurance Program and in auditing and assessing plant operation activities to assure safe operation and compliance to program requirements. A Quality Control Engineer, reporting to the Assistant Plant Superintendent and permanently assigned to the Unit 1 station, implements the Operational Quality Control Program on a day to day basis.

Our initial review indicated a lack of organizational independence of the Quality Control Engineer from personnel directly responsible for maintenance, modification and operation of plant facilities. We discussed this deficiency at a meeting held with AP&L's management. AP&L stated that the Quality Control Engineer would be more organizationally independent by reporting to the Assistant Plant Superintendent and that he would have direct communication to the Chief Quality Assurance Coordinator. We consider this organizational move acceptable in strengthening the independence of the onsite QA staff.

Significant areas of responsibilities of the Quality Control Engineer are:

(1) Developing and implementing operational quality control procedures.

(2) Monitoring calibration and control of measuring and testing devices to assure calibration is performed in accordance with approved procedures and that adequate labeling of instruments is provided indicating status of calibration.

(3) Participating in the maintenance and modification of safety related equipment to assure the applicable regulations, standards, codes, and quality requirements are complied with. This includes activity associated with the procurement, repair and inspection of safety related components, systems and structures.

4. Assuring that nonconformances are adequately described and dispositioned on a nonconformance report and that cause and corrective action is determined when applicable.
5. Controlling the record filing system to assure proper filing and maintenance of quality control documents.
6. Conducting planned and periodic audits, providing an independent check and assessment of all significant plant operation activities including maintenance, modification, fuel handling and storage.

Based on our review, we have concluded that the organization as presented in the FSAR and amendments and the organizational change described above satisfy the requirements of 10 CFR 50, Appendix B and are acceptable.

17.2 Audits

The Chief Quality Assurance Coordinator is responsible for conducting systematic and detailed audits on all activities related to the Operational Quality Control Program and procedures. This includes the review of procedures to assure they are meaningful and provide the required codes, standards and criteria. Audit results including corrective action of deficient areas are formally documented and submitted to upper management and responsible departments. The Chief Quality Control Coordinator is responsible to assure that corrective action is properly implemented.

The Quality Control Engineering audit activity has not been completely defined as yet due to the early stages of plant operating procedures. However, Quality Control Engineering will place particular emphasis in performing audits on all significant plant operations to assure they are in accordance with the Quality Control Plan and applicable procedures. Quality Control Engineering has access to upper management to assure proper recognition is given to audit results and corrective action.

We conclude that AP&L recognizes the importance of thorough and independent audits and that their audit program is acceptable.

17.3 Fuel

As part of our QA review, we have evaluated AP&L plans for review of fuel design and manufacture to assure its long term integrity. AP&L has and will utilize Middle South Services, Inc. (MSS) as an independent fuel QA consultant for fuel design and manufacture. Together with MSS, AP&L has and will continue to conduct design reviews, design and manufacturing audits and detailed physical examinations of fuel upon receipt at ANO-1.

The applicant has described the design and manufacturing features of the ANO-1 fuel which are intended to minimize possible fuel failures resulting from clad hydriding, UO_2 -clad interaction, or clad collapse. These include restriction of possible moisture and hydrocarbon contaminants in the UO_2 pellets, chamfered and dished fuel pellets, and prepressurized fuel rods with top and bottom void regions to allow for bidirectional expansion. These actions represent current state of the art actions that should minimize fuel failures during plant operation. Although we consider such actions appropriate, it may be necessary to impose further requirements with regard to plant operation pending completion of our current review of fuel densification.

We have concluded that AP&L's QA program should reduce the probability of fuel failures during ANO-1 operation.

17.4 Conclusion

Based on our review of the QA Program defined by AP&L, we have concluded that this program complies with the requirements of 10 CFR Part 50, Appendix B, industry standards ANSI N45.2 and draft ANS 3.2, and is acceptable for use during operation of ANO-1.

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

The report of the ACRS on the operation review of Arkansas Nuclear One - Unit 1 will be placed in the Commission's Public Document Room and will be published in a supplement, by the Regulatory Staff, to this Safety Evaluation. The staff will also discuss further its evaluation of several items still considered outstanding. The supplement will be published prior to the final determination regarding issuance of an operating license.

19.0 COMMON DEFENSE AND SECURITY

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

20.0 FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to financial data and information required to establish the financial qualifications of an applicant for a facility operating license are 10 CFR 50 Part 33(f) and 10 CFR 50, Appendix C. We have reviewed the financial information presented in the application and have concluded that the applicant is financially qualified to operate ANO-1. We have also examined the Annual Report for AP&L for 1972; our examination does not cause us to change our judgement of the applicant's financial qualifications. A detailed discussion of the basis for our conclusion is presented in Appendix D.

21.0 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors licensed under 10 CFR Part 50.

21.1 Preoperational Storage of Nuclear Fuel

The Commission's regulations in Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also to be the holder of a license under 10 CFR Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

AP&L, with respect to the ANO-1, is subject to the foregoing requirements, and took the following steps, as required.

AP&L furnished to the Commission proof of financial protection in the amount of $1,000,000, in the form of a nuclear energy liability insurance policy.

Further, AP&L executed an Indemnity Agreement with the Commission as of the effective date of its pertinent preoperational fuel storage license (November 8, 1972). AP&L paid the annual indemnity fee applicable to preoperational fuel storage.

21.2 Operating License

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been executed. The amount of financial protection which must be maintained for reactors which have a rated capacity of 100,000 electrical kilowatts or more is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which is $95 million.

Accordingly, no license authorizing operation of the ANO-1 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement or amendment executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended and that the policy limits have been increased to an amount that meets the requirements of the Commission's regulations for reactor operation. Similarly, no operating license will be issued until an appropriate amendment to the present indemnity agreement has been issued. AP&L will be required to pay an annual fee for operating license indemnity as provided in AEC regulations.

21.3 Conclusion

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140, concerning preoperational storage of fuel, are being satisfied and that, prior to issuance of any operating license, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and to execution of an appropriate indemnity agreement or amendment thereto with the Commission.

22.0 CONCLUSIONS

Based on our evaluation of the application as set forth above, we have concluded that:

1. The application for a facility license filed by the Arkansas Power & Light Company dated April 19, 1971, as amended (Amendments Nos. 1 through 37) complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter 1; and
2. Construction of Arkansas Nuclear One - Unit 1 (the facility) has proceeded and there is reasonable assurance that it will be substantially completed, in conformity with Provisional Construction Permit No. CPPR-57, the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
3. The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission; and
4. There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1; and

5. The applicant is technically and financially qualified to engage in the activities authorized by this license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Before an operating license will be issued to the Arkansas Power & Light Company for operation of Arkansas Nuclear One - Unit 1, the unit must be completed in conformity with the provisional construction permit, the application, the Act, and the rules and regulations of the Commission. Such completeness of construction as is required for safe operation at the authorized power level must be verified by the Commission's Directorate of Regulatory Operations prior to license issuance. In addition, satisfactory resolution of outstanding matters such as fuel densification and the consequences of high energy line rupture will be required.

Further, before an operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR Part 140.

APPENDIX A

CHRONOLOGY OF THE REGULATORY STAFF'S OPERATING LICENSE REVIEW OF ARKANSAS NUCLEAR ONE, UNIT 1

April 19, 1971	Submittal of Amendment No. 20 - Application for Operating License
June 3, 1971	Letter from Arkansas Pollution Control Comm. transmitting water quality certificate for Arkansas Nuclear One, Units 1 and 2
June 14, 1971	Submittal of Environmental Report
July 28, 1971	Initial meeting with applicant for OL review
August 11, 1971	Request for additional information
September 27, 1971	Submittal of Amendment No. 21
November 1, 1971	Request for additional information
December 13, 1971	Request for additional information
December 14, 1971	Submittal of Amendment No. 22
December 16, 1971	Meeting with applicant to discuss certain areas of the OL application that required additional information
January 21, 1972	Submittal of Amendment No. 23
February 29, 1972	Submittal of Amendment No. 24
March 7, 1972	Letter from APLC advising that APLC intends to adopt provisions of AEC Inservice Inspection Program
March 10, 1972	Letter from APLC transmitting 1971 annual financial report
March 24, 1972	Meeting with applicant concerning review of operating license application

March 24, 1972	Meeting with applicant to discuss meteorology
March 31, 1972	Submittal of Amendment No. 25
April 6, 1972	Letter to applicant to review schedule
April 11, 1972	Letter to applicant transmitting Draft Criteria on Industrial Security
April 21, 1972	Submittal of Amendment No. 26
April 24, 1972	Letter to applicant re public document room
April 25/26, 1972	Site visit for arrangement review
April 28, 1972	Report on Site Visit (April 25-26)
May 8, 1972	ECCS Evaluation
May 23, 1972	Meeting with applicant to discuss review problems
June 1, 1972	Letter to applicant on B&W topical reports
June 1, 1972	Letter to applicant requesting extension of CPPR-57
June 14, 1972	Order extending Construction Permit
June 20, 1972	Letter from applicant furnishing up-to-date listing of B&W topical reports
June 22, 1972	Internal memo requesting added info for FSAR
June 30, 1972	AEC letter to applicant requesting added info for FSAR
July 3, 1972	Internal memo re technical assistance request ECCS Evaluation
July 13, 1972	AEC letter to applicant requesting added info for FSAR
July 14, 1972	Meeting with applicant to discuss radioactive releases and radwaste systems
July 17, 1972	AEC memo to applicant transmitting copy of our letter to B&W requesting added info

August 4, 1972	Submittal of Amendment No. 27
August 18, 1972	Submittal of Amendment No. 28
August 23, 1972	AEC letter to applicant asking for latest financial data
August 23, 1972	AEC letter to applicant on containment leak test
August 24, 1972	AEC letter to applicant on reactor coolant leak detection apparatus
August 31, 1972	Meeting with applicant on Technical Specifications
September 5, 1972	AEC letter requesting additional info on containment
September 8, 1972	Submittal of Amendment No. 29
September 8, 1972	Submittal of Industrial Security Plan and draft Emergency Plan
September 14, 1972	Visit to site for Quality Assurance/Control review
September 15, 1972	Submittal of Amendment No. 30
September 26, 1972	AEC letter to applicant regarding effects of failure of non-Category I systems
October 12, 1972	AEC letter to applicant noting acceptance of B&W Topical Report BAW-10047, Rev. 1
October 12, 1972	AEC issued public notice of consideration of license
October 18, 1972	AEC letter to applicant noting acceptance of B&W Topical Report BAW-10013
October 20, 1972	Applicant response on effects of failure of non-Category I systems
October 25, 1972	Letter from applicant reviewing status of reference B&W Topical reports
October 31, 1972	AEC letter to applicant requesting further information on pressure and hydrogen in containment

November 1, 1972	AEC letter to applicant on DHRS valve interlocks
November 2, 1972	AEC letter to applicant on outstanding mechanical design questions
November 2, 1972	Meeting with applicant on electrical drawing review
November 13, 1972	AEC letter to applicant on tendon surveillance
November 15, 1972	AEC letter to applicant on steam line break concern
November 15, 1972	Submittal of Amendment No. 31
November 20, 1972	AEC letter to applicant requesting fuel densification analysis
November 20, 1972	Meeting with applicant on electrical drawing review
November 29, 1972	Site visit for electrical review
November 30, 1972	Submittal of Amendment No. 32
December 14, 1972	AEC letter to applicant requesting analysis of line breaks outside containment
December 21, 1972	Submittal of Amendment No. 33
December 27, 1972	AEC letter to applicant requesting information on active valve testing
January 2, 1973	Site visit for radwaste review
January 23, 1973	Meeting with applicant on electrical review and active valve testing
January 24, 1973	AEC letter to applicant noting acceptance of B&W Topical Report BAW-10029
February 7, 1973	AEC letter to applicant noting requirements deriving from electrical review
February 9, 1973	Submittal of Amendment No. 34

February 28, 1973	Submittal of Amendment No. 35
March 2, 1973	Meeting with applicant on high energy line rupture outside containment
March 5, 1973	AEC letter to applicant requesting piping stress summary
March 7, 1973	AEC letter to applicant on generic control circuit question
March 13, 1973	Applicant responded to AEC letter on requirements deriving from electrical review
March 14, 1973	Applicant requested extension of Construction Permit period
March 23, 1973	Applicant letter providing piping stress summary
April 6, 1973	Submittal of Amendment No. 36
April 11, 1973	Applicant responded to AEC letter on generic control circuit question
April 13, 1973	Applicant submitted interim report on fuel densification analysis
April 20, 1973	AEC letter to applicant on outstanding instrument and control matters
April 23, 1973	AEC letter to applicant on dike design for emergency cooling pond
April 23, 1973	Applicant letter on steam line break committing to modification before exceeding 1% power
April 27, 1973	Submittal of Amendment No. 37
May 3, 1973	Site visit to review steam line break
May 4, 1973	ACRS Subcommittee tour of site
May 11, 1973	Applicant response on pond dike design
May 11, 1973	Applicant response on electrical and control items

APPENDIX B

NOAA REPORT

ON SITE METEOROLOGY

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

January 11, 1973

50-313

R3.3

RECEIVED
JAN 17 1973
U.S. ATOMIC ENERGY COMMISSION
Regulatory
Mail Section

Dr. Joseph M. Hendrie
Deputy Director for Technical Review
Directorate of Licensing, U.S.A.E.C.
Washington, D. C. 20545

Dear Dr. Hendrie:

This refers to the letter of September 14, 1972, from A. Schwencer, Chief, Pressurized Water Reactors Branch No. 4, Directorate of Licensing, requesting comments on the following:

Arkansas Nuclear One, Unit 1
Arkansas Power and Light Company
Final Safety Analysis Report
Amendment No. 29 dated 9/8/72

These comments are attached.

Sincerely,

Isaac Van der Hoven

Isaac Van der Hoven, Chief
Air Resources Environmental Lab.
Air Resources Laboratories

Attachment

cc: E.H. Markee, USAEC

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

Comments on

Arkansas Nuclear One, Unit 1
Arkansas Power and Light Company
Final Safety Analysis Report
Amendment No. 29 dated 9/8/72

Prepared by

Air Resources Environmental Laboratory
National Oceanic and Atmospheric Administration
January 11, 1973

The basis for our most recent evaluation of the diffusion characteristics of the site is the data presented in tables 2.A-30 and 31 of Amendment No. 29. These data cover a one-year period, approximately half of which winds were measured at 190 ft. and the remainder at 40 ft. Because of effluent emission from rooftop vents, we have assumed a ground release and therefore have reduced the 190 ft. wind speeds to an equivalent 40-ft. level by means of the power law function suggested in the ASME Guide.

For the short-term (0-2 hours) release we have estimated from the joint frequency of wind speed, direction and temperature gradient in the vertical that a relative concentration of 7×10^{-4} sec m^{-3} will be exceeded five percent of the time at the minimum exclusion distance of 1046 m. This assumes a ground source and a building wake factor of $cA = 1100\ m^2$. The concentration value is in close agreement with the value shown by the applicant in figure 2A-16.

We have not estimated concentrations for periods from 2 hours to 30 days since the meteorological data are not presented for these periods.

For the average annual relative concentration we have estimated that the maximum value occurs with winds from the east. From the data compilation shown in tables 2.A-30 and 31 we estimate a 24 percent frequency of winds from the east divided among Pasquill types F, D and B at a frequency of 12, 8 and 4 percent and speeds of 2, 3 and 3 m/sec, respectively. The resulting average annual relative concentration at 1046 m is 1×10^{-5} sec m^{-3}.

APPENDIX C

General Information Required for Consideration of the Effects of a Piping System Break Outside Containment

The following is a general list of information required for AEC review of the effects of a piping system break outside containment, including the double ended rupture of the largest pipe in the main steam and feed-water systems, and for AEC review of any proposed design changes that may be found necessary. Since piping layouts are substantially different from plant to plant, applicants and licensees should determine on an individual plant basis the applicability of each of the following items for inclu[illegible] their submittals.

1. The systems (or portions of systems) for which protection against pipe whip is required should be identified. Protection from pipe whip need not be provided if any of the following conditions will exist:

   (a) Both of the following piping system conditions are met:

      (1) the service temperature is less than 200°F; and

      (2) the design pressure is 275 psig or less; or

   (b) The piping is physically separated (or isolated) from structures, systems, or components important to safety by protective barriers, or restrained from whipping by plant design features, such as concrete encasement; or

   (c) Following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any structure, system, or component important to safety; or

(d) The internal energy level[1] associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

2. "Design basis break locations should be selected in accordance with the following pipe whip protection criteria; however, where pipes carrying high energy fluid are routed in the vicinity of structures and systems necessary for safe shutdown of the nuclear plant, supplemental protection of those structures and systems shall be provided to cope with the environmental effects (including the effects of jet impingement) of a single postulated open crack at the most adverse location(s) with regard to those essential structures and systems, the length of the crack being chosen not to exceed the critical crack size. The critical crack size is taken to be 1/2 the pipe diameter in length and 1/2 the wall thickness in width."

The criteria used to determine the design basis piping break locations in the piping systems should be equivalent to the following:

(a) ASME Section III Code Class I piping[2] breaks should be postulated to occur at the following locations in each piping run[3] or branch run:

(1) the terminal ends;

(2) any intermediate locations between terminal ends where the primary plus secondary stress intensities S_m (circumferential

or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4] exceeds 2.0 S_m[5] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) any intermediate locations between terminal ends where the cumulative usage factor (U)[6] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) at intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) the terminal ends;

(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 $(S_h + S_A)$[7] or the expansion stresses exceed 0.8 S_A; and

(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions[4] exceeds 2.0 S_m[5] for ferritic steel, and 2.4 S_m for austenitic steel;

(3) any intermediate locations between terminal ends where the cumulative usage factor (U)[6] derived from the piping fatigue analysis and based on all normal, upset, and testing plant conditions exceeds 0.1; and

(4) at intermediate locations in addition to those determined by (1) and (2) above, selected on a reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

(b) ASME Section III Code Class 2 and 3 piping breaks should be postulated to occur at the following locations in each piping run or branch run:

(1) the terminal ends;

(2) any intermediate locations between terminal ends where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 (S_h + S_A)[7] or the expansion stresses exceed 0.8 S_A; and

(3) intermediate locations in addition to these determined by (2) above, selected on reasonable basis as necessary to provide protection. As a minimum, there should be two intermediate locations for each piping run or branch run.

3. The criteria used to determine the pipe break orientation at the break locations as specified under 2 above should be equivalent to the following:

   (a) Longitudinal[8] breaks in piping runs and branch runs, 4 inches nominal pipe size and larger, and/or

   (b) Circumferential[9] breaks in piping runs and branch runs exceeding 1 inch nominal pipe size.

4. A summary should be provided of the dynamic analyses applicable to the design of Category I piping and associated supports which determine the resulting loadings as a result of a postulated pipe break including:

   (a) The locations and number of design basis breaks on which the dynamic analyses are based.

   (b) The postulated rupture orientation, such as a circumferential and/or longitudinal break(s), for each postulated design basis break location.

   (c) A description of the forcing functions used for the pipe whip dynamic analyses including the direction, rise time, magnitude, duration, and initial conditions that adequately represent the jet stream dynamics and the system pressure difference.

   (d) Diagrams of mathematical models used for the dynamic analysis.

   (e) A summary of the analyses which demonstrates that unrestrained motion of ruptured lines will not damage to an unacceptable degree, structures, systems, or components important to safety, such as the control room.

5. A description should be provided of the measures, as applicable, to protect against pipe whip, blowdown jet and reactive forces including:

   (a) Pipe restraint design to prevent pipe whip impact;

   (b) Protective provisions for structures, systems, and components required for safety against pipe whip and blowdown jet and reactive forces;

   (c) Separation of redundant features;

   (d) Provisions to separate physically piping and other components of redundant features; and

   (e) A descriptive of the typical pipe whip restraints and a summary of number and location of all restraints in each system.

6. The procedures that will be used to evaluate the structural adequacy of Category I structures and to design new seismic Category I structures should be provided including:

   (a) The method of evaluating stresses, e.g., the working stress method and/or the ultimate strength method that will be used;

   (b) The allowable design stresses and/or strains; and

   (c) The load factors and the load combinations.

7. The structural design loads, including the pressure and temperature transients, the dead, live and equipment loads; and the pipe and equipment static, thermal, and dynamic reactions should be provided.

8. Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations and the buildings as a whole

should be analyzed for eventual reversal of loads due to the postulated accident.

9. If new openings are to be provided in existing structures, the capabilities of the modified structures to carry the design loads should be demonstrated.

10. Verification that failure of any structure, including nonseismic Category I structures, caused by the accident, will not cause failure of any other structure in a manner to adversely affect:

    (a) Mitigation of the consequences of the accidents; and

    (b) Capability to bring the unit(s) to a cold shutdown condition.

11. Verification that rupture of a pipe carrying high energy fluid will not directly or indirectly result in:

    (a) Loss of required redundancy in any portion of the protection system (as defined in IEEE-279), Class IE electric system (as defined in IEEE-308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of that accident and place the reactor(s) in a cold shutdown condition; or

    (b) "Environmentally induced failures caused by a leak or rupture of the pipe which would not of itself result in protective action but does disable protection functions. In this regard, a loss of redundancy is permitted but a loss of function is not permitted. For such situations plant shutdown is required."

12. Assurance should be provided that the control room will be habitable and its equipment functional after a steam line or feedwater line break or that the capability for shutdown and cooldown of the unit(s) will be available in another habitable area.

13. Environmental qualification should be demonstrated by test for that electrical equipment required to function in the steam-air environment resulting from a high energy fluid line break. The information required for our review should include the following:

    (a) Identification of all electrical equipment necessary to meet requirements of 11 above. The time after the accident in which they are required to operate should be given.

    (b) The test conditions and the results of test data showing that the systems will perform their intended function in the environment resulting from the postulated accident and time interval of the accident. Environmental conditions used for the tests should be selected from a conservative evaluation of accident conditions.

    (c) The results of a study of steam systems identifying locations where barriers will be required to prevent steam jet impingement from disabling a protection system. The design criteria for the barriers should be stated and the capability of the equipment to survive within the protected environment should be described

    (d) An evaluation of the capability for safety related electrical equipment in the control room to function in the environment that may exist following a pipe break accident should be provided.

Environmental conditions used for the evaluation should be selected from conservative calculations of accident conditions.

(e) An evaluation to assure that the onsite power distribution system and onsite sources (diesels and batteries) will remain operable throughout the event.

14. Design diagrams and drawings of the steam and feedwater lines including branch lines showing the routing from containment to the turbine building should be provided. The drawings should show elevations and include the location relative to the piping runs of safety related equipment including ventilation equipment, intakes, and ducts.
15. A discussion should be provided of the potential for flooding of safety related equipment in the event of failure of a feedwater line or any other line carrying high energy fluid.
16. A description should be provided of the quality control and inspection programs that will be required or have been utilized for piping systems outside containment.
17. If leak detection equipment is to be used in the proposed modifications, a discussion of its capabilities should be provided.
18. A summary should be provided of the emergency procedures that would be followed after a pipe break accident, including the automatic and manual operations required to place the reactor unit(s) in a cold shutdown condition. The estimated times following the accident for all equipment and personnel operational actions should be included in the procedure summary.

19. A description should be provided of the seismic and quality classification of the high energy fluid piping systems including the steam and feedwater piping that run near structures, systems, or components important to safety.

20. A description should be provided of the assumptions, methods, and results of analyses, including steam gener[illegible]r blowdown, used to calculate the pressure and temperature transients in compartments, pipe tunnels, intermediate buildings, and the turbine building following a pipe rupture in these areas. The equipment assumed to function in the analyses should be identified and the capability of systems required to function to meet a single active component failure should be described.

21. A description should be provided of the methods or analyses performed to demonstrate that there will be no adverse effects on the primary and/or secondary containment structures due to a pipe rupture outside these structures.

Footnotes

[1] The internal fluid energy level associated with the pipe break reaction may take into account any line restrictions (e.g., flow limiter) between the pressure source and break location, and the effects of either single-ended or double-ended flow conditions, as applicable. The energy level in a whipping pipe may be considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

[2] Piping is a pressure retaining component consisting of straight or curved pipe and pipe fittings (e.g., elbows, tees, and reducers).

[3] A piping run interconnects components such as pressure vessels, pumps, and rigidly fixed valves that may act to restrain pipe movement beyond that required for design thermal displacement. A branch run differs from a piping run only in that it originates at a piping intersection, as a branch of the main pipe run.

[4] Operational plant conditions include normal reactor operation, upset conditions (e.g., anticipated operational occurrences) and testing conditions.

[5] S_m is the design stress intensity as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Plant Components."

[6] U is the cumulative usage factor as specified in Section III of the ASME Boiler and Pressure Vessel Code, "Nuclear Power Plant Components."

[7] S_h is the stress calculated by the rules of NC-3600 and ND-3600 for Class 2 and 3 components, respectively, of the ASME Code Section III Winter 1972 Addenda.

S_A is the allowable stress range for expansion stress calculated by the rules of NC-3600 of the ASME Code, Section III, or the USA Standard Code for Pressure Piping, ANSI B31.1.0-1967.

[8] Longitudinal breaks are parallel to the pipe axis and oriented at any point around the pipe circumference. The break area is equal to the effective cross-sectional flow area upstream of the break location. Dynamic forces resulting from such breaks are assumed to cause lateral pipe movements in the direction normal to the pipe axis.

[9] Circumferential breaks are perpendicular to the pipe axis, and the break area is equivalent to the internal cross-sectional area of the ruptured pipe. Dynamic forces resulting from such breaks are assumed to separate the piping axially, and cause shipping in any direction normal to the pipe axis.

APPENDIX D

FINANCIAL QUALIFICATIONS

The Commission's regulations which relate to the fincancial data and information required to establish financial qualifications for an applicant for an operating license are 10 CFR 50.33(f) and 10 CFR 50, Appendix C. The basic application of Arkansas Power and Light Company, Amendment Nos. 3, 19, 20, and 29, and the accompanying certified annual financial statements of the applicant provides the financial information required by the Commission's regulations. This information includes the estimated annual costs of operating "Arkansas Nuclear One - Unit 1" for the first five years of operation plus the estimated cost of permanently shutting down the facility and maintaining it in a safe shutdown condition.

Our evaluation of the financial data submitted by the applicant, summarized below, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) to operate Arkansas Nuclear One - Unit 1, and if necessary permanently shut down the facility and maintain it in a safe shutdown condition.

Arkansas Nuclear One, Unit 1, will be used as an integral part of the applicant's total operating system. Operation of Unit 1 will be financed substantially from internally generated funds, principally retained earnings and provision for depreciation. The remainder of the required funds will be obtained from sale of debt and/or equity securities, and from short-term loans needed to meet requirements on a temporary basis. Operating costs for the first five years are presently estimated by the applicants to be (in millions of dollars) \$9.2; \$40.7; \$40.6; \$40.8; and \$40.9 in that order. These costs include amounts for operation, maintenance, fuel cost, insurance, overhead, depreciation, interest on investment, and taxes. In addition, the applicants estimate the cost of permanently shutting down the facility (based on 1972 cost levels) will be of an order of magnitude of \$10 million, and that an annual cost of \$40,000 will be incurred to maintain the facility in a safe shutdown condition. Operating revenues and retained earnings will provide the funds to cover cost of shutdown and surveillance.

Arkansas Power and Light Company is adequately financed and has significant resources at its command. As of December 31, 1971, cash and net receivables totaled \$15.1 million. Long-term debt represented 57.8% of total capitalization and 50.9% of the net investment in utility plant. The applicant's Dun and Bradstreet credit rating is 5A1 (the highest category) and Moody's Investors Service rates the company's first mortgage bonds as A (higher medium grade).

Operating revenue of \$166.1 million for 1971 was up 69% over 1966, and net income, after taxes, of \$28.9 million was up 84% over 1966. The volume of electric energy sales over the same five years has increased 77% to 13,843 million kilowatt hours in 1971. The number of times interest earned on long-term debt has decreased from 3.2 in 1966 to 2.7 in 1971. The pertinent financial ratios indicate an adequate financial position; these are in line with ratios of the electric utility industry as a whole. A summary analysis reflecting the ratios and other pertinent data for the applicant is attached as as Table D-1.

In brief, these ratios as of December 31, 1971 are: long-term debt to net utility plant - .51; net plant to capitalization - 1.13; proprietary ratio - .36; operating ratio - .76; rate of earnings before interest on total investment - 6.7%; rate of earnings on stockholders' equity - 11.3%; times interest earned on long-term debt - 2.66; and retained earnings - $31.0 million.

Arkansas Power and Light Company is an operating subsidiary of Middle South Utilities, Inc. Middle South Utilities' certified consolidated financial statements for calendar year 1971 include its six principal subsidiaries - Arkansas, Louisiana, and Mississippi Power & Light Companies, New Orleans Public Service Inc. Our examination of Middle South Utilities, Inc., is adequately financed and has significant resources at its command. As of December 31, 1971, cash and net receivables totaled $78.9 million. Long-term debt represented 58.4% of total capitalization and 52.2% of the net investment in utility plant. The company's Dun and Bradstreet credit rating is 5A1 (the highest category).

Consolidated operating revenue of $506.3 million for 1971 was up 65% over 1966, and consolidated net income, after taxes and preferred

TABLE D-1 ARKANSAS POWER AND LIGHT COMPANY

FINANCIAL ANALYSIS

	(dollars in millions) Calendar Year Ended December 31		
	1969	1970	1971
Long-term debt	$ 263.8	$ 288.6	$ 348.4
Utility plant (net)	512.4	562.7	684.9
Ratio - debt to fixed plant	.51	.51	.51
Utility plant (net)	512.4	562.7	684.9
Capitalization	463.3	503.5	603.3
Ratio of net plant to capitalization	1.10	1.12	1.13
Stockholders' equity	199.5	214.9	254.9
Total assets	537.5	586.8	709.9
Proprietary ratio	.37	.37	.36
Earnings available to common equity	19.1	21.9	26.1
Common equity	148.0	163.4	203.4
Rate of earnings on common equity	12.9%	13.4%	12.8%
Net income	21.8	24.7	28.9
Stockholders' equity	199.5	214.9	254.9
Rate of earnings on stockholders' equity	10.9%	11.5%	11.3%
Net income before interest	33.5	39.0	47.3
Liabilities and capital	537.5	586.8	709.9
Rate of earnings on total investment	6.2%	6.6%	6.7%
Net income before interest	33.5	39.0	47.3
Interest on long-term debt	10.5	13.6	17.8
No. of times long-term interest earned	3.19	2.87	2.66
Net income	21.8	24.7	28.9
Total revenues	138.7	152.6	173.6
Net income ratio	.16	.16	.16
Total utility operating expenses	105.2	113.6	126.3
Total utility operating revenues	136.0	149.3	166.1
Operating ratio	.77	.76	.76
Utility plant (gross)	644.1	708.4	845.6
Utility operating revenues	136.0	149.3	166.1
Ratio of plant investment to revenues	4.74	4.74	5.09

TABLE D-1 (Continued)

Capitalization:	1970 Amount	1970 % of Total	1971 Amount	1971 % of Total
	(dollars in millions) Calendar Year Ended December 31			
Long-term debt	$288.6	57.3%	$348.4	57.8%
Preferred stock	51.5	10.2	51.5	8.5
Common stock & surplus	163.4	32.5	203.4	33.7
Total	$503.5	100.0%	$603.3	100.0%

Moody's Bond Rating:	First Mortgage	A
	Sinking Fund Debentures	Baa
Dun & Bradstreet Credit Rating		5A1