December 29, 1970

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

DOCKET NO. 50-269

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1.0 INTRODUCTION

The Duke Power Company (applicant) by application dated November 28, 1966 and as subsequently amended, requested a license to construct and operate three pressurized water reactors, identified as Units 1, 2, and 3 at its Oconee Nuclear Station in Oconee County, South Carolina. The Atomic Energy Commission reported the results of its review prior to construction in a Safety Evaluation dated August 4, 1967. Following a public hearing before an Atomic Safety and Licensing Board in Walhalla, South Carolina on August 29-30, 1967 and September 12, 1967, the Director of Reactor Licensing issued Provisional Construction Permits CPPR-33, -34, and -35 for Units 1, 2, and 3, respectively, on November 6, 1967.

On June 2, 1969 the applicant filed, as Amendment 7, the Final Safety Analysis Report required by Section 50.34(b) of Chapter 10 of the Code of Federal Regulations as a prerequisite to obtaining an operating license for each unit.

Although the regulatory staff's review of the Final Safety Analysis Report, as amended, considered all three units of the Oconee Nuclear Station, Unit 1 is the only unit whose state of completion warrants issuance of an operating license at this time.

In addition, the Commission's Advisory Committee on Reactor Safeguards has also considered this project and has met with both the applicant and the regulatory staff to discuss it. The report of the Advisory Committee on Reactor Safeguards, dated September 23, 1970, is included as Appendix B. Reports by our consultants on meteorology, hydrology, ecological (Fish and Wildlife) considerations, and seismic design are included as Appendices C through F. A staff financial analysis is set forth in Appendix G.

The application for an operating license for Unit 1 requests a licensed core thermal power level of 2568 megawatts as compared to the 2452 megawatts thermal (MWt) requested prior to start of construction. Our evaluation of the engineered safety features and our accident analyses have been performed for the higher level; further, our analyses included the 16 megawatts of reactor coolant pump heat in the reactor coolant system in addition to the 2568 megawatts from the reactor core. The proposed license would initially authorize operation at core power levels up to and including 2452 MWt and would permit operation at core power levels up to 2568 MWt if and when authorized by the regulatory staff after review of the performance of the reactor to assure that the core can be operated safely at such levels.

Our technical safety review with respect to issuing an operating license for Unit 1 has been based on the applicant's Final Safety Analysis Report (Amendment 7) and subsequent Amendments 8 through 24 inclusive (Amendments 1 through 6 were related to the construction permit review), all of which are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D. C. In the course of our review of the material submitted, we held a number of meetings with representatives of the applicant; the nuclear steam supplier, the Babcock & Wilcox Company; and the designer of the reactor containment building, the Bechtel Corporation; to discuss the plant design and construction and the proposed operation. A chronology of our review is presented in Appendix A of this evaluation.

Based on our evaluation of the plant, summarized in subsequent sections of this report, we have concluded that Unit 1 of the Oconee Nuclear Station can be operated, as described in this section, without undue risk to the health and safety of the public.

Subsequent to issuance of an operating license, the unit will be required to operate in accordance with the terms of the operating license and the Commission's regulations under the surveillance of the Commission's regulatory staff.

2.0 FACILITY DESCRIPTION

Unit 1 is one of three reactors to be operated at the Oconee Nuclear Station. Units 2 and 3, which are currently under construction, are in most respects identical in design to Unit 1. The units will share a number of auxiliary systems when Units 2 and 3 become operational, including the service water system, portions of the condenser cooling water system, and the auxiliary power supplies. However, the engineered safety feature components except for portions of the hydrogen purge systems, will not be shared between the units. The controls for Units 1 and 2 will be located in separate portions of a common control room (Unit 3 will have a separate control room).

Cooling water to condense the steam from the turbine generator of Unit 1 and of Units 2 and 3 in the future, will be obtained from an artificial lake, Lake Keowee. Lake Keowee was created for the Oconee Station by erecting the Keowee dam to the immediate north of the reactor facility, and the Little River Dam about 5 miles south of the facility. A dike was provided to obtain proper elevation for the common inlet water canal. The Keowee Hydro Station, located just below the Keowee Dam, will provide standby auxiliary power to the Oconee Station.

Unit 1 employs a closed cycle pressurized water reactor. The reactor fuel is slightly enriched uranium dioxide in the form of ceramic pellets contained in zircaloy tubes. Water serves as both the moderator and the coolant for the reactor. The reactor coolant system consists of two separate coolant loops, each provided with a steam generator and two 25% capacity pumps to circulate the coolant. The heated water flows through the steam generators where heat is transferred to the secondary (steam) system. The water then flows back to the pumps to repeat the cycle. The system pressure is controlled by the use of a pressurizer connected to the reactor coolant system by a thermally isolated line.

The steam is used to drive the unit's turbine generator. In addition, this steam powers the normal and emergency pumps which return secondary system condensate (feedwater) to the steam generators. This steam also performs the auxiliary functions of (1) powering the air ejector exhaust system which removes non-condensable gases from the secondary system and (2) preheating the feedwater before it is returned to the steam generators.

Three condensers (one for each of three low pressure stages of the turbine) are used to condense the steam. The heat of condensing steam is rejected to the circulating water system and, under normal conditions, is discharged to Lake Keowee. There is a turbine bypass line which is designed to permit the unit to withstand a turbine trip without requiring a reactor trip.

The reactor coolant system is housed inside the reactor building which is a steel-lined, leak-tight, prestressed concrete structure. This building provides a barrier to the release to the environment of radioactive fission products that might be released inside the building in the event of an accident. Auxiliary systems, including the high pressure injection and chemical addition system, the waste handling system, additional cooling systems, and a reactor building penetration room ventilation system are housed separately, principally in the adjacent auxiliary building. The auxiliary building also houses components of the engineered safety features, the reactor control and protection systems, and the control room. A fuel handling facility common to both Units 1 and 2, is provided for storage of spent fuel, for removing fuel from and introducing fuel into the reactor building, and for handling fuel casks to be received and removed from the facility.

The reactivity of the reactor core is controlled by top entry control elements (neutron absorbers) that are moved vertically within the core by individual control drives. Boric acid dissolved in the coolant also is used as a neutron absorber to provide long-term reactivity control.

A reactor protection system is provided that automatically initiates appropriate actions whenever plant conditions monitored by the system approach preestablished limits. The reactor protection system acts to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The engineered safety features that are provided include an emergency core cooling system that will cool the reactor core in the event of an accident that results in loss of the normal coolant. Reactor building emergency cooling units and spray systems are provided for removal of heat from the building atmosphere should such action be required. A reactor building penetration room ventilation system is provided to remove radioactivity from reactor building penetration leakage in the event of an accident. Also, a hydrogen control system has been provided to limit the accumulation of hydrogen within the containment by means of purging to the atmosphere in the event of a loss-of-coolant accident.

3.0 SITE AND ENVIRONMENT

3.1 Site Location and Description

The Oconee Station is located in Oconee County, South Carolina, about 8 miles northeast of Seneca, South Carolina. The site is adjacent to Lake Keowee which was formed by impounding the Keowee and Little Rivers with separate dams and then joining the lakes by a canal about one-half mile north of the site. The nuclear station is about eight-tenths of a mile west of the Keowee River.

Anderson, South Carolina, the nearest population center (1960 population of 41,136), is 21 miles to the south.

The minimum radius of the exclusion area* for the Oconee Station is 1 mile. The applicant has chosen a distance of 6 miles as the outer boundary of the Low Population Zone**(LPZ). The estimated population within the LPZ is 3,400 people in 1970 and 8,900 people by 2010. The applicant estimates that the transient population within the LPZ will be 2,000 in 1970 increasing to 19,000 by 2010. We have calculated offsite doses for the design basis accidents that have the greatest potential for radiological consequences and have found the calculated doses to be less than the guidelines of 10 CFR Part 100.

3.2 Meteorology

During our review of the Oconee site, prior to issuance of the construction permits, we and our meteorological consultant, the National Oceanic and Atmospheric Administration (NOAA), formerly the Environmental Science Services Administration (ESSA), assumed a "valley" diffusion model to characterize the meteorology of the site. It was assumed that the effluent released as a consequence of a reactor accident would be channeled generally down the Keowee River

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*Exclusion area is defined in the Commission's Site Criteria, 10 CFR Part 100, as that area surrounding the reactor in which the reactor licensee has the authority to determine all activities including removal of personnel and property from the area.

**Low population zone is defined in the Commission's Site Criteria, 10 CFR Part 100, as the area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident.

Valley to the nearest site boundary. Subsequent observations, however, proved that the "valley" model was not appropriate for the Oconee site. To provide a technical basis for a meteorological model, the applicant conducted 15 gas-tracer diffusion experiments using sulfur hexafluoride (SF_6) gas under inversion conditions. In all cases, the measured centerline concentration of the gas tracer was lower than that predicted assuming Pasquill type F diffusion conditions and a wind speed of 1 meter per second. For the worst case, the measured concentration (at 680 meters) was about one-half of that calculated using the assumed diffusion conditions. In all other cases, the measured concentration was a smaller fraction of the predicted concentration.

An examination of data obtained for a period of 1 year for the joint frequency of wind speed, direction, and stability condition (ΔT) indicates that a diffusion rate equivalent to, or worse than, that associated with Pasquill Type F conditions and a wind speed of 1 m/sec occurs less than 5% of the time.

In evaluating the radiological consequences of the design basis accidents, we have employed our standard meteorological model using Pasquill Type F diffusion conditions, a wind speed of 1 meter per second (which was shown to be applicable by the onsite data), credit for the effect of building wake and an additional correction factor of 2.2 that we have concluded is justified by the gas tracer (SF_6) measurements discussed above. The comments of our consultant, NOAA, attached as Appendix C, support the use of this model including the building wake credit and correction factor.

3.3 Hydrology

We evaluated the requirements for flood protection of the facility prior to issuance of a construction permit and during our current review.

The applicant, using a maximum hypothetical precipitation of 26.6 inches of rainfall within a 48-hour period occurring over the entire affected drainage areas, calculated that the resulting flood would create a lake stage of 808 ft above MSL (mean sea level). Plant grade and critical plant components, at 796 ft above MSL, are protected against floods to a height of 815 ft above MSL by the Keowee Dam and intake canal dike. We have made an independent analysis of the anticipated wave effects and have concluded that the 7 feet of freeboard between the calculated flood stage and the protection level of 815 ft above MSL is adequate to protect the nuclear plant.

Based on the considerations discussed above, we and our hydrological consultant, the U.S. Department of Interior, Geological Survey,conclude that the hydrological conditions related to public health and safety during operation at Oconee Station are acceptable. The comments of the U.S. Geological Survey are attached as Appendix D.

3.4 Geology and Seismology

During our review of this site prior to issuance of construction permits, we and our consultant, the U.S. Geological Survey, concluded that the geology of the site provides an adequate founding medium for the facility buildings and structures. The seismic ground accelerations used by the applicant were (a) 0.05 g for the Operating Basis Earthquake* and (b) either 0.10 g or 0.15 g (0.10 g for items on bedrock and 0.15 g for items where bedrock is covered by overburden) for the Design Basis Earthquake.** Prior to construction we and our consultant, the U.S. Coast & Geodetic Survey, concluded that these values were acceptable for the site. Since no adverse information, affecting the adequacy of the founding medium, was developed during excavation or construction, we have concluded that these ground acceleration values are still acceptable.

A strong motion seismograph has been installed to record data related to ground motion in the event of a seismic disturbance at or near this site. These data would be employed in an evaluation of the effects of the seismic disturbance to assure the capability for continued safe operation of the plant. The Technical Specifications require periodic surveillance and testing of this seismograph.

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*"Operating Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems and components, necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional.

**"Design Basis Earthquake" for a reactor site is that earthquake which produces vibratory ground motion for which those structures, systems, and components, necessary to shut down the reactor and maintain the unit in a safe shutdown condition without undue risk to the health and safety of the public are designed to remain functional.

3.5 Environmental Radiation Monitoring

The principal requirements for the applicant's environmental radiation monitoring program are listed in the Technical Specifications. The applicant provided preoperational environmental monitoring data obtained from a program initiated in January 1969. These data provide information on the background radioactivity in the Oconee Nuclear Station area prior to plant startup and we have concluded that they provide acceptable reference data for the continuing environmental radiation monitoring program. The preoperational program included analyses of samples of water, airborne particulates, rain, settled dust, silt (river and lake), vegetation, aquatic vegetation, algae and plankton, fish, milk, and animals. No anomalies in environmental radiation levels have been indicated by the preoperational data thus far reported.

The operational environmental monitoring program will be expanded to include two additional onsite air monitoring stations, a continuous water sampling station on the Keowee River, and a thermoluminescent dosimeter network within the exclusion radius.

The Fish and Wildlife Service of the U.S. Department of the Interior has also reviewed the applicant's program and its recommendations have been incorporated into the applicant's environmental radiation monitoring program. The report of the Fish and Wildlife Service is attached as Appendix E. We have concluded that that the applicant's program will be adequate for monitoring the radiological effects of plant operation on the environs and for assessing the effects of releases of radioactivity to the environment from operation of the plant on the health and safety of the public.

4.0 REACTOR DESIGN

4.1 General

The design of the Babcock & Wilcox Company (B&W) reactors for the Oconee facility is similar, in most respects, to the design of other pressurized water reactors that we have recently approved for operation. They employ full- and part-length control rods, a chemical neutron absorber, and Zircaloy fuel cladding. A unique feature of the B&W design is the use of internal vent valves to prevent steam binding in the event of a loss-of-coolant accident (LOCA) resulting from the rupture of a main coolant inlet pipe to the reactor (cold-leg break). The initial cores for Oconee Units 1, 2, and 3 will be of the same basic design except that Unit 2 will employ burnable boron rods and Unit 3 will utilize fuel from Unit 1.

4.2 Nuclear and Thermal-Hydraulic Design

Nuclear Design

Our review of the nuclear design of the Oconee reactors was based on the information provided by the applicant in the Final Safety Analysis Report 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 designs, and has provided examples to demonstrate the ability of these methods to predict the results of critical experiments using UO_2 and PuO_2-UO_2 fuel. We have concluded that these examples demonstrate the validity of the methods used to predict the reactivity of large power reactor cores.

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

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 the physics characteristics of the reactors.

To allow for changes of reactivity due to reactor heatup, operating conditions, and fuel depletion, the cold, clean Unit 1 core has a significant amount of excess reactivity. The applicant has provided substantial information showing that means have been incorporated into the design to control this excess reactivity at all times through use of soluble boron in the reactor coolant and movable control rods that can be inserted in the core. Addition of boron to the coolant and insertion of the control rods into the core introduce negative reactivity. The soluble boron alone is capable of maintaining the cold core subcritical with 1% negative reactivity even with all control rods removed (as could be the case during the refueling shutdown). For all expected conditions of operation, the control rod assemblies alone have sufficient negative reactivity worth to bring the reactor to a hot shutdown condition (subcritical with at least 1% negative reactivity) even with the most reactive control rod assembly withdrawn and thus not contributing to the shutdown.

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 and the soluble boron system to assure shutdown capability.

The basic instrumentation for monitoring the nuclear power level and distribution in the Oconee reactors 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 detectors that are spaced approximately 90° apart around the reactor pressure vessel. Also, 52 assemblies of self-powered in-core neutron detectors are available for in-core mapping. Each assembly can measure local neutron flux at seven elevations in the core. Special calibration tubes and temperature detectors are provided in the Oconee 1 core. Normally, the output of these detectors will be readout through the plant computer; however, a backup readout system is provided for selected detectors (this capability is required by the Technical Specifications). The applicant's initial position was that the in-core detectors were provided for fuel management purposes, and would be used in the startup program to obtain correlations with out-of-core detector responses. We concluded, however, that the out-of-core detectors are adequate for detecting power maldistributions originating from axial xenon instability and misplaced control rods, only if a power distribution mapping capability is provided by the in-core detectors to calibrate the out-of-core detectors periodically and to investigate any power distribution anomalies detected by the out-of-core detectors. 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).

During plant operation, changes in the core power level or in the control rod configuration can cause time-dependent variations (oscillations) in local power level as a result of variations in the concentration of fission products and their radioactive decay products. The most significant fission product-decay product chain with regard to local core power level oscillations is the decay of iodine-135 to xenon-135, since the xenon is a strong absorber of thermal neutrons. These local power level oscillations can occur even though the average power level of the core is maintained constant, and the magnitude of the oscillations may decrease (converge), remain constant, or increase (diverge) with time.

We have reviewed the applicant's analyses of xenon-induced oscillations which are reported in three B&W Topical Reports BAW-10010 Part 1 "Stability Margin for Xenon Oscillations Modal 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 analysis further indicated that axial xenon oscillations, which are slow changes taking place over several hours, can be controlled by having the reactor operator change the position of the eight part-length axial power shaping rods. In addition, the applicant has agreed to perform tests during the initial startup of Unit 1 to demonstrate the stability of the core against xenon-induced reactivity fluctuations.

The ACRS in its letter (see Appendix B) recommended that we follow the measurements and analyses related to these tests. We will review these measurements and analyses and, in Amendment No. 23, the applicant has stated its intent to cooperate with us in this matter. As added assurance that power maldistributions will not go undetected should they occur, the Technical Specifications (1) require axial and radial power distribution monitoring and control measures to be in effect, and (2) limit the beginning of life (BOL) positive moderator coefficient. These Technical Specifications are comparable to those developed for the Point Beach facility.

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

Thermal-Hydraulic Design

Our review of the thermal-hydraulic design was based on the information presented in the FSAR and in two B&W Topical Reports; BAW-10012, "Reactor Vessel Model Flow Test;" (October 1969) and BAW-10021,"TEMP, Thermal Enthalpy Mixing Program," (April 1970) both of which are incorporated by reference in the FSAR. BAW-10012 is a proprietary report.

The objectives of the vessel model testing described in BAW-10012 were to (1) measure core inlet and outlet flow distributions, (2) measure vessel pressure drop for design optimization, and (3) measure mixing within the vessel. A 1/6-scale model of the vessel was built and operated at scaled flow rates. On the basis of our review of BAW-10012, we concluded that the modeling of flow conditions during phases where one or more of the four pumps were not running did not simulate reactor conditions. In the reactor, reverse flow will occur in a loop with an idle pump, whereas this condition was not permitted in the model. In addition, single-loop operation (i.e., 2 pumps in the same loop running, with the 2 pumps in the other loop not running) was not simulated. The model data indicate that, for the case of all pumps running, there is a preferred flow distribution to the inner zone of fuel elements, and the flow to this zone is about 5% to 10% above the average flow. The flow distribution results from the model were not, however, used as input to the thermal design. Instead, an arbitrary flow reduction of 5% below average in the hot assembly was conservatively assumed.

The Oconee thermal design is based on the W-3 DNB correlation developed by Westinghouse, and the TEMP thermal-hydraulic design computer program developed by B&W.

The W-3 correlation has been in general use for several years. We have reviewed the initial correlation, and subsequent modifications, and have concluded that it is a conservative correlation. We stated in our Safety Evaluation at the construction permit stage of our review of Oconee (August 1967; page 17) that the design could be approved on the basis of the W-3 correlation if the research and development effort to develop a B&W correlation was not completed. B&W is continuing its research in the area of rod bundle heat transfer but an acceptable correlation was not developed in time for use in the Oconee Unit 1 design and the Oconee Unit 1 design used the W-3 correlation.

The coolant flow rates and enthalpies in the core are calculated by the TEMP computer program. This program permits the calculation of individual fuel assembly coolant flow rates. The "hot assembly" flow rate is calculated to be 0.87 of the average flow rate at design overpower conditions; the below-average flow is attributed to maldistribution and friction and acceleration losses. The TEMP program permits energy exchange between parallel flow channels. In this way a control rod channel, with less direct heat input, can receive energy from an adjacent higher-rated channel. The net result is to decrease the enthalpy rise in the hotter channel, and to increase the margin to DNB. (The ratio of local heat flux at which departure from nucleate boiling occurs to the predicted local heat flux is termed DNBR, the DNB ratio).

We have utilized the COBRA-II subchannel analysis program, described in a Battelle-Northwest Report, BNWL-1229, to independently verify the results obtained by the TEMP program. COBRA-II has features that TEMP does not, such as the ability to consider lateral mass flow rate from channel to channel. The results of our calculations show that TEMP conservatively predicts the enthalpy rise and that the use of TEMP and W-3 has resulted in conservative thermal-hydraulic design for the Oconee reactors. At the rated power the calculated DNBR is 2.00; at 114% of rated power it is 1.55. The corresponding values for operation at rated power and full flow rates are 1.84 and 1.40, assuming that there is abnormal bypass flow around the core through a defective vent valve. Since the above DNBR values are greater than the design over-power DNBR value of 1.3, we conclude that the thermal-hydraulic design for the Oconee reactors is suitably conservative for four pump operation.

Our review of proposed operation with less than all four pumps operating revealed a lack of adequate verification of proper flow distribution across the core. Consequently we have required: (a) greater margins, in terms of minimum DNBR, for partial-loop operation (the increased margins will be obtained by limiting the permissible power level for such operation), and (b) restrictions that will permit 2-pump, single-loop operation only for test purposes, until the applicant has submitted a report of the tests to justify single-loop operation. These restrictions are included in the Technical Specifications.

The startup tests will provide information to check the validity of the applicant's assumption of relatively uniform flow distribution during partial-loop operation. We have concluded that operation with one pump off, or with one pump in each loop off, is acceptable; but that operation with the two pumps on in one loop, and the two pumps off in the other loop, is acceptable for startup testing purposes only.

The applicant has provided thermocouples in the inital Oconee Unit 1 core in order to verify core performance during initial startup tests and operation. The applicant did not intend to maintain these thermocouples in the core on a continuing basis. The ACRS in its letter (see Appendix B) noted the desirability of continuing the use of some of these thermocouples. The applicant has since indicated, in Amendment No. 23, that the thermocouples, considered to be part of the incore instrumentation, will be retained as long as they provide reliable results, and that removal of the thermocouples will be preceded by discussions with the regulatory staff.

4.3 Fuel Design

The applicant originally proposed the use of unpressurized fuel for the Oconee units. In Amendment No. 17 (Revision 9 to the FSAR) dated August 11, 1970, the applicant introduced a modified design with helium prepressurized fuel rods, similar to those proposed and accepted for use in the H. B. Robinson plant, in 80 of the peripheral fuel assemblies in the initial core for Unit 1. All fuel subsequently fabricated for Units 1, 2, and 3 will use prepressurized fuel.

The Unit 1 initial core will contain 97 fuel assemblies using the original unpressurized fuel rod design. Approximately one-half of the Unit 1 initial core fuel assemblies using unpressurized fuel rods and approximately one-half of those using prepressurized fuel rods will be utilized in the first cycle core of Unit 3.

The stated purpose for the use of the helium prepressurized fuel rods is to minimize clad stresses and strains from routine reactor power and pressure cycles. Prepressurization of the fuel rods with helium partially offsets the reactor coolant pressure stresses, reduces the net compressive stresses on the clad over the fuel cycle, and serves to lengthen the period of time before the fuel pellets contact the clad as a result of the combined action of fuel pellet swelling and clad creep. In this respect prepressurization of the Oconee fuel rods should improve the fuel rod integrity over the fuel lifetime.

On the basis of our initial review, we requested and the applicant provided (in Amendment No. 18) information on (1) the effect of the fuel on the initial and second fuel cycle reactivity levels, (2) the effect of the reactivity worth of ejected and stuck rods, (3) the effect of the fuel on moderator and power coefficients at beginning of life, (4) the effect of the fuel on the currently predicted consequences of a loss-of-coolant accident, (5) detailed fuel rod design changes to accommodate the pressurization, (6) results of evaluations of the internal fuel rod pressure as a function of burnup, for variations in pellet density, enrichment, and temperature, (7) results of evaluations of fuel rod performance under anticipated transients, and (8) the proposed fuel rod surveillance program.

Based on our review of the information presented in the FSAR, including that contained in Amendment Nos. 17 and 18, we have concluded that the incorporation of prepressurized fuel in the Oconee units is acceptable. Because of the intended recycling of Unit 1 fuel and because the Oconee Zircaloy fuel rod designs, both unpressurized and the prepressurized, are the first such designs for a B&W power reactor core, we have required a surveillance program for both types of fuel rods. The details of the surveillance program are included in the Technical Specifications.

5.0 REACTOR COOLANT SYSTEM

5.1 General

The reactor coolant system, including all vessels, pumps, and piping is designed for an internal pressure of 2500 psig and a temperature of 650°F with the exception of the pressurizer, surge line and a portion of the spray line which are designed for 670°F. The system has been designed to withstand, within the stress limits of the codes used in the design, the normal loads of mechanical, hydraulic, and thermal origin, plus those due to anticipated transients and the operating basis earthquake.

5.2 Reactor Coolant System Components

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

Piping which is part of the reactor coolant system has been designed to the ANSI B31.7 Code for Nuclear Power Piping, dated February 1968, including the June 1968 errata.

Nondestructive examination requirements for reactor coolant system pumps and valves include radiography of castings, ultrasonic testing of forgings, dye penetrant inspection of pump and valve body surfaces, and radiography of circumferential weldments. This program upgrades the nondestructive testing of pumps and valves within the reactor coolant pressure boundary to essentially that of the ASME Code for Pumps and Valves for Nuclear Power.

Our evaluation of specific components is discussed in the remainder of this section.

Reactor Vessel

As noted above, the reactor vessel has been designed and fabricated in accordance with Section III of the ASME Boiler and Pressure Vessel Code. Applicable Code Cases are 1332, 1335, and 1336. Fabrication materials are low alloy steel plate, Type SA-533, Grade B, Class I and forging steel Type SA-508-64, Class 2. The vessel interior is clad with Type 304 austenitic stainless steel. The requirements for nondestructive examinations have been limited to those required by Section III, except that head and shell plate material and flange forgings have been given a 100% volumetric examination using both longitudinal and shear wave ultrasonic test techniques.

The effect of accumulated neutron exposure on the vessel is to raise the nominal transition temperature at which the material may be subject to brittle fracture. Above this transition temperature the material behaves in a ductile manner. The Technical Specifications require that the vessel temperature be kept well above the transition temperature for all conditions of operation. We have concluded that (1) the accumulated neutron exposure value used for the design of the reactor vessels is acceptable and (2) adequate Technical Specifications have been established to assure that the vessel temperature and pressure will be maintained within acceptable limits.

On the basis of our review, we have concluded that the reactor vessel as designed and fabricated, is acceptable.

Reactor Internals

We and our seismic design consultant, John A. Blume and Associates, Engineers, performed a detailed review of the methods of analysis used by B&W to determine the combined loading stresses on the reactor internals and have concluded that the design of the internals is acceptable. The final report from our consultant is attached as Appendix F.

Internals Vent Values

On page 54 of our August 4, 1967 Safety Evaluation of the Oconee Nuclear Station, we noted (1) that the applicant had acknowledged the possibility of core flooding being prevented by formation of a vapor lock or "steam bubble" between the core and a water leg in a steam generator after a cold leg pipe break and (2) that the applicant was considering means of prevention. As reported in the FSAR, the means selected was a system of hinged-disc vent valves installed in the core support shield in the upper plenum region above the core. In the event that steam pressure buildup in the upper plenum region exceeds the pressure in the annular inlet region by 1 psi during a cold leg break, the valves will open, relieving the steam pressure and permitting emergency coolant to cool the core.

Proprietary Topical Report BAW-10005 "Internals Vent Valve Evaluation" (June 1970), which the applicant has incorporated in the FSAR by reference, describes the results of extensive analysis and testing performed to demonstrate the capability of the valves to perform as intended under accident conditions. On the basis of our review, we have concluded that (1) the vent valves will perform their intended function without significant vibration during normal operation: (2) the valves have adequate capacity to permit the core to be cooled even if one valve were to fail closed; (3) the predicted plastic deformation resulting from the impact with the vessel wall when the valve is initially opened under LOCA conditions is acceptable: (4) the valves can and must be inspected, removed, and replaced after installation: (5) during normal operation, core bypass flow caused by valve seat leakage, would not be significant with all valves in place; and (6) the core bypass flow that would result if one valve disc were completely removed (4.6% reduction in core flow) is acceptable (see Section 4.2 of this evaluation).

Control Rod Drives

The rack and pinion control rod drive design, accepted at the construction permit stage of our review, will not be used. B&W has developed a control rod drive that utilizes the roller nut principle of operation. The applicant will use this roller nut type of drive for the Oconee units. Control rod drives of this type have been operating in the Shippingport Pressurized Water Reactor since 1957.

We have reviewed the basic design, development history, and test program for these drives as described in Topical Report BAW-10007, "Control Rod Drive System Test Program," (June 1969) and Supplement 1, thereto (June 1970).

Based on our review of the design and testing program, as reported in BAW-10007 and Supplement 1, thereto, and the successful experience with roller nut drives in reactor service, we have concluded that the use of this type of drive is acceptable for Unit 1.

Coolant Pump Replacement and Associated Piping

The reactor coolant pumps originally installed in Unit 1 have been replaced with Westinghouse pumps similar in design to those approved for several other plants presently under construction.

The applicant has stated that the conclusions of all previous analyses performed for the Unit 1 reactor coolant system remain valid for the system with the replacement pumps installed. The replacement pumps offer greater resistance to backflow than the original pumps. The changes in predicted backflows have been documented in Amendment No. 18 and provide a basis for establishing the acceptability of the actual flows measured during the startup tests.

Modifications to the Unit 1 reactor coolant piping necessary to accommodate the replacement pumps consisted of the addition of a stainless steel transition piece at each pump inlet and a small-angle stainless-steel-clad carbon steel elbow at each pump outlet.

In Amendment No. 20 the applicant presented the results of the stress analysis of the Unit 1 reactor coolant system with the Westinghouse pumps incorporated. These results confirm that the modified system configuration with the replacement pumps, is in compliance with the ANSI B31.7 Code for Nuclear Power Piping.

The primary coolant pipe sections that were removed for rework to accommodate the new pumps (and a small area of pump suction piping that was not removed) were found to have some minor indications of cracking in the stainless steel cladding. The cracking was considered to be the result of deficiencies in the quality assurance program. Based on an extensive investigation which included review of shop fabrication records and a 100% dye-penetrant test of all Unit 1 clad piping at the site, the applicant has determined, and we agree, that all Unit 1 clad piping affected by these deficiencies has been identified and appropriate corrective action has been taken. The details of the applicant's investigation referred to above have been discussed with us and will be included in a formal report to be incorporated in the records of this application in December 1970.

Based on this additional information we have concluded that the pump replacement modification is acceptable.

Steam Generators

The once-through steam generator is a new design based on a B&W research, development and testing program which is described in Proprietary Topical Report, BAW-1002, "Once-Through Steam Generator Research and Development Report," (August 1969) and Supplement 1 thereto (June 1970). The significant safety related aspects of the program are discussed below.

Several tests were performed to show that the steam generator tubes can withstand differential temperatures greater than those used in design without failure. Under normal operation the tubes are hotter than the shell and will be compression-loaded axially. A buckling test was performed to determine that the compression loading required to cause buckling was significantly greater than that expected in the full size unit.

The measured natural frequency of the tubes over the expected range of compression loads was between 30 and 40 cycles per second (H_z). A successful limited vibration test was performed with a single tube driven at 35 H_z. The applicant presented an analysis showing that under operating conditions, the vortex shedding frequencies near the bottom tube sheet and near the top tube sheet due to feedwater and steamflow are not in the measured natural frequency range and therefore are not expected to excite significant motion.

A 37-tube full-length model was subjected to two simulated primary side blowdown events. The fastest event started with a nominal pressure of 2200 psig and a temperature of 578°F on the primary side and a nominal pressure of 1350 psig on the secondary side. As the event progressed the pressure across the tube and tube sheets reversed from about 850 psi initially in one direction to about 1000 psi, at about 8 seconds, in the reverse direction. The second test was not quite as severe in that the initial differential pressure was about 700 psi and reversed to about 800 psi in about 100 seconds. This model was also subjected to a simulated steam line failure. The water in the steam generator was allowed to flash to steam with the pressure dropping to zero in 25 seconds. Just prior to this step change in steam flow, the model was operated at simulated 100% load primary and secondary conditions. The resultant temperature differential across the tube sheet was shown to be within design limits. Further, hydrostatic testing and examination of the model following these and other transients tests confirmed structural integrity of the model.

Sections of two sets of full size tubes with design center spacing, were used to evaluate the effects of normal and tangential impinging jets of 3 gpm borated water maintained at nominal reactor coolant pressure and temperature. The test, which lasted 9.5 hours, demonstrated no significant erosion or enlargement of the 1/16-inch diameter holes used to simulate the leaks.

Since chemical cleaning will be required to remove crud buildup in the steam generators, we required information to demonstrate that a careful, well planned and pretested cleaning procedure had been developed, and that temperatures, pressures, soak and drain times, and corrosion properties of the solvent had been taken into account. Based on our review of that information, we have concluded that an adequate cleaning procedure has been developed, and that its proper use will not lead to a significant potential for multiple tube failures.

A field testing program for one of the two Unit 1 steam generators will make use of 60 thermocouples (in 10 of the 15,531 tubes) and 10 differential pressure transducers to verify that actual performance is as predicted from the calculations and from extrapolation of the 7-, 19-, and 37-tube model test results. Data will be obtained on temperature distribution within the tube bundle and across the tube sheet. The pressure transducers will remain on the steam generator for long-term monitoring purposes.

Because the applicant could not assure us that the feedwater ring header supplying the steam generator would withstand the jet forces associated with a postulated break in the nearby reactor coolant piping, we requested additional analyses of the consequences of postulated breaks in both the hot-leg and cold-leg reactor coolant piping within the steam generator compartment. The applicant reported the results of these analyses in Amendment No. 17 and concluded that for certain types of breaks additional pipe restraint capability would need to be provided in order to limit the energy in the broken pipe and to distribute the resulting forces over a larger surface area on the steam generator shell. These restraints will be installed prior to operation. We have reviewed the applicant's analyses and the proposed design revisions and we have concluded that with the installation of the additional restraints, there is reasonable assurance that a reactor coolant pipe break will not rupture the shell of the steam generator as a result of pipe whip effects. Those sections of piping whose failure could lead to jet forces capable of damaging the feedwater ring header will, as a Technical Specification requirement, be subjected to stringent surveillance during the life of the plant. Because of this surveillance, and the low probability of a failure in the piping at the specific locations of concern, we conclude that this design is acceptable for the Oconee Station.

Based on our review of the information contained in the application, including the B&W Topical Reports, we conclude that there is reasonable assurance that the Oconee Unit 1 steam generators, the first production units of this design can be operated safely. Prior to operation at power levels above 2452 MWt, we will review the results obtained from the field test program to determine that adequate design verification has been obtained. Also, we have required that special attention be given in the inservice inspection program to those sections of reactor coolant piping which, if they were to fail, could result in jet forces of such magnitude and direction as to cause the failure of a steam generator feedwater ring header.

5.3 Evaluation of Other Class I (Seismic)* Mechanical Equipment

All Class I (seismic) components and systems, including engineered safety features and other systems important to the safety and operation

*See Section 6.1 for definition of Class I (seismic) structures, systems, and components.

of the plant, have been designed to withstand the loads resulting from the design basis earthquake, in conjunction with other applicable loads, without loss of function. In response to our request, the applicant submitted, in Amendment No. 15, the results of supplemental analyses performed for Class I (seismic) equipment that compared the specified accelerations for this equipment with the accelerations predicted from floor response spectra. In addition, Amendment No. 18 provided the results of the dynamic analysis of the reactor coolant system surge line. On the basis of our review we and our seismic design consultant have concluded that the analytical techniques employed for Class I (seismic) equipment are acceptable.

We have discussed with the applicant the adequacy of the measures taken to (1) verify the adequacy of the vendor's method of certification for Class I (seismic) equipment, (2) identify the design organizations involved in seismic design and their responsibilities, and (3) assure that the documented procedures to provide for the interchange of design information between the involved organizations were used in an acceptable manner. We have concluded that the applicant's program in this area was acceptable.

5.4 Missile Protection

The applicant has described the evaluations made to assess the measures taken to protect the reactor coolant system, the reactor building liner, and associated engineered safety features from missiles that might be generated as a result of component failures inside the reactor building. The postulated missiles include valves, valve bonnets, valve steams, temperature sensors and control rod drives. We have reviewed the information provided by the applicant, including that given in Amendment Nos. 12 and 20 in response to our request for further information, and have concluded that the design features provided, consisting principally of concrete shield walls, equipment supports, and piping restraints, will afford adequate protection against missiles inside the reactor building.

The primary pump-motor flywheels to be used at Oconee are of Westinghouse design and manufacture and are similar to those used or proposed for use in other PWR plants. The flywheels are fabricated of vacuum degassed A533B steel plate, have been subjected to rigid quality control during the fabrication, will receive preservice baseline volumetric and surface examinations, and will be monitored during operation by an inservice inspection program that meets our requirements. We have concluded that these measures provide reasonable assurance that the integrity of the flywheels will be maintained.

5.5 Inservice Inspection

The program required to monitor the effects of radiation on the pressure vessel is comparable to those we have approved for recently-licensed operating facilities, and is acceptable.

The applicant has applied the rules of Section XI of the ASME Code, Inservice Inspection of Nuclear Reactor Coolant System, as the basis for the inservice inspection program that is required by the Technical Specifications. Periodic inservice inspections of the primary pump motor flywheels will also be conducted.

The applicant will review the inservice inspection program with us after 5 years of reactor operation. It may then be modified based on experience gained during these 5 years. At that time, we will also require the applicant to perform such inspections of components outside the reactor coolant pressure boundary as deemed necessary to provide continuing assurance of structural integrity.

On the basis of our review, we have concluded that the applicant's inservice inspection program is acceptable.

5.6 Vibration Monitoring

Amendment No. 17 to the application outlined a proposed reactor internals vibration monitoring system for Unit 1. The system consisted of four biaxial accelerometers to measure the midspan vibratory motions of the pressure vessel surveillance specimen holder tubes, and the midplane vibratory response of the thermal shield. Each accelerometer was to be capable of measuring frequencies over a range of 2 to 300 cycles per second (H_z) and accelerations up to 30 g. Upon completion of preoperational testing, the reactor internals were to be removed and inspected for evidence of fretting and wear. We questioned the adequacy of this monitoring system since the proposed instrumentation would not directly measure the response of several components of interest, i.e., the control rod assembly guide tubes, the upper and lower fuel support structures, and the core barrel, nor would it provide for the direct measurement of the intensity and frequency of the basic hydraulic forces. As a result of additional discussions, the applicant improved the system, as described in Amendment 19, by attaching three additional accelerometers to the plenum cylinder above the core for the purpose of measuring the shell-mode vibrations of the plenum cylinder. The applicant made a further improvement, as described in Amendment No. 20 by adding an additional accelerometer to the upper core support barrel (approximately 2 feet below the internals vent valve assembly at the location of one of the reactor inlet nozzles) to measure the response of the upper core support barrels, and to monitor the effects of the combined forces of fluid flow and inlet-outlet pressure variation. The applicant has also agreed to establish, on the basis of analyses, a "go, no-go" limit for the monitor on the upper core support barrel together with a supporting technical basis. In the event that this limit is exceeded during the startup tests, the applicant has agreed to perform additional analyses or vibration tests of this critical internal component to verify its acceptability. To aid in interpreting the results obtained from the vibration monitors installed on the Unit 1 reactor internals, B&W expects to utilize data obtained by shaker-table testing of a later production set of reactor internals that are identical to the Oconee Unit 1 internals. The data derived from these tests will establish base line information for use in interpreting the Oconee test data.

We have concluded that the applicant's preoperational vibration monitoring program is acceptable with the additions discussed above.

In its letter (see Appendix B) the ACRS recommended consideration of available developmental techniques as an aid in ascertaining displacements, changes in vibration characteristics, and the presence of loose parts in the reactor coolant system. As stated in Amendment No. 23, the applicant is exploring means of utilizing such developmental techniques and has agreed to perform an appropriate neutron noise analysis based on monitoring nuclear instrumentation during initial startup and low power tests. If feasible, the applicant will also install accelerometers at key locations in the coolant system, such as the inlets to the steam generators and the reactor pressure vessel, to monitor the system during operation. The neutron noise measurements will provide reference data that may prove of use in the subsequent detection of changes in the integrity of components within the reactor pressure vessel. We consider these commitments by the applicant to be responsive to the recommendations of the ACRS and to our position that such inservice vibration monitoring should be performed to the extent that it is feasible.

5.7 Leak Detection

The systems for detection of leakage in the reactor coolant pressure boundary monitor radioactivity levels in air particulates and in gases, and monitor water level in the reactor building sump. The instrumentation is redundant, diverse, and provides adequate alarm features. The sensitivity of these systems is consistent with their primary purpose of detecting any leak in the primary coolant system boundary that would be indicative of incipient failure.

The Technical Specifications establish required surveillance methods, minimum instrumentation that must be maintained, the need for safety evaluations to be performed upon detection of any leak, and the time permitted to complete these evaluations prior to specific mandatory action.

We have concluded that the proposed leak detection systems are acceptable.

5.8 Fuel Failure Detection

The radioactivity concentration in the reactor coolant is continuously monitored for indication of a sudden fuel failure by a sodium iodide crystal scintillation detector radiation monitor, located on a 1 to 3 gpm bypass line downstream from the reactor coolant letdown coolers so as to provide delay time to allow N_{16} gamma activity (half-life of 7.1 seconds) to decay before the coolant reaches the monitor. This delay reduces gamma ray background and improves monitor sensitivity. This monitor will activate an alarm in the control room upon detection of a high activity level. An alarm is also provided for low flow in the bypass line. We conclude that this system is acceptable.

6.0 REACTOR BUILDING AND CLASS I (SEISMIC) STRUCTURES

6.1 General Structural Design

The applicant established a basic criterion for design purposes that required that there be no loss of a function for the worst loss-of-coolant accident combined with seismic conditions, where that function is related to safety. For convenience, we have referred to structures, systems, and components that meet this criterion as being Class I (seismic) items. In this classification are all structures, described by the applicant, in Appendix 5A of the FSAR, as structures that are designed to prevent uncontrolled releases of radioactivity and to withstand all design loadings without loss of function. Appendix 1C of the FSAR classifies all systems in a manner that identifies those piping systems, including valves and components, that are designed to prevent uncontrolled releases of radioactivity and to withstand all loadings without loss of function.

We have reviewed the classifications for the facility structures and systems, as set forth in Appendix 5A and Appendix 1C, and have concluded that the classifications are appropriate and that all items important to safety have been designed to Class I (seismic) standards.

The Class I (seismic) structures at Oconee Station Unit 1 include the reactor building, portions of the auxiliary building (housing the control room, penetration room, fuel pool, radioactive waste disposal systems, engineered safety features, and associated instrumentation and power systems outside the reactor building), portions of the cooling water intake system (underwater weir, canal dike, and pumping station), principal structural components of the Keowee Hydro Station, and portions of the turbine building to the extent required to protect the main steam lines and those portions of the circulating water system that supply water to engineered safety feature coolers.

The environmental conditions that were considered in the structural design include the operating basis earthquake (OBE), the design basis earthquake (DBE), potential floods, and wind and missiles due to a postulated 300 mph tornado. We have concluded that these conditions were used for the design in an acceptable manner.

6.2 Reactor Building Structural Design

The reactor building is principally of prestressed concrete construction with design details similar to those of the Point Beach containment. The structures interior to the reactor building are of massive reinforced concrete construction. The auxiliary building is of reinforced concrete columm, beam, and slab construction. The turbine building is of structural steel with steel panel siding. The loads, load combinations and methods of application are in accord with current good practice, and are acceptable.

On the basis of our review and that of our seismic design consultant, we conclude that the Class I (seismic) structures, systems, and components of Unit 1 are designed to accommodate all applicable loads and are acceptable. The report of our seismic design consultant is attached as Appendix F.

With regard to both tornado and turbine-generated missiles, Class I (seismic) components in the auxiliary building will either be protected by concrete walls and roofs designed to prevent potential missile penetration, or be separated to prevent failures in redundant systems from such missiles. We have concluded that the reactor building, spent fuel building, and other Class I (seismic) structures, components and systems have been adequately protected against potential missiles.

The Oconee Nuclear Station Unit 1 reactor building has a free volume of 1.9 x 10^6 cubic feet and a design pressure of 59 psig. We have evaluated the pressure transients that might occur in the containment in the event of a loss-of-coolant accident assuming various sizes of primary coolant system breaks. For the range of postulated break sizes up to and including the double-ended severance of the largest reactor coolant pipe, the largest calculated peak containment pressure is 53.9 psig and occurs with a 5 ft^2 rupture for the 36-inch diameter hot leg pipe. The design pressure of the containment exceeds this calculated pressure by more than 9% and is acceptable.

We have also considered the pressure transient that could occur in the containment in the extremely unlikely event that the loss-of-coolant accident pipe break that is postulated to cause a loss-of-coolant accident also caused the loss of the feedwater ring header of one of the steam generators. Based on actuation of minimum required heat removal systems at 25 seconds after the pipe break and the use of the Tagami condensing heat transfer correlation, the applicant, in Amendment No. 20 calculates a peak design pressure of 59.3 psig occuring 51 seconds after the rupture. Although this peak pressure resulting from the combined blowdown of the primary and secondary systems slightly exceeds the design pressure (by 0.3 psi), we have concluded that it is acceptable because the reactor coolant piping layout and restraints act to minimize the kinds and location of reactor coolant pipe breaks that could cause such a loss of the feedwater ring.

Two horizontal tendons and one vertical tendon were left out of the Unit 1 reactor building structure as a result of construction error. The applicant evaluated the possible effect of this omission on the capability of the Unit 1 structure to withstand design loadings and the impact of missiles. At our request, the results of this evaluation were submitted in Amendment No. 17 for our review. On the basis of our review of that evaluation we have concluded that the structure with these tendons missing is acceptable since building stresses remain within allowable design values and the area of the reactor building involved is shielded from a direct turbine missile strike.

6.3 Reactor Building Testing and Surveillance

Initial proof testing of the reactor building structure will consist of a preoperational structural test at 115% of the 59 psig design pressure. A preoperational initial leak rate test will be conducted at the 59 psig design pressure for 24 hours followed by an additional test at a reduced pressure of 29.5 psig. The allowable maximum test leak rate has been set at 0.25% of the contained air weight per day.

For confirmation of continued structural integrity during service life, the applicant will perform a surveillance program on nine tendons distributed throughout the reactor building structure. Periodic visual inspections of the building interior, including the liner and penetrations, will also be performed. To assure continued leak tightness of the building, the applicant will perform periodic integrated building leakage rate tests and more frequent periodic leak rate tests of various parts of the building such as penetrations and air locks. The details of this program are provided in the Technical Specifications.

The above programs provide appropriate means for determining the initial and continued acceptability of the integrity and leak tightness of the reactor building.

6.4 Penetration Room Ventilation System

We have reviewed the design of the penetration room provided for each unit and have concluded that the negative pressure that will be produced in the sealed room by the two independent fans will assure that postaccident containment radioactive leakage into this room will be filtered to reduce its radioactivity prior to release to the atmosphere (See Section 11.2 of this evaluation). As initially designed, the system was deficient in that: (1) there was no remote indication of filter flow, and (2) the design required an operator to go to the ventilation room, in the vicinity of the filter assemblies, under accident conditions to either correct for reduced flow resulting from filter blockage or to reestablish flow by valve realignment in the event that flow was lost in one of the filter assemblies.

The applicant revised the design to correct these deficiencies. The key features of the design revision were (1) the elimination of a manually adjusted bleed air valve (which was a potential source of uncontrolled release), (2) addition of a second vacuum breaker valve, (3) provisions for the remote operation of all valves that may require operation during the course of an accident, (4) provisions for remote readout of filter flow, and (5) provisions to lock open the valves upstream of the filter assemblies.

Based on the calculations provided in Amendment No. 18, we have concluded that remote monitoring of filter conditions at 4-hour intervals under accident conditions is sufficient to detect a loss-of-filter-flow problem and permit corrective action to be taken in a timely manner.

As noted in Section 6.3 above, the allowable integrated leak rate for the reactor building, under accident pressure conditions, is 0.25% of the contained air weight per day. The Technical Specifications require that no more than 50% of this leakage leave the containment without passing through the penetration room ventilation system. Verification of the permitted leakage distribution will be based on local leak rate testing of individual penetrations.

Based on our review of the penetration room ventilation system design and the provisions available for testing the individual penetrations and the system filters, we have concluded that the installed penetration room ventilation system is acceptable.

7.0 ENGINEERED SAFETY FEATURES

7.1 Emergency Core Cooling Systems

The principal equipment of the emergency core cooling system (ECCS) consists of (1) three high pressure injection pumps which provide makeup cooling to the reactor inlet piping, (2) three low pressure injection pumps, and (3) two core flooding tanks, all connected directly to the reactor pressure vessel. This system provides redundant capability to inject borated cooling water rapidly into the core in the event of a loss-of-coolant accident and to maintain the coolant liquid level above the level of the core for an indefinite period of time following the postulated accident.

In order to comply with Criterion 44 of the Commission's proposed General Design Criteria, the applicant modified the design of the Oconee ECCS from that proposed at the construction permit stage of our review. The system was modified so as to (a) provide redundant recirculation lines from the reactor building emergency sump to the low pressure injection pumps, (b) extend the reactor building containment to include the sump valves, and (c) provide additional valved tie lines in both the high pressure and low pressure injection systems. We have concluded that the system as modified is capable of providing emergency cooling even if there should be a failure of any active component in the ECCS.

The original analysis of this emergency core cooling system to determine performance in the event of a loss-of-coolant accident (LOCA) was based on a hydraulic analysis using the FLASH-1 computer program. This program was limited in that the entire primary system was represented by only three control volumes. The need to develop a more sophisticated analytical program was recognized. The FLASH-2 program, with the capability of considering

20 control volumes and 40 flow paths, was subsequently developed. During construction of the Oconee plant, B&W modified the FLASH-2 program further, notably expanding its capacity so that as many as 40 control volumes and 80 flow paths could be considered in an analysis.

The modified FLASH-2 program developed by B&W did not, however, include the capability of representing the core by distinct control volumes. Because recent evaluations of ECCS performance for PWR plants using more sophisticated computer codes had raised questions regarding the conservatism of previous predictions for large breaks in cold leg coolant piping, we advised the applicant that we considered such representation to be necessary in order to properly characterize the complex reactor core hydraulics occuring during the blowdown transient. We therefore requested the applicant to reevaluate the ECCS performance during a LOCA using a more detailed representation of the core and reactor coolant system.

In Amendment No. 21 to its application, the applicant submitted Appendix 14B which contained the multi-node computer code analysis of the LOCA. The analysis of the LOCA was performed using 27 control volumes to represent the reactor coolant system. In this model 3 control volumes were used to represent the average core characteristics and 5 control volumes were used to represent the theoretical hot channel in the core. Using this model, the applicant evaluated a range of break sizes for the cold leg (inlet) piping up to and including the complete severance of the largest cold leg pipe. The applicant's analysis indicated that cladding temperature transients would be terminated at temperatures below 2300°F and that the peak temperature of 2260°F occurred for an accident involving a 6 ft^2 rupture in the largest cold leg pipe (this break area is equivalent to 1.4 times the cross-sectional flow area of the pipe). The applicant also analyzed the consequences of complete severance of the largest hot leg (outlet) pipe and calculated a peak clad temperature of 1730°F.

In order to obtain an independent confirmation of the applicant's evaluation, we requested the Idaho Nuclear Corporation to perform analyses of the Oconee ECCS performance in the event of a LOCA using digital codes they have developed as part of the Commission-sponsored light water reactor safety program. These analyses were performed with the knowledge and cooperation of the applicant and B&W, and the final results have been made available to them. The results of the independent analyses were in good agreement with those reported by the applicant, and provided additional information regarding the Oconee ECCS performance.

The applicant has stated, in Amendment No. 23, that a report is in preparation by B&W which will present detailed information regarding the computer code formulation, the equations used for the analyses, the empirical correlations assumed for predicting both heat transfer and hydraulic phenomena, and the sensitivity of certain system representations during blowdown of the primary system. This report will be submitted for our review prior to power operation of Oconee Unit 1. The ACRS noted in its letter (See Appendix B) that the Committee wishes to be kept informed of the continued studies of ECCS performance. We will review the additional information and discuss the results of our review with the Committee prior to Oconee Unit No. 1 power operation.

Core flooding tanks are provided as part of the ECCS to reflood the core during the initial stages of a LOCA resulting from large pipe breaks. The core flooding tanks respond immediately in the event of an accident. The high pressure injection (HPI) system requires 10 seconds to become operational and the low pressure injection (LPI) system requires 25 seconds to become operational.

Each core flooding tank is connected directly to the reactor pressure vessel by a 14-inch line. This line is provided with two check valves, in series, and a motor-operated stop valve. During normal operation the check valves prevent high pressure (2185 psig) reactor coolant flow to the lower pressure (600 psig) core flooding tank and the motor-operated valve is maintained in the open position. When the reactor coolant system is below 600 psig, during reactor pressurization and depressurization (normal startup and shutdown conditions), the motor-operated valve on each tank is closed to prevent core flooding tank inventory from entering the reactor coolant system.

In the event of a LOCA, the motor-operated valve must be in the open position and remain open until the core flooding tank has functioned as designed. Should this valve be closed prior to or during a LOCA, the core flooding tank inventory would not be available to reflood the core.

To assure that the motor-operated valves are in the open position during normal operation, and remain open in the event of a LOCA, we have required that (1) there be two independent means of determining valve position, (2) a not-open condition be alarmed in the control room, and (3) the power to the motor drive be removed during normal operation without causing loss of either means of valve position indication or the alarm. The applicant has provided this capability, as noted in Amendment No. 20.

We conclude that the emergency core cooling system will (1) limit the peak clad temperature to well below the clad melting temperature, (2) limit the fuel clad-water reaction to less than 1% of the total clad mass, (3) terminate the clad temperature transient before the geometry necessary for cooling is lost and before the clad is so embrittled as to fail upon quenching and (4) reduce the core temperature and then maintain core and coolant temperature levels in a subcooled condition until accident recovery operations can be accomplished.

In summary, we conclude that the emergency core cooling system is acceptable and will provide adequate protection for any loss-of-coolant accident.

7.2 Containment Spray and Cooling Systems

In the event of a loss-of-coolant accident and the attendant release of reactor coolant system energy to the Unit 1 reactor building, there are two diverse means of removing this heat to prevent excessive building pressure. These two means are a building spray system and a building cooling system using air fan coolers.

Since there are two equal capacity spray subsystems and three equal capacity air fan cooler subsystems, any of the following combinations of equipment will provide sufficient heat removal capacity: (1) both spray subsystems, (2) all three air fan coolers, or (3) two of the air fan coolers plus one of the spray subsystems.

Spray Systems

There are two independent 1500 gpm capacity spray subsystems each of which can provide 50% of the reactor building accident cooling requirement. These systems obtain borated water initially from the borated water storage tank through the interconnections with the low pressure injection system. Upon depletion of this source of spray water system operation is continued by taking water from the reactor building emergency sump. No single failure can cause loss of more than one of these two 50% capacity systems.

Spray pump delivery performance (discharge pressure and flow) will be tested periodically. Isolation valves will be tested "dry" to assure that the valve stems travel to the full open position, as required. Spray nozzles will be air tested. The required test frequencies and acceptance criteria are included in the Technical Specifications.

Since the solution used in the spray systems is acidic (pH less than 5.0), we questioned the applicant regarding the potential for stress corrosion of the emergency core cooling system piping inside containment. In Amendment No. 18 the applicant described the capability provided to monitor the spray solution during the recirculation mode of operation and to increase the pH of the solution to levels that will reduce the potential for stress corrosion.

Based on our review of the design of the spray system, testing provisions, and pH control capability, we conclude that the spray systems are acceptable.

Reactor Building Cooling System

There are three identical reactor building cooling units that together provide 100% of the reactor building accident cooling requirements. Cooling water to each cooling unit is supplied from the low pressure service water system through redundant piping and valves. On the discharge side of each unit the water flow is continuously monitored during both normal and accident conditions. Downstream of the flow monitors all three 8-inch cooling water lines converge into a common 18-inch line, having a single manual valve, prior to discharge into the 30-inch condenser cooling water system crossover header. Although the valve in this 18-inch line will normally be open at all times, at our request the applicant has agreed to lock this valve in the full open position to prevent its inadvertent closure and the resulting loss of all three reactor building coolers.

With this added precautionary measure, we conclude that the reactor building cooling system is acceptable.

7.3 Containment Isolation Systems

In the FSAR the applicant lists four classifications of isolation for all fluid penetrations. We find these classifications acceptable. We have reviewed the individual system flow diagrams and have concluded that the isolation methods used are in conformance with the applicant's classification requirements and that no single failure of an active component, including valve actuation instrumentation channels that must function in an accident situation, could result in loss of isolation or intolerable leakage.

The refueling tubes exit into the bottom of the spent fuel pool under 36 feet of water. The radioactivity contained in leakage that might escape to the atmosphere by this route would be reduced by retention while passing through the pool water and would not contribute significantly to potential accident doses. Under accident conditions the normal sump drain will have a water seal to minimize leakage.

We have reviewed the information provided by the applicant and conclude that the isolation methods used will limit leakage to acceptable levels.

7.4 Post-Accident Hydrogen Control

At the time the applicant was granted a construction permit for the Oconee Nuclear Station, the provision of means to control the post-accident concentration of hydrogen within the reactor building was not a design basis requirement. Since that time we have concluded that such control measures should be provided in the design of a new nuclear facility. However, a requirement to add, eliminate, or modify structures, systems, or components in a facility such as Oconee Unit No. 1 for which a construction permit is in effect, is considered to be "backfitting", in the context of Section 50.109 of 10 CFR Part 50.

In accordance with this Section the applicant would be required to change the Oconee Unit No. 1 facility to provide hydrogen control measures beyond those presently intended for installation, only if the Commission were to find that such action would provide substantial, additional protection required for the public health and safety.

During the construction of the Oconee facility the subject of post-accident hydrogen accumulation was discussed with the applicant. As a consequence the applicant performed an evaluation to determine the potential for accumulation of hydrogen in the reactor building following a loss-of-coolant accident, and provided means to monitor the accumulation of hydrogen and to purge the reactor building in order to control the hydrogen concentration. We have independently assessed the potential for accumulation of hydrogen for Oconee Unit No. 1, using generally more conservative assumptions and have estimated the radiological consequences of purging the reactor building. The assumptions used in our evaluation were:

Post-Accident Hydrogen Generation Assumptions

(1) Fraction of fission product radiation energy absorbed by the coolant

(a) Beta

(1) Betas from fission products in the fuel rods: 0

(2) Betas from fission products intimately mixed with coolant: 1.0

	(b) Gamma (1) Gammas from fission products in the fuel rods, coolant in core region: 0.1 (2) Gammas from fission products intimately mixed with coolant, all coolant: 1.0
(2) $G(H_2)$-Assumed H_2 evolution from absorption of radiation	0.5 molecules/100ev
(3) Extent of metal-water reaction (percentage of fuel cladding that reacts with water)	5
(4) Corrosion rates for materials exposed to the reactor building spray solution	In accordance with applicant's assumptions in FSAR.
(5) Fission product distribution model	(a) 50% of the halogens and 1% of the solids present in the core are intimately mixed with the coolant water. (b) All noble gases are released to the reactor building. (c) All other fission products remain in fuel rods.

Containment Purge Assumptions

(1) Hydrogen concentration limit	4 vol. percent
(2) Purge system iodine filter efficiencies	(a) organic iodine 0.70 (b) inorganic iodine 0.90

Assumptions for Dose Calculations

- The thyroid dose resulting from purging of the reactor building was estimated for an infinite time at the low population zone distance. The meteorological conditions were assumed to be the annual average conditions applicable to the site area.

The results of our calculations indicate that purging operations to maintain hydrogen concentrations of less than 4% by volume would not need to be initiated until about 14 days after the occurrence of a loss-of-coolant accident. The controlling dose that would result from the purging is that to the thyroid. We calculate a potential thyroid dose of about 8 Rem at the low population zone boundary. For comparison purposes it is noted that the calculated 30-day thyroid dose at the low population zone boundary for the loss-of-coolant accident is 200 Rem if purging is not considered.

We have not completed our evaluation to determine what measures, if any, beyond those to be provided by the applicant for post-accident hydrogen control will be required for Oconee Unit No. 1 in accordance with the Commission's "backfit" policy (10 CFR 50.109). This evaluation may not be completed for some time. Because the calculated doses resulting from the loss-of-coolant accident including those resulting from the purging are well within 10 CFR Part 100 guideline values, we have concluded that Oconee Unit No. 1 should be licensed for operation at this time as presently designed.

8.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

8.1 Reactor Protection and Control System

The adequacy of the reactor protection system instrumentation for Oconee Unit 1 was evaluated by comparison with the Commission's proposed General Design Criteria published July 11, 1967, and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE-279) dated August 28, 1968.

The reactor protection system consists of four identical "chains" of relay contacts with all contacts in all four "chains" normally closed. Any of the abnormal operating conditions listed in Table 7-1 of the FSAR will cause a set of contacts in each of the four "chains" to open thus breaking all four "chains". If one of these "chains" fails, proper operation of only two of the remaining three "chains" will initiate signals to open all reactor trip breakers causing all rods, except the part-length axial power shaping rods, to be inserted into the reactor core to make it subcritical. The reactor trip breakers are arranged in two groups and all breakers in either group are required to open in order to trip the reactor (make it subcritical). The Technical Specifications require monthly testing to verify that all reactor trip breakers are operational. Continued operation with a reactor trip breaker that fails to open upon command is not a permitted mode of operation because the design only has a single degree of redundancy in groups of reactor trip breakers.

The ACRS recommended (see Appendix B) that the applicant accelerate its studies of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram when required during anticipated transients. In Amendment No. 23 the applicant noted that preliminary results of studies performed by B&W had been discussed with the regulatory staff and that, upon receipt of additional guidelines, B&W would complete these studies and the results would be submitted to the staff for evaluation. We have since provided the additional guidance to B&W and will review the results of the completed studies as they become available.

We have concluded that the reactor protection and control system meets our proposed General Design Criteria and the applicable IEEE criteria and is acceptable. (See Sections 8.5 and 8.6 of this evaluation for discussions of cable and equipment installation and accident condition testing).

8.2 Initiation and Control of Engineered Safety Features

The engineered safety feature actuation system consists of eight channels. Two independent channels are provided for each engineered safety feature system by using a "split-bus" concept.

We have reviewed the schematic diagrams and the test procedures for the engineered safety feature actuation circuits. The entire system, from the sensors to the actuated components (e.g., pumps, valves) and including bypass provisions, can be tested during reactor operation. During the periodic tests, the channel under test is not incapacitated and a valid trip signal will actuate both channels associated with each engineered safety feature system. Each actuated component has its own unit control module. Although an integrated system test cannot be performed during reactor operation, the individual components can be actuated one at a time, using the associated unit control module, in a manner that adequately tests the action required under accident conditions. We have concluded that an acceptable means of completely testing the engineered safety feature actuation circuits during reactor operation is provided.

In the analysis of the loss-of-coolant accident provided by the applicant in order to show that the fuel temperature can be kept within acceptable limits, it is assumed that a reactor trip occurs in addition to the injection of emergency core coolant. Because of the importance of the emergency core cooling capability, we require all pressurized water reactor systems to be equipped with diverse sources of signals to initiate the actuation of the necessary components.

In the system initially proposed for Oconee in the event of a loss-of-coolant accident, the emergency core cooling systems (high pressure injection system and low pressure injection system) were shown to be activated by a low reactor coolant pressure signal or a high reactor building pressure signal, but the reactor was shown to be tripped only by the low reactor coolant pressure signal. As recommended in the ACRS letter (Appendix B) we have required the applicant to provide an additional, diverse signal for reactor trip for use with the emergency core cooling systems. As noted in Amendment No. 23, the applicant has indicated that the reactor will also be required to trip on high reactor building pressure.

We have concluded that the engineered safety feature actuation system meets our proposed General Design Criteria and the applicable IEEE criteria and is acceptable.

8.3 Offsite Power

Offsite power is available to Unit 1 from the 230 kilovolt (kV) switchyard via a 230/4.16-kV startup transformer. Four 230-kV transmission lines converge at the site via several rights-of-way. The 230-kV switchyard is arranged into a breaker-and-a-half configuration and each circuit breaker is provided with dual trip coils supplied from the two independent 125-Vdc station switching power systems. Circuit protection is provided by redundant relaying. The applicant has stated that the Duke system is designed to withstand the loss of any generating unit within its network. They will demonstrate 100% load rejection capability prior to commerical operation of Oconee Unit 1. We have concluded that the availability of offsite power is acceptable.

8.4 Onsite Power

Onsite power is provided by two 87.5-MVA hydroelectric generators. This power is available either through the 230-kV switchyard and the 45/60 MVA Unit 1 startup transformers or through the 13.8-kV underground feeder which utilizes its own 12/16/20 MVA transformer. The maximum emergency power demand upon initiation of accident conditions would be 4.8 MVA. Each hydro unit has capacity well in excess of this 4.8 MVA requirement, via either circuit, for operation of the engineered safety feature loads.

Three 4.16 kV buses serving engineered safety feature loads are provided for Unit 1 and these buses are connected to both of the Unit 1 4.16 kV main feeder buses. The sources of power which are automatically connected to the main feeder buses, in the order that they are connected, are:

(1) the 230 kV switchyard via the unit's startup transformer;

(2 the preselected hydro unit via the 13.8-kV underground feeder and the station's standby buses; and

(3) the other hydro unit via a 230-kV overhead line, the 230-kV switchyard and the unit's startup transformer.

Also the following sources of power or startup transformers can be made available manually:

(1) One of the three gas turbines located 30 miles away at Lee Steam Station via an independent overhead 100-kV transmission system (this will be available for Unit 1 operation).

(2) Oconee Unit 2 or 3 startup transformers (as they become available) via the station's emergency startup bus; and

(3) Oconee Unit 2 or 3 main generators via the standby buses (as Units 2 and 3 are completed).

In evaluating these power sources, we have considered the gas turbine as a temporary substitute power source for use primarily during the periods when the hydro units are not available. The applicant has estimated these periods to be approximately 24 hours each year plus 4 days once every 10 years when the common penstock will be drained for inspection and maintenance. During these periods the gas turbine will be run at rated speed, with no load, and will be directly connected, through the Oconee 100-kV switchyard over the isolated line, to the standby buses for automatic selection in the event that the 230-kV power is lost.

As initially proposed, the three 600-volt motor control centers for engineered safety features would have received power via an automatic transfer device from two of the three 4160-volt engineered safety feature buses. On the basis of our review, we concluded that use of the automatic transfer feature would unnecessarily reduce the independence of redundant engineered safety feature equipment. In Amendment No. 17 the applicant eliminated the provisions for automatic transfer of loads between redundant engineered safety feature buses.

Each of the four distribution panels associated with the 125-Vdc Instrumentation and Control Power System for Unit 1 receives power via diode assemblies from either of two 125-V battery buses, one in the associated unit and one in another unit. The 125-V batteries for both Unit 1 and Unit 2 will be available prior to operation of Unit 1. Although these diode assemblies automatically connect the batteries to the redundant dc buses, we have concluded that use of these diodes is acceptable for the Oconee units because:

(1) the failure (open or short circuit) of a single diode would not result in a loss of power to any bus or load;

(2) diode monitors, that are capable of immediately detecting an open or shorted diode, are provided for each diode assembly; and

(3) although, if all overload devices were to fail to function, a single fault could result in the loss of power to one 120-Vac vital instrument bus, one 125-Vdc power panel and both battery buses that supply power to that dc panel, the loss of power to these buses and their loads would not reduce the capability of the protection system below that required to meet the minimum safety requirements of any unit.

We have concluded that the onsite power system is acceptable.

8.5 Cable and Equipment Separation and Fire Prevention

We have reviewed the applicant's design provisions and installation arrangement plans relating (1) to the preservation of the independence of redundant safety equipment by means of identification and separation, and (2) to the prevention of fires through derating of power cables and proper tray loading. We have found these design provisions and installation arrangements to be acceptable.

8.6 Accident Condition Testing

The applicant has listed the equipment and instrumentation that must be operable during and subsequent to an accident and has described the temperature, humidity, pressure, radiation, and seismic qualification tests performed on this equipment. We have reviewed this information and have concluded that the test program and its results are acceptable and that the essential instrumentation and controls will function properly in the accident environment.

9.0 CONTROL OF RADIOACTIVE EFFLUENTS

Liquid and gaseous waste handling facilities are designed to process waste fluids generated by the plant so that discharge of liquid and gaseous effluents to the environment will be minimized. Liquid waste is processed both by direct removal of radioactive material with ion exchange resins and by evaporative separation. Using these methods the volume of radioactive waste will be greatly reduced and the purified liquid streams will either be reused or discharged. Small quantities of radioactive liquid waste will be released routinely to the Keowee Hydro Station tail race where the waste will be diluted and discharged to the Keowee River.

The limits on routine radwaste releases from the three units that are planned for operation at the Oconee Nuclear Station will require that the combined releases from the three units when added together be within the limits specified in 10 CFR Part 20. The specific limits for both liquid and gaseous effluents are included in the Technical Specifications. Under normal operating conditions, however, it is expected that liquid waste releases will contain radioactivity in concentrations that are less than 1% of the 10 CFR Part 20 limits and that the concentrations in gaseous releases will be only a few percent of the 10 CFR Part 20 limits.

Liquid wastes are collected according to expected radioactivity content: wastes containing the highest activity are routed to the waste holdup tanks, intermediate activity wastes are routed to the high activity waste tanks, and low activity wastes are routed to the low activity waste tanks. Low activity wastes can also be present in the condensate test tanks (which, although not defined as a part of the waste disposal system, have been evaluated as such since they are a source of direct release of radioactivity to the plant radioactive waste discharge line).

In addition to holdup, other means are available to reduce the radioactivity in the liquid wastes before release. A waste evaporator and a coolant bleed evaporator are provided. These have the ability to remove radioactivity by evaporation, returning the distillate to the coolant bleed holdup tanks for reuse as reactor coolant makeup, and routing the concentrate, under appropriate conditions, to the solid waste drumming station for packaging as solid waste. Demineralizers also are provided in the coolant treatment system, and these can be used to remove radioactivity from liquid wastes prior to release.

Liquid waste releases are made on a batch basis. As a result of frequent operation of the onsite hydro-station, almost all liquid waste releases are expected to be mixed in a dilution flow substantially greater than the minimum 30 cubic feet per second dilution flow that would be available if the hydro station is not operating. In all cases, the radioactivity content of the waste is measured prior to release and monitored during release.

Oconee Station has been designed and built to minimize the possibility of an accidental release of liquid radioactive waste. The plant design includes the location of all liquid radioactive waste treatment system components below grade in Class I (seismic) structures. Therefore, in order for liquid radioactive wastes to be accidentally discharged, they must be inadvertently pumped to the environment. This pumping capability is controlled from the Unit 1 control room. Further, the radiation monitors on the liquid waste discharge line will terminate the discharge of radioactive liquids if the concentration in the discharge line when mixed with the minimum Keowee Hydro Plant flow (30 cubic feet per second) would exceed 10 CFR Part 20 limits. The Technical Specifications require that liquid wastes be discharged only if (1) concentrations within the limits of 10 CFR Part 20 can be achieved considering no more than the minimum 30 cubic feet per second dilution flow, and (2) the effluent line radiation monitors are operable. The Technical Specifications also require duplicate sampling and analyses of the contents of the low level waste tanks and the condensate test tank prior to initiating any liquid discharge from these tanks. We have, however, evaluated the consequences of a postulated accidental release of liquid waste resulting from a multiplicity of operator errors. We assumed that the contents of the low level waste tanks were inadvertently pumped to the Keowee hydro plant tailrace. This would result in radioactivity concentrations in the tailrace several times 10 CFR Part 20 limits, assuming a minimum dilution flow of 30 cubic feet per second in the tailrace. However, even if a person were to derive 1 day's supply of drinking water directly from the tailrace (the nearest drinking water supply is the Clemson intake 13.7 miles downstream) the resulting dose to the person would be a few percent of his allowable accumulated yearly limit. Because of additional dilution and the approximately 2.5 days required for water from the tailrace to reach the Clemson intake (allowing substantial decay) the resulting dose at that location would be further reduced. In addition, the Clemson water supply, which is owned by the Duke Power Company, is monitored for radioactivity and, if necessary, its use can be terminated for up to 1-1/2 days (storage capacity) to permit a further reduction in radioactivity entering the water supply.

Gaseous radioactive wastes, apart from steam generator or heat exchanger leakage, will be collected principally from the various liquid storage tanks associated with the reactor plant. All gaseous radioactive waste releases will be monitored during discharge. In addition, any release from the waste gas collection system or the reactor building will be analyzed for activity prior to release. The air ejector exhaust on the secondary system also is regularly monitored for activity to detect radioactivity releases that could occur as a result of steam generator leakage. Similarly, low pressure cooling water systems used to cool components containing reactor coolant are monitored regularly to detect radioactive in-leakage. The consequences of a rupture of a waste gas decay tank are noted in Section 11.0 of this evaluation.

No solid plant wastes will be permanently stored at the Oconee site and all solid wastes collected and temporarily kept at the site must be shipped offsite for ultimate disposal at an AEC licensed disposal site.

We have concluded that the radioactive waste system and the procedures for the control of radioactivity releases from Oconee Unit No. 1 are acceptable.

10.0 AUXILIARY SYSTEMS

The auxiliary systems necessary to assure safe plant shutdown include (1) the chemical addition and sampling system, (2) the high pressure injection system,(3) the component cooling system,(4) the low pressure injection system, (5) the service water system, and (6) the condenser circulating water system. The systems necessary to assure safe handling and adequate cooling for spent fuel include (1) the spent fuel cooling system, (2) the fuel handling systems, (3) the service water system, and (4) the recirculated cooling water system. The high and low pressure injection systems were evaluated as subsystems of the emergency core cooling system discussed in Section 7.1 of this evaluation. Our evaluation of the other auxiliary systems noted above is discussed below.

10.1 Chemical Addition and Sampling System

The boron content of the reactor coolant system serves as a principal means of reactivity control. The boron concentration is adjusted periodically to compensate for fuel burnup. The boron concentration must be reduced significantly when returning to power operation from a cold shutdown. The concentration is reduced by adding water to the reactor letdown storage tank. Dilution is automatically prohibited unless certain control rods are withdrawn to a preset position and an integrated flow timing device (set to add a predetermined, safe amount of dilution water to the letdown storage tank) has been activated.

The dilution flow entering the letdown storage tank is supplied to the reactor coolant system by the high pressure injection system at a maximum rate of 70 gpm. It is stopped automatically when the predetermined integrated flow time has elapsed or the compensating control rods have been inserted to a preset position. Also, whenever dilution is in progress, this is indicated to the reactor operator by lights on the control console. The consequences of accidental boron dilution have been evaluated by the applicant. We agree with the applicant that such incidents are highly unlikely and that even if they were to occur, they would not be expected to result in significant radiological consequences.

The presence of oxygen, chlorides, and fluorides in more than trace quantities will be detected by periodic sampling. The Technical Specifications require that timely corrective action be taken in the event that there is a significant increase in any of these unwanted impurities.

On the basis of our review we have concluded that the chemical addition and sampling system is acceptable.

10.2 Component Cooling System

The component cooling system provides cooling to the reactor coolant letdown coolers, the reactor coolant pumps, the control rod drives, and the pressurizer quench tank. It also serves as a barrier between the reactor coolant and the service water system in the event of a reactor coolant system leak into the component cooling system.

Only the isolation valves of this system would be required to function during accident conditions. Our evaluation of reactor building isolation is discussed in Section 7.3 of this evaluation. The complete loss of all cooling water flow during normal operation would not require an immediate reactor shutdown. However, the applicant has stated in the FSAR that procedures will require the operator to shut down the reactor under these conditions in order to protect the control rod drive coils.

On the basis of our review we have concluded that the component cooling system is acceptable.

10.3 Service Water System

The Class I (seismic) service water system consists of a low pressure service water (LPSW) system and a high pressure service water (HPSW) system. The station will have two LPSW systems.

One will be shared by Units 1 and 2 and the other, of almost identical design, will service Unit 3. The principal safety related use of the LPSW systems is to provide cooling to the low pressure injection and decay heat coolers outside containment and to the reactor building coolers inside containment. Each LPSW system takes its water supply from the condenser circulating water system through three 15,000 gpm pumps. Two pumps are supplied by one suction line, and the third pump by another suction line (Unit 3 will have only one pump per suction line). The HPSW system will also be available as a backup source at the LPSW system pump discharge. Low pressure service water is provided to the redundant low pressure injection coolers and the reactor building coolers through redundant supply lines. Two pumps are sufficient to provide all LPSW system performance requirements following a loss-of-coolant accident. The third pump provides protection against loss of a pump due to a single failure under accident conditions. All pumps are powered from the emergency power system.

A single high pressure service water system (HPSW) is provided primarily for fire protection services but this system could also function as a backup to the LPSW system. Water is provided to the HPSW system by two 6000 gpm pumps and one 500 gpm pump. One 6000 gpm pump is adequate for fire protection services. Manual isolation valves are provided so that water may be supplied to the system from any of the three condenser circulating water system inlet headers. In addition, there are 100,000 gallons of water stored in an elevated tank for use as a backup supply for the fire protection systems.

We have concluded that the HPSW and LPSW systems will provide all needed normal and emergency services and are acceptable. (See Section 7.2 of this evaluation for additional discussion of the LPSW system as related to the reactor building coolers).

10.4 Condenser Circulating Water System

The condenser circulating water system for Units 1, 2, and 3 using intake water from the Little River arm of Lake Keowee, receives essentially all waste heat from each unit and, under normal conditions, deposits it in Lake Keowee through the station discharge structure located in the Keowee River arm of the lake. Structures, piping, and equipment in this system that are essential to maintaining the unit in a safe operating or shutdown condition are of Class I (seismic) design. These include the intake canal weir and dike, the station intake structures, pumps, conduits, and cross-overs, normal and emergency discharge conduits and valves, and connections to other Class I (seismic) intermediate cooling systems.

This system takes on special safety significance during reactor shutdown under various normal and abnormal conditions. During normal reactor shutdown conditions, core decay heat must be removed to prevent overheating. There are two principal intermediate heat removal routes: (1) by way of the steam and power conversion system (steam generators and main condensers), and (2) by way of the low pressure injection and low pressure service water systems. The heat removal capacity of the steam and power conversion system route is adequate to permit the loss of the low pressure injection route. Redundancy within the steam conversion system is such that the heat removal adequacy of this system is not impaired by single failures of components, equipment, or piping.

The adequacy of decay heat cooling during reactor shutdown under abnormal conditions has been evaluated. In the event that all ac power is lost, all motor-driven pumps would be inoperative. To accommodate such conditions, the Oconee Station has been provided with a siphon-effect emergency discharge line from the unit condensers to the Keowee River. This line has an emergency discharge valve which opens automatically upon loss of power. As long as there is water in the intake canal (760 feet elevation at the bottom) the unit condensers will have adequate cooling capacity to remove core decay heat.

In the unlikely event that the water level in Lake Keowee should fall below 770 feet, an underwater weir in the intake canal would act as a dam capable of retaining a large amount of water to serve as an emergency cooling pond. By operator action, the condenser circulating water system normal discharge paths would be closed and an emergency discharge conduit, provided for this contingency, would be opened, permitting cooling by recirculation of the cooling pond water. The capacity of this cooling pond is adequate to provide core decay heat cooling indefinitely as long as electric power is available to run the condenser cooling pumps in the intake structures. In the extremely unlikely event that both the preferred and standby power sources were to become unavailable (loss of the 230 kV switchyard and the Hydro Station), emergency power can be provided to the station within approximately 30 minutes through an onsite 100 kV switchyard located at elevation 800 feet west of the reactor building. Since there would be enough condensate storage available in the steam and power conversion system to remove decay heat by venting steam to the atmosphere (instead of condensing it) for about 20 hours, the 30 minutes required to activate this reserve backup power source is acceptable.

On the basis of our review, we have concluded that the condenser circulating water system is acceptable.

10.5 Spent Fuel Cooling System

The spent fuel cooling system is provided with two circulation pumps and two heat exchangers. The heat exchangers reject heat to the recirculated cooling water system, which, in turn rejects its heat to the condenser circulating water system for discharge into Lake Keowee. The failure of one pump and one heat exchanger with 1-2/3 cores in storage could result in an increase in the temperature of the storage pool water to about 205° F over a long period of time. The loss of all pool cooling with 1-2/3 cores in storage could result in the attainment of this temperature in about 9 hours. Each pool is provided with diverse alarms (high temperature, low coolant flow, and low pool level). In the event of a complete loss of cooling, adequate time will be available to restore normal cooling and, using borated water from the borated water storage tank, to replenish any water lost through evaporation prior to any fuel damage.

We have concluded that the spent fuel cooling system is acceptable.

10.6 Spent Fuel Handling System

All spent fuel handling operations, prior to cask removal, will be performed under water. The spent fuel racks will be covered with a minimum of 23-1/2 feet of water and the fuel transfer canal will provide a minimum of 9-1/2 feet of water over fuel in transit. Working area radiation levels will be kept below 2.5 mR/hr.

This system has two basic sections, one inside the reactor building and one outside the reactor building. Inside the reactor building, fuel assemblies are removed from the reactor core one at a time, using the main fuel transfer handling bridge, and transported underwater to the fuel transfer station. Each assembly is then vertically placed either (temporarily) in one of four available positions in the fuel handling rack (spaced on 24 inch centers to prevent a critical configuration) or directly into one of the two fuel transfer baskets (the other basket is reserved for handling new fuel assemblies). Once in the basket the spent fuel assembly is rotated to a horizontal position and moved from the building to the fuel pool through the transfer tube. There are two fuel handling bridges inside containment. Fuel will not be handled by these two bridges at the same time with the exception that it will be permissible to simultaneously handle one fuel assembly over the core area (for relocation purposes) while the control rod or orifice rod of a spent fuel assembly is being transferred to a new fuel assembly which is still in the second fuel transfer basket.

The section located outside of the reactor building is housed in the fuel storage facility which will be shared with Unit 2. The roof and the walls of the facility plus the walls of the fuel pool itself are reinforced concrete and are designed to withstand the effects of missiles that might arise from a tornado or a turbine failure. This facility consists of a pool equipped with spent fuel storage racks, a fuel storage handling bridge and provisions for transferring the fuel between each reactor building and the pool. A fuel storage handling bridge will be used to maneuver the individual fuel assemblies one at a time during fuel handling operations.

At one end of the pool there is space for loading spent fuel into a shipping cask and directly over this area there is a 100 ton fuel storage building crane used to move shipping casks to the fuel loading area. The supports for this crane do not permit it to travel over the spent fuel storage rack area.

The pool itself is an integral part of a separate Class I (seismic) structure constructed of reinforced concrete. The pool is lined with 1/4-inch-thick stainless steel and rests on bedrock. At our request the applicant presented an analysis showing the consequences of dropping a loaded fuel cask into the pool. The analysis indicates that a fuel cask, if dropped from the highest possible elevation when attached to the 100-ton crane, would not strike the fuel stored in the spent fuel racks; it could crush some of the bedrock or filler concrete used to smooth the bedrock and thus deform the steel liner, but it would not cause the steel liner to lose its leak-tight integrity. Based on our review of this analysis we have concluded that the applicant has provided reasonable protection against water loss and against damage to fuel in the storage racks of the pool as the result of an accident involving dropping of a fuel cask.

The Technical Specifications prohibit the storage of irradiated fuel in the storage pool until the applicant either installs suitable filters to reduce the calculated potential radiological dose from the refueling accident (see Section 11.3) to well below the 10 CFR Part 100 guideline values, or submits additional information proving that the iodine retention capability of the pool water is sufficient to eliminate the need for the filters.

On the basis of our review and the conditions imposed by the Technical Specifications, we have concluded that the spent fuel handling system for Unit 1 is acceptable.

10.7 Recirculated Cooling Water System

Although the recirculated cooling water system does not penetrate the reactor building, it does come in contact with other systems containing radioactive or potentially radioactive fluids. For this reason, the return line to the system surge tank is monitored for radioactivity as it leaves the auxiliary building.

The failure of any one of the three valves in the closed position in the lines leading to and returning from the redundant spent fuel coolers can cause the loss of these coolers. However, since these valves can be repaired or replaced within a short period of time and spent fuel cooling can be lost for several hours without uncovering stored spent fuel, this is considered acceptable.

We have concluded that the recirculated cooling water system is acceptable.

11.0 ANALYSES OF RADIOLOGICAL CONSEQUENCES FROM DESIGN BASIS ACCIDENTS

11.1 General

In order to assess the safety margins of the plant design, a number of operating transients were considered by the applicant, including rod withdrawal during startup and at power, moderator dilution, loss of coolant flow, loss of electrical load, loss of ac power, and starting of a pump in an idle coolant loop. The reactor control and protection system is designed so that corrective action is taken automatically to cope with any of these transients. Based on our evaluation of the information submitted by the applicant and our evaluations of other PWR designs at the operating license stage, we conclude that the Oconee Unit 1 control and protection system design is such that these transients can be terminated without damage to the core or to the reactor coolant pressure boundary, and with no offsite radiological consequences.

The applicant and we have evaluated the consequences of potential accidents, including a control rod ejection accident, an accident involving rupture of a gas decay tank, a steamline break accident, a steam generator tube rupture accident, a loss-of-coolant accident, and a refueling accident.

The calculated offsite radiological doses that might result from the control rod ejection accident, the steam generator tube rupture accident, the steam line break accident, and the accident involving rupture of a gas decay tank are well within the 10 CFR Part 100 guidelines.

Our evaluations of the loss-of-coolant accident and the refueling accident are discussed in the following sections.

11.2 Loss-of-Coolant Accident

The design basis loss-of-coolant accident (LOCA) for Oconee Unit No. 1 is similar to that evaluated for other PWR plants in that a double-ended break in the largest pipe of the reactor coolant system is assumed. The AEC Safety Guide 4 - "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (published November 2, 1970) describes the general basis upon which our independent analysis of this accident was performed. We assumed that the reactor had been operating at a power level of 2568 MWt prior to the accident. Reactor building leakage is assumed to be constant at the Technical Specification limit of 0.25% of building volume per day for the first 24 hours and at 45% of that rate thereafter. The effects of radiological decay during holdup are considered and credit is given for the reduction of radioactive material in the leakage which reaches the atmosphere by way of the penetration room ventilation system filters. Fifty percent of all leakage is assumed to go through the penetration room filters which are considered 90% efficient in removal of iodine. The remaining 50% of the leakage is assumed to be released without treatment. No credit was assumed for reactor building spray system (boric acid solutions) for the removal of iodine.

For the calculation of the 2-hour dose at the site boundary we used an atmospheric dispersion factor corresponding to Pasquill Type "F" stability, with a 1 meter per second wind speed, a terrain correction factor of 2.2, and an appropriate building wake effect. We calculated the potential doses at the site boundary for a 2-hour period to be 190 Rem to the thyroid and 2 Rem to the whole body. At the low population zone boundary our calculated potential doses for a 30-day period are 200 Rem to the thyroid and 1 Rem to the whole body.

11.3 Fuel Handling Accident

We have evaluated the potential consequences of a fuel handling accident, in which it is postulated that a fuel assembly is dropped in the spent fuel pool or transfer canal. We assumed that: (1) all 208 rods in the dropped bundle are damaged, (2) the accident occurs 72 hours after shutdown of the core (2568 MWt) from which the dropped bundle has been removed, (3) 20% of the noble gases and 10% of the iodine in the dropped fuel bundle are released to the refueling water, (4) the dropped fuel bundle has been removed from a region of the core which has been generating 1.68 times the average core power, (5) 90% of the released iodine is retained in the refueling water, (6) the fission products released from the pool are discharged to the atmosphere at ground level, and (7) the same meteorological conditions exist as were assumed for the loss-of-coolant accident. The resultant calculated 2-hour doses at the site boundary are 300 Rem to the thyroid and less than 1 Rem to the whole body.

Because the probability of occurrence of a fuel handling accident is relatively greater than the probability of the occurence of other design basis accidents, we have informed the applicant that means should be provided to reduce the calculated potential doses for the fuel handling accident to well within the guideline values given in 10 CFR Part 100.

The applicant contends that evidence can be developed to demonstrate that the pool water will retain much more of the iodine released to it than the 90% that we assume in our analysis. In Amendment No. 17, the applicant stated that suitable filters will be installed in the spent fuel building ventilation system for the removal of iodine, prior to the movement of irradiated fuel into the spent fuel pool, unless the needed justification for the use of higher pool water partition factors is submitted and determined to be acceptable by the staff.

The Technical Specifications prohibit removal of irradiated fuel from the reactor building until the applicant has taken measures to reduce the calculated doses from a refueling accident to values well within the 10 CFR Part 100 guideline dose levels. We conclude that there is ample time for the applicant to install filters or obtain approval of a higher partition factor prior to the first scheduled refueling period for Unit 1.

11.4 Conclusions

We have calculated offsite doses for the design basis accidents that have the greatest potential for offsite consequences using assumptions consistent with those we have used in previous safety reviews of PWR plants and have found the resulting calculated doses to be less than the guideline values of 10 CFR Part 100.

12.0 CONDUCT OF OPERATIONS

12.1 Technical Qualifications

The Duke Power Company has extensive experience in the design, construction, and operation of electric generating plants. The applicant's involvement in nuclear power generation extends back to the early 1950's. Key personnel from all departments associated with the design, construction, and operation of power generating facilities have participated in the nuclear training programs and have contributed to development of nuclear standards in the industry. Duke officers and technical personnel including the Oconee Station plant superintendent, participated in the 5-year operating program at the Carolinas-Virginia Test Reactor (CVTR) in the areas of planning, management, training, research, engineering, and operations.

In addition to the three Oconee Units, the applicant has applied to the AEC for construction permits on two pressurized water reactors for its McGuire Station on Lake Norman.

On the basis of our review of the applicant's engineering, construction, and operating organization we conclude that the applicant has demonstrated adequate technical competence to complete construction, conduct necessary preoperational testing, and operate Unit 1.

12.2 Operating Organization and Training

The minimum qualifications for key personnel in the applicant's operating organization for Unit 1 are in general agreement with ANSI 18.1 "Proposed Standard for Selection and Training of Personnel for Nuclear Power Plants" and are acceptable.

The applicant's operating organization consists of three main groups under the direction of a station superintendent. These groups are (1) an operations group, headed by a graduate engineer having responsibility for the operating personnel; (2) a technical support group headed by a graduate engineer having responsibility for station chemistry, health physics, instrumentation and control maintenance, and station performance surveillance; (3) a general maintenance group having responsibility for maintenance of mechanical and electrical equipment.

The proposed shift complement for operation of Unit 1 consists of one shift supervisor licensed as a senior reactor operator, one control operator licensed as a reactor operator, one assistant control operator liscensed as a reactor operator and two utility operators. After significant experience with power operation of the Oconee Station has been obtained, the applicant will be given an opportunity to justify a four-man shift crew; however for initial operation the Technical Specifications require five-man shifts for licensed operation.

The training program outlined for the operating staff is a conventional program provided by the applicant and the nuclear steam supply system vendor, supplemented by assignment of key personnel to the Saxton reactor facility to gain operating experience. We have concluded that the training program is acceptable.

Review and audit of station operations, maintenance and technical matters will be performed by two committees, a Station Review Committee and a General Office Review Committee.

The General Office Review Committee is composed of a chairman, three members from the Steam Production Department (including the Oconee Station Superintendent) and three members from the Engineering Department, all appointed by the Executive Vice President for Power Operations. The committee may be expanded to include outside consultants when necessary or desirable. It is the responsibility of this committee to review all procedures, procedure changes, design changes, and abnormal occurences that may affect public health and safety. In accomplishing this task, the committee is charged with auditing station records, logs, reports, tests, minutes of the Station Review Committee meetings, and making written recommendations to the appropriate Vice President with a copy of such recommendations also being sent to the Executive Vice President for Power Operations. Records of all meetings will be kept on file at the station.

The Station Review Committee is composed of the Assistant Superintendent (designated Chairman) the Operating-Engineer, Technical Support Engineer and at least two other station supervisory personnel. The Committee meets monthly and at the call of the Chairman and is charged with reviewing all new procedures and prcedure changes, proposed tests, proposed station design changes, and abnormal occurences. The Committee also reviews station operation for matters of potential safety significance. The Committee is charged with keeping minutes of all meetings and distributing a copy of these minutes to the Station Superintendent, the Manager of Steam Production and to the Chairman of the General Office Review Committee (discussed above). Findings of this Committee are forwarded to the Station Superintendent for appropriate action.

Preoperational testing of equipment and systems at the site and initial plant operation will be performed by the applicant's personnel with technical support from the B&W Nuclear Power Generating Division's engineers.

We conclude that the applicant's organization is acceptably staffed and technically qualified to perform its operational duties subject to satisfactory completion of licensing examinations of personnel requiring licenses (see 10 CFR Part 55).

12.3 Emergency Planning

The applicant has prepared an Oconee Station emergency plan for dealing with incidents that might involve releases of radioactivity. The plan considers a broad spectrum of accidents that could affect both onsite personnel and the public in unrestricted areas. The emergency plan provides for the shift supervisor to be in direct charge of all emergency operations and to act as emergency coordinator until specifically provided responsible relief by the Station Superintendent. Under this arrangement the shift supervisor will be responsible for protection of other plant personnel, take necessary onsite remedial action to terminate the incident, establish access control to the affected areas, collect preliminary data, obtain necessary outside aid and notify management.

Reliable means of communication are provided within the station by telephone between the control room, various parts of the plant, the Visitors' Center and the Keowee Hydro Plant, and by an onsite public address system. Communications outside the plant include the telephone, microwave communications with several other of the applicant's facilities; and two-way radio communications among (1) the control room, (2) a Duke Power substation at Central, South Carolina, (3) an emergency vehicle, and (4) a boat.

Continuous wind speed and direction data are telemetered to the station control room. The supervisor also has available in the control room information (e.g., reactor building pressure, temperature and radiation levels) that can be used to evaluate the magnitude of a potential accident. Additional emergency instruments and equipment will be available.

In the event of an emergency that involves areas beyond the jurisdiction of the applicant, arrangements have been made to establish an Emergency Control Center in Walhalla, South Carolina to obtain the assistance of local, State, and Federal agencies. The support groups will, if necessary, establish road blocks, perform radiation monitoring work, and institute other applicable protective measures.

As the various agencies responsible for the public health and safety respond and the Emergency Control Center becomes operable, responsibility for protection of the general public will be transferred from the Shift Supervisor to the Emergency Control Center with the Shift Supervisor remaining responsible for the protection of onsite personnel and station property.

Provisions have been made for medical support including, if required, treatment of radiation-contaminated patients. These include a first aid room within the restricted area of the station and space at the Oconee Memorial Hospital in Seneca, South Carolina. Plant personnel will be trained in first aid procedures and in methods of decontaminating injured personnel. The hospital staff has been trained in radiological health and contamination control. A physician at Memorial Clinic in Seneca, South Carolina, serving as the company doctor for the Oconee Nuclear Station, has been trained, at an AEC sponsored seminar at Brookhaven National Laboratory, in medical planning and care in radiation accidents.

We conclude that the applicant's emergency plan conforms to the requirements for emergency plans as presented in the proposed change to 10 CFR Part 50.34 of the Commission's regulations and is acceptable.

12.4 Industrial Security

Provisions for industrial security described by the applicant in Amendment No. 11 include perimeter fencing, gate and door access control and a closed-circuit television system coupled with a remote control lock system for off-hour identification and admission of personnel to the facility. Appropriate plans have been developed to control access to Unit 1 of construction personnel working on the units still under construction. We have concluded that the applicant has taken reasonable measures to provide for the security of the facility.

13.0 TECHNICAL SPECIFICATIONS

The Technical Specifications in an operating license define safety limits and limiting safety system settings, limiting conditions for operation, periodic surveillance requirements, certain design features, and administrative controls for the operating plant. These specifications cannot be changed without prior approval of the AEC. The applicant's proposed Technical Specifications have been modified, in Amendment No. 24, as a result of our review, to describe more definitively the allowable conditions for plant operation. The Technical Specifications, as approved by the regulatory staff, will be available for examination in the Commission's Public Document Room.

Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures in excess of 10 CFR Part 20 limits and that means are provided for keeping the release of radioactivity from the plant within ranges that we consider as low as practicable. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features to mitigate the consequences of unlikely accidents will be available.

14.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS reported on the application for construction of the Oconee Nuclear Station at the proposed site in a letter dated July 11, 1967. The applicant has been responsive to the recommendations made by the ACRS in that letter, and we conclude that the matters raised have been resolved satisfactorily during the design and construction of the Oconee Nuclear Station Unit 1.

The ACRS completed its review of the application for an operating license for Oconee Unit 1 at its 125th meeting held September 17-19, 1970. A copy of the ACRS letter, dated September 23, 1970, is attached as Appendix B.

In its letter the ACRS made several recommendations and noted several items all of which have been considered in the individual sections of this evaluation. These include: operation of Unit 1 while Units 2 and 3 are under construction (Section 12.4); regulatory review of reactor operation prior to operation at 2568 MWt (Section 1.0); diverse means of reactor trip for use with emergency core cooling system (Section 8.2); continued use of thermocouples in the reactor core (Section 4.2); normal radioactive waste release management (Section 9.0): decision on matter of filters in fuel pool building exhaust system (Section 10.6); efficacy of emergency core cooling system (Section 7.1); tests for absence of divergent azimuthal xenon oscillations (Section 4.2); reactor pressure vessel fracture toughness (Section 5.2); studies of means for preventing common mode failures and of design features to mitigate the consequences of failure to scram during anticipated transients (Section 8.1): studies of means to detect primary system loose parts, displacements, and changes in vibration characteristics (Section 5.6); and the effects of hydrogen evolution and control on the health and safety of the public (Section 7.4).

Apart from those recommendations requiring further advances in the technology, which will be resolved as suitable approaches are developed, the applicant has resolved the ACRS items.

The ACRS concluded in its letter that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, the Oconee Nuclear Station Unit 1 can be operated at core power levels up to 2568 MWt without undue risk to the health and safety of the public.

15.0 COMMON DEFENSE AND SECURITY

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

16.0 FINANCIAL QUALIFICATIONS

The Commission's regulations that relate to the financial data and information required to establish financial qualifications for an applicant for operating licenses are 10 CFR Part 50.33(f) and 10 CFR Part 50 Appendix C. Duke Power Company's application as amended by Amendment No. 22 thereto, and the accompanying certified annual financial statements provided the financial information required by the Commission's regulations.

These submittals contain the estimated annual operating costs for the first 5 years of operation plus the estimated cost of permanently shutting down Oconee Unit 1 and maintaining it in a safe condition, if and when it may become necessary. The estimated operating costs are $10,025,442 per year for the 5-year period. Such costs include costs of operation, maintenance, fuel, ad valorem taxes and insurance. The applicant's estimate of the cost of permanently shutting down the facility and maintaining it in a safe condition is $500,000, based on the reactor (with fuel removed) and associated nuclear systems remaining in place, isolated by fencing and monitored periodically by guards. The funds for these estimated costs would be obtained from the electrical operating revenues derived from system-wide operation of Duke Power Company (Amendment No. 22).

We have examined the certified financial statements of Duke Power Company to determine whether the Company is financially qualified to meet these estimated costs. The information contained in the 1969 Annual Report indicates that electric operating revenues for 1969 totaled $342.2 million; operating expenses (including taxes) were $266.1 million; the interest on the long-term debt was earned 2.7 times: and the net income for the year was $54.4 million, of which $39.4 million was distributed as dividends to the stockholders and $14.2 million (net) was retained for use in the business.

As of December 31, 1969, the Company's assets totaled $1,399 million, most of which was invested in electric plants ($1,284.1 million) and earnings reinvested in the business totaled $66.9 million. Financial ratios computed from the 1969 financial statements indicate a sound financial condition; e.g., long-term debt to total capitalization -0.55, and to net utility plant -0.52: net plant to capitalization - 1.07; the operating ratio -0.78; and the rates of return on common - 12.3%, on stockholders' investment - 10.1% and on total investment - 5.6%. The record of the Company's operations over the past 5 years reflects that operating revenues increased from $234.4 million in 1965 to $342.2 million in 1969 or 46%; net income increased from $40.8 million to $54.4 million or 33%; and net investment in utility plant from $711.2 million to $1,284 million or 81%. Moody's Investors Service rates the Company's first mortgage bonds as Aa (high grade) and its debentures as A (higher medium grade). The Company's Dun and Bradstreet Credit Rating is AaA1.

Our evaluation of the financial data submitted by the applicant, summarized above, provides reasonable assurance that the applicant possesses or can obtain the necessary funds to meet the requirements of 10 CFR 50.33(f) with respect to operation of Oconee Unit No. 1. A copy of the staff's financial analysis is attached as Appendix G.

17.0 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, licensees for facilities such as power reactors under 10 CFR Part 50.

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

Duke Power Company has furnished to the Commission proof of financial protection in the amount of $1,000,000, in the form of a Nuclear Energy Liability Insurance Association policy (Nuclear Energy Liability Policy, facility form) No. NF-182.

Further, Duke Power Company executed Indemnity Agreement No. B-44 with the Commission as of March 24, 1970, the effective date of its preoperational fuel storage license, SNM-1180.
Duke Power Company has paid the annual indemnity fee applicable to preoperational fuel storage.

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for the Oconee Nuclear Station, Unit 1 reactor (which has a rated capacity of more than 100,000 electrical kilowatts) is the maximum amount available from private sources, i.e., the combined capacity of the two nuclear liability insurance pools, which amount is currently $82 million. Accordingly, no license authorizing operation of the Oconee Nuclear Station Unit 1 will be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing, on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation.

Similarly, no operating license will be issued until an appropriate amendment to the present indemnity agreement has been executed. Duke Power Company will be required to pay an annual fee for operating license indemnity as provided in our regulations, at the rate of $30 per each thousand kilowatts of thermal capacity authorized in its operating license.

On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating 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 as to execution of an appropriate indemnity agreement with the Commission.

18.0 CONCLUSIONS

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

1. The application for facility license filed by the Duke Power Company, dated November 28, 1966, as amended (Amendments Nos. 1 through 24) 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 the Oconee Nuclear Station Unit 1 (the facility) has proceeded and there is reasonable assurance that it will be completed, in conformity with Provisional Construction Permit No. CPPR-33, 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 operating license, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1; and
6. The applicable provisions of 10 CFR Part 140 have been satisfied; and
7. 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 Duke Power Company for Oconee Nuclear Station Unit 1, the facility 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 Division of Compliance prior to license issuance.

APPENDIX A

CHRONOLOGY OF

REGULATORY REVIEW OF THE DUKE POWER COMPANY

OCONEE NUCLEAR STATION UNIT NOS. 1, 2, AND 3

SUBSEQUENT TO CONSTRUCTION PERMIT NOS. CPPR-33, 34- AND 35

ISSUED NOVEMBER 6, 1967

	Date	
1.	June 2, 1969	FSAR submitted as Amendment 7 to Duke's Application for licenses for the Oconee Nuclear Station.
2.	August 5, 1969	Meeting with Duke to discuss general aspects of our review.
3.	August 13-14, 1969	Visit to Duke engineering offices and Oconee construction site with our seismic design consultant to discuss status of design and observe construction progress at the site.
4.	September 15, 1969	Application Amendment 8 submitted, providing information on quality assurance (QA) and piping system classification and incorporating seven B&W Topical Reports by reference.
5.	September 18, 1969	Meeting with Duke on QA.
6.	September 24, 1969	Meeting with Duke to discuss thermal hydraulics design.
7.	November 17-18, 1969	Meeting with Duke to discuss site instrumentation, electrical systems, reactor physics, steam generators and vent valves, conduct of operations and initial tests.
8.	November 28, 1969	AEC-DRL letter to Duke requesting information on sustained DNB analysis as covered in B&W Topical Report BAW-10014.

9.	January 21-22, 1970	Meeting with Duke to discuss B&W Topical Reports, preoperational testing and electrical penetrations.
10.	February 9, 1970	Application Amendment 9 submitted, providing information in response to our QA review and incorporating two B&W Topical Reports by reference.
11.	February 13, 1970	AEC-DRL letter to Duke requesting additional information to continue review (major question list).
12.	February 19, 1970	Site visit with our meteorology consultant to witness gas diffusion testing and discuss Duke's efforts to resolve meteorology problems.
13.	March 3, 1970	AEC-DRL letter to Duke requesting information on core internals and accident analyses.
14.	March 16, 1970	Application Amendment 10 submitted, providing the final stress analysis report on the reactor coolant system.
15.	March 19-20, 1970	Meeting with Duke at B&W, Lynchburg, Virginia facility to discuss thermal hydraulic design analyses.
16.	March 27, 1970	AEC-DRL letter to Duke requesting information on thermal hydraulic design codes.
17.	March 30-31, 1970	Meeting with Duke at B&W, Lynchburg, Virginia facility to discuss core internals design analyses.
18.	April 2, 1970	Meeting with Duke to discuss instrumentation and electrical system drawings.
19.	April 3, 1970	Meeting with Duke at Bechtel's Gaithersburg, Maryland facility to discuss structural and piping design analyses.
20.	April 15, 1970	AEC-DRL letter to Duke requesting additional information on accident analyses.

21.	April 20, 1970	Application Amendment 11 submitted, providing response to our letters of November 28, 1969 and February 13, 1970, and incorporating one B&W Topical Report by reference.
22.	April 22, 1970	AEC-DRL letter to Duke requesting information on core internals.
23.	May 25, 1970	Application Amendment 12 submitted, providing responses to our letters and incorporating two B&W Topical Reports by reference.
24.	June 18, 1970	Meeting with Duke on Technical Specifications.
25.	June 22, 1970	Application Amendment 13 submitted, providing responses to AEC requests for information and incorporating three B&W Topical Reports by reference.
26.	June 23, 1970	ACRS Subcommittee meeting and site visit.
27.	June 25, 1970	Meeting with Duke to inform B&W in detail of our concerns with reference to potential deficiencies in ECCS analyses.
28.	July 9, 1970	Application Amendment 14 submitted, providing by reference a Duke report containing proprietary answers to our letters of February 13, 1970 and March 3, 1970. Also incorporated,by reference, four B&W Topical Reports.
29.	July 9, 1970	Application Amendment 15 submitted, providing answers to several of our letters including instrumentation qualification tests and meteorology measurements.
30.	July 15, 1970	AEC-DRL letter to Duke requesting additional information on predicted core performance during LOCA conditions.

31.	July 23, 1970	Application Amendment No. 16 submitted, including corrections to FSAR reflecting lower Unit 1 leak rate, improvement in penetration room filter system and proposing a restricted area boundary containing residences and members of the general public.
32.	July 30, 1970	Meeting with Duke to discuss partial-loop operation.
33.	July 31, 1970	ACRS Subcommittee meeting.
34.	August 11, 1970	Application Amendment No. 17 submitted, including Unit No. 1 reactor coolant pump substitution, change to pressurized fuel, and answers to several outstanding questions.
35.	August 14, 1970	Meeting with Duke to discuss outstanding review areas.
36.	August 14, 1970	ACRS full committee meeting to discuss technical aspects of design.
37.	August 19-20, 1970	Meeting at Idaho Nuclear Corporation to discuss RELAP-3 LOCA analysis.
38.	August 21, 1970	Meeting with Duke to discuss reactor internals vibration program.
39.	August 28, 1970	Application Amendment No. 18 submitted, including information on pressurized fuel, and answers to other outstanding concerns.
40.	September 4, 1970	Application Amendment No. 19 submitted including information on power imbalance reactor protection, additional reactor internals vibration monitoring capability and on reactor building pressure analysis.
41.	September 9, 1970	ACRS Subcommittee meeting.

42.	September 14, 1970	Application Amendment No. 20 submitted, including information on additional feature to assure core flooding tank availability, on fuel surveillance, on partial flow operation, on reactor building pressure analysis, loss-of-coolant accident analysis, pump replacement analysis, reactor internals instrument additions, and on additional reactor coolant pipe restraints.
43.	September 17, 1970	Application Amendment No. 21 submitted including information on additional analysis of loss-of-coolant accidents.
44.	September 18, 1970	ACRS full committee meeting to discuss technical aspects of design.
45.	September 23, 1970	ACRS letter to the Chairman of the AEC on review of Oconee Unit 1.
46.	October 1, 1970	AEC-DRL letter to Duke requesting Financial data.
47.	October 22, 1970	Application Amendment No. 22 submitted providing requested financial data.
48.	October 29, 1970	Meeting with Duke on Technical Specifications.
49.	November 19, 1970	Meeting with Duke on Technical Specifications.
50.	November 19, 1970	Application Amendment No. 23 submitted including responses to concerns expressed in ACRS letter dated September 23, 1970.
51.	December 14, 1970	Application Amendment No. 24 submitting revised proposed Technical Specifications.

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 23, 1970

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON OCONEE NUCLEAR STATION UNIT NO. 1

Dear Dr. Seaborg:

During its 125th meeting, September 17-19, 1970, the Advisory Committee on Reactor Safeguards completed its review of the application of the Duke Power Company for a license to operate Unit 1 of the Oconee Nuclear Station at power levels up to 2568 MW(t). The Committee met with the applicant during its 124th meeting, August 13-15, 1970 and Subcommittee meetings were held on June 23, 1970, at the site and on July 31, 1970 and September 9, 1970, in Washington, D. C. In the course of the review, the Committee had the benefit of discussions with representatives and consultants of the applicant, the Babcock and Wilcox Company, the Bechtel Corporation, and the AEC Regulatory Staff, and of study of the documents listed.

The Oconee Station is located in a rural area of Oconee County, South Carolina. The nearest population center is Anderson, 21 miles south, with a population of about 41,000. The minimum exclusion distance for the completed three-unit power station will be one mile and the Low Population Zone radius will be six miles containing about 3,400 people. The water supply for the plant is taken from Lake Keowee which was created by the applicant. The lake and associated recreational facilities are expected to attract a transient population to the area.

The application covers Oconee Units 1, 2, and 3, but this report applies only to Unit 1, which will employ the first of the Babcock and Wilcox two-loop, four-pump, pressurized water reactor, nuclear steam supply systems. The three units are designed to be nearly identical, but some facilities and services are shared in various arrangements. The Committee has reviewed the temporary arrangements necessitated by operation of Unit 1 while Units 2 and 3 are still under construction. It is believed that the proposed physical measures and administrative procedures to isolate the operating unit from construction activities are adequate.

Rec'd Off. Dir. of Reg.
Date 9/2...
Time ...

DR-2812

The Committee reported to you on the construction permit application for this power station on July 11, 1967. At that time the proposed operating power was to have been 2452 MW(t); the current proposal for operating at powers as high as 2568 MW(t) is justified by the applicant, primarily on the basis of a flatter power distribution. Prior to operation at the higher power level, reactor operation should be reviewed by the Regulatory Staff.

The prestressed concrete containment building is similar to those for the Palisades and Point Beach plants which have been reviewed recently for operation.

The Committee recommends that the applicant accelerate his studies of means of preventing common failure modes from negating scram action and of design features to make tolerable the consequences of failure to scram when required during anticipated transients. As solutions develop and are evaluated by the Regulatory Staff, appropriate action should be proposed and taken by the applicant on a reasonable time scale. The Committee wishes to be kept informed.

The applicant has proposed using a power-to-flow ratio signal as a diverse means to cause shutdown of the reactor if emergency core cooling action should be initiated. The Committee believes it is necessary that either the equipment associated with this signal be demonstrated to be able to survive the accident environment for an adequate time or a different, diverse trip signal be employed. This matter should be resolved to the satisfaction of the Regulatory Staff.

The Committee suggests that developmental techniques, such as neutron noise analysis and use of accelerometers, be considered as an aid in ascertaining displacements, changes in vibration characteristics, and the presence of loose parts in the primary systems. The Committee notes the desirability of the continuing use of some thermocouples in the core.

The Committee has commented in previous reports on the development of systems to control the buildup of hydrogen in the containment which might follow in the unlikely event of a loss-of-coolant accident. The applicant proposes to make use of a purging technique after a suitable time delay subsequent to the accident. Relatively high off-site doses possibly could result following purging of the containment. The Committee recommends that purging systems be incorporated in the plant but that the primary protection in this regard should utilize a hydrogen control method which keeps the hydrogen concentration within safe limits by means other than purging. The hydrogen control system and provisions for containment atmosphere mixing and sampling should have redundancy and instrumentation suitable for an engineered safety feature; these should be made available within the first two years of power operation. The Committee wishes to be kept informed of the resolution of this matter.

The applicant stated that the amount of radioactivity in liquid wastes normally will not be greater than one percent of 10 CFR Part 20 limiting concentrations after dilution with the minimum flow (30 cfs) below the Keowee dam. Larger flows will have proportionately smaller limiting concentrations. The mean annual discharge from the Keowee dam is expected to be 1,100 cu. ft./sec. The off-gas system has holding tank and filtering capability and gas release rates are not expected to exceed a few percent of 10 CFR Part 20 limits.

In order to protect against the postulated consequences of the accidental dropping of a fuel element, the applicant has stated that either, he will install filters in the fuel pool building exhaust system, or the equivalent control and protection will be assured by another method. This matter should be resolved to the satisfaction of the Regulatory Staff within the first year of power operation.

Improved calculational techniques are being applied to the analysis of the efficacy of the emergency core cooling system in the unlikely event of a loss-of-coolant accident. Interim results appear to be acceptable, but further calculations are needed and some phenomena important to the course of the accident require further study. This matter should be resolved in a manner satisfactory to the Regulatory Staff prior to operation at power. The Committee wishes to be kept informed.

The reactor is calculated to have a positive moderator coefficient of reactivity at power which will become negative as boron is removed from the coolant concurrent with build-up of fission products and fuel burnup. The applicant plans to perform tests to verify that divergent azimuthal xenon oscillations cannot occur in this reactor. The Committee recommends that the Regulatory Staff follow the measurements and analyses related to these tests.

A conservative method of defining pressure vessel fracture toughness should be employed that is satisfactory to the Regulatory Staff.

Other problems relating to large water reactors which have been identified by the Regulatory Staff and the ACRS and cited in previous reports to you should be dealt with appropriately by the Staff and applicant in the Oconee Unit 1 power plant as suitable approaches are developed.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing there is reasonable assurance the Oconee Nuclear Plant Unit 1 can be operated at power levels up to 2568 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Original Signed by
Joseph M. Hendrie

Joseph M. Hendrie
Chairman

Additional comments by Dr. W. R. Stratton are presented below:

"The high off-site doses which are stated to accompany the proposed purging operation are based on calculations which include a number of assumptions which I believe to be overly conservative. It is my opinion that the situation, should it ever arise, would be much less severe and that the proposed purge system would provide adequate protection for the health and safety of the public in this regard and therefore the additional hydrogen control equipment required by this letter is not necessary."

Attachment: List of References

References:

1. Amendment No. 7 to Duke Power Company Application for Oconee Nuclear Station, Units 1, 2, and 3, consisting of Final Safety Analysis Report, Volumes I and II, received June 4, 1969

2. Amendments Nos. 8 through 21 and Revised Amendment No. 13 to the License Application.

APPENDIX C

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

AUG

Peter A. Morris, Director
Division of Reactor Licensing

SAFETY ANALYSIS REPORT

Reference is made to the letter of July 29, 1970, from R. C. DeYoung, Assistant Director for Pressurized Water Reactors, DRL, to the Environmental Science Services Administration requesting comments on the following safety analysis report:

Oconee Nuclear Stations Units 1, 2, and 3
Duke Power Company
Final Safety Analysis Report
Amendment No. 15 dated July 9, 1970 and
Amendment No. 16 dated July 23, 1970

Review by the Air Resource Environmental Laboratory, ESSA, has now been completed and their comments are enclosed.

E. E. Sinclair, acting

Milton Shaw, Director
Division of Reactor Development
and Technology

Enclosure:
Comments (Orig. & 1 cy.)

cc: R. C. DeYoung, Assistant Director for Pressurized Water Reactors, DRL
H. L. Price, Director, REG

(95)

APPENDIX C

Comments on

Oconee Nuclear Station Units 1, 2, and 3
Duke Power Company
Final Safety Analysis Report
Amendment No. 15 dated July 9, 1970 and
Amendment No. 16 Dated July 23, 1970

Prepared by

Air Resources Environmental Laboratory
Environmental Science Services Administration
July 29, 1970

The fifteen gas tracer experiments conducted on the site under poor (inversion) conditions show that in all cases the centerline concentration was lower than that which would have been predicted by the use of the equivalent Pasquill Type diffusion rates. The closest agreement for a stability category of Type F was in test 1 b (see Table 2A-2) where at a distance of 680 m a centerline concentration of 4×10^{-5} sec m^{-3} was measured at a wind speed of 5.4 m/sec. The equivalent estimated concentration for Pasquill Type F would be 2.4×10^{-4} sec m^{-3}, a factor of 6 higher than the measured value. It must, however, be assumed that a building wake effect is part of the cause for this difference. Allowing a cA factor of 1270 m^2 for this effect brings the estimated value to a factor of 2.2 higher than the measured value.

An examination of the joint frequency tabulation of wind at the top of the 150-ft tower under slightly stable and moderately to strongly stable conditions shows a cumulative frequency of 9 percent for a diffusion rate equal to or worse than Pasquill Type F and 1.5 m/sec. Extrapolating the data to the 5 percent level of probability results in a diffusion rate equivalent to Type F and 1 m/sec.

Although a wind speed calibration check made in October 1969 indicated the speed to be reading low by a factor of 1.4, there is no rigorous way to determine how long this situation has persisted and to what extent the data in the joint frequency tables of speed and temperature lapse rate were affected.

It was obvious from a site visit by AEC and ESSA personnel in February 1970 that the terrain within the site boundary is very complicated and that it is difficult to make near-surface measurements which would be representative of the general flow of air in the area of the reactor complex. Because of the wooded nature of the terrain it was felt that the measurement above tree-top level at the top of the 150-ft micro-wave tower would most nearly represent the ambient flow from the reactor complex, although speeds would be somewhat over-estimated with regard to near-surface conditions. However, it could well be that this overestimation is compensated for by the under-estimation due to calibration errors.

In summary, for the short-term release (0-2 hours) it appears from the data presented that at the site boundary of 1.6 km, assuming an effective ground release, the use of Pasquill F diffusion, a 1 m/sec wind speed, a factor of 2.2 better diffusion because of site characteristics quantitatively shown by onsite diffusion experiments, and a cA factor of 1270 m^2 because of building wake effect is appropriately conservative. The resulting concentration would be approximately 1×10^{-4} sec m^{-3}.

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

JUN 1 9 1970

Mr. Harold Price
Director of Regulation
U.S. Atomic Energy Commission
7920 Norfolk Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to a request by Mr. Roger S. Boyd, is a review of the hydrologic aspects of the Oconee Nuclear Station Unit Nos. 1, 2 and 3 - AEC Docket Nos. 50-269, 270, and 278 - proposed by the Duke Power Company. 287

This review was prepared by P. J. Carpenter and has been discussed with members of your staff. We have no objections to your making this review a part of the public record.

Sincerely yours,

E. C. Reedman

Acting Director

Enclosure

cc: Walter G. Belter, AEC

Duke Power Company
Oconee Nuclear Station
Units 1, 2, and 3

AEC Docket Numbers 50-269, 270, and 287

The Oconee Nuclear Station is located on the eastern shore of Lake Keowee, approximately 8 miles northeast of Seneca, Oconee County, South Carolina. Each unit will use a pressurized-water reactor with an ultimate power output rated at 2,584 megawatts thermal or 866 megawatts electrical.

A copy of the hydrology review (dated June 9, 1967) prepared at the construction license stage by E. L. Meyer is attached. This hydrology review is in regard to the application for an operating license and is based on a review of the "Preliminary" and "Final Safety Analysis Reports" and an independent check of available data and literature. Because no additional hydrologic data or analyses are presented in the "Final Safety Analysis Report" comments made in the earlier review are still applicable and comments made below will correct and supplement them.

The plant grade is established at an elevation of 796 feet above mean sea level. Because the plant grade is separated from Lake Keowee by a topographic ridge varying from 25 to 150 feet high, and is some 130 feet above the flood pool of Hartwell Reservoir (immediately downstream of the site), flooding of the site would be unlikely.

Water for the once-through condenser cooling of all three units will be taken from Lake Keowee at an approximate rate of 4,700 cubic feet per second. The cooling water will be released immediately upstream from Lake Keowee Dam and the Keowee Hydroelectric Station owned by Duke Power Company. The applicant states that a submerged weir placed in the intake canal will provide an emergency cooling pond with adequate storage for safe shutdown of the plant in the event of loss of water level in Lake Keowee. Lake Keowee is formed by dams on the Little and Keowee Rivers. If the water level were lost in the Lake by failure of either dam, the Keowee and Little Rivers would be separated by a topographic ridge and the natural flow of Little River only would then pass the intake. Based on streamflow records collected at surrounding nearby sites it appears that for a period of any seven consecutive days the flow of Little River at the site may fall below 20 cubic feet per second with an average recurrence of once every 40 years. Hence, the integrity and adequate capacity of the emergency cooling pond should be assured.

Ground-water supplies in the area should not be affected by the accidental release of radionuclides at the site because the hydraulic gradient of the unconfined water in the relatively impermeable saprolite which mantles the area tends to follow the surface topography, is directed toward (southeast) the Keowee River, and should not be materially altered by the water level in Lake Keowee.

Radionuclides, such as Cesium-137, released in the liquid effluent at the plant site could be deposited and concentrated in the slower moving portions of the Hartwell Reservoir immediately downstream of the site. Large flood flows could subsequently resuspend those deposits and move them downstream. As Hartwell Reservoir is used as a municipal water supply and recreation area the radioactive liquid releases should be kept at a level as low as practical and the environmental monitoring program should include sediment samples from possible areas of deposition downstream of the site.

APPENDIX E

ADDRESS ONLY THE DIRECTOR
BUREAU OF SPORT FISHERIES
AND WILDLIFE

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
BUREAU OF SPORT FISHERIES AND WILDLIFE
WASHINGTON, D.C. 20240

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

Dear Mr. Price:

This is in response to Mr. Boyd's letter transmitting for our comments copies of materials related to the application by the Duke Power Company for an operating license for the Oconee Nuclear Station, Units 1, 2, and 3, Oconee County, South Carolina, AEC Docket Nos. 50-269, 50-270, and 50-287. We have reviewed the Final Safety Analysis Report, including its amendments, the company's draft environmental statement dated July 1970, and the company's letter of October 30, supplementing the statement. As a part of this review the comments of the Center for Estuarine and Menhaden Research, now a part of the National Marine Fisheries Service of the Department of Commerce, were obtained and are included herein.

The station site is adjacent to the company's Keowee Dam and Hydroelectric Station now under construction on the Keowee River just upstream from the existing federally-owned Hartwell Reservoir. Each unit of the nuclear station will use a pressurized water reactor with an output of about 2,584 MWt (886 MWe). A radioactive waste disposal system, fuel handling system, auxiliary structures, and other onsite facilities required for a complete and operable nuclear powerplant would be provided. Construction permits for all three units were issued by the AEC on November 6, 1967. Commercial operation is scheduled for Unit 1 in May 1971; Unit 2 in May 1972; and Unit 3 in June 1973.

About 4,740 c.f.s. of water would be conveyed to the station from the Little River arm of Lake Keowee through an intake conduit to the station to cool the condensers of all three units. Normal cooling water discharges would be into the Keowee River arm of Lake Keowee about 3,700 feet from the hydroelectric station intake. Emergency discharge of cooling waters and normal discharge of liquid effluent from the waste treatment facilities would be into the Keowee Dam tailrace at the headwaters of Hartwell Reservoir.

Fishery resources of Hartwell Reservoir include largemouth bass, crappies, carp, and suckers. In addition, striped bass and walleye have been stocked in the reservoir and trout in the tailwater area. These resources support moderate sport fishing and a minor commercial fishery. Lake Keowee will support fishery resources very similar to those of Hartwell Reservoir, and will provide additional sport fishing opportunity in this area.

The company indicates that (1) studies of the thermal effects of the Oconee station will be included in its ongoing monitoring program to determine the impact on the aquatic resources and the environment, (2) the condenser tubes will be cleaned by mechanical means without using chemicals, (3) the radioactive wastes will be released in concentrations as low as practicable and below the allowable limits, (4) the velocity of the water entering the station intake structure will be low enough to prevent a significant loss of fish through the structure into the plant, and (5) pre-operational environmental radiological monitoring studies will be continued and similar studies will be continued after plant operation.

We are concerned that the thermal and radiological effluents may cause significant damage to fish life and the aquatic environment, and that a significant number of fish may be lost into the intake structure and destroyed. If the surveys establish that the heated or radioactive effluents discharged into Lake Keowee and its tailrace result in changes in Lake Keowee, its tailrace, or Hartwell Reservoir that are significantly detrimental to the fish and wildlife resources or the environment, corrective measures should be taken to reduce the temperature and the radionuclide content of the effluent. Should the studies show that significant numbers of fish are withdrawn with the cooling water, suitable fish protective devices should be installed at the intake structure to reduce the damaging effects to within acceptable limits.

In view of the importance of the sport fishery in Hartwell Reservoir and the fishery potential of Lake Keowee, it is imperative that every effort be made to protect these valuable resource from possible damage from radioactive contamination, heated water, and losses into the intake structure. Therefore, we recommend that the Commission require the company to:

1. Continue to cooperate with the Bureau of Sport Fisheries and Wildlife, other concerned Federal agencies, and the appropriate State agencies in developing plans for radiological and environmental surveys.

2. Continue to conduct such surveys to determine the effects of the plant on the environment and prepare a report of the pre-operational surveys and provide copies of them to the Director, Bureau of Sport Fisheries and Wildlife for evaluation prior to reactor operation.

3. Conduct post-operational ecological and radiological surveys following plans developed in cooperation with the Bureau of Sport Fisheries and Wildlife and other Federal and State agencies, analyze the data, and prepare and submit reports annually until it has been conclusively demonstrated that no significant adverse conditions exist. Copies of these reports should be submitted to the Director, Bureau of Sport Fisheries and Wildlife for evaluation.

4. Make modifications in project structures and operations as may be determined necessary to protect the fish and wildlife resources and the environment as a result of the radiological and environmental surveys.

The opportunity for providing our comments is appreciated.

Sincerely yours,

Samuel P. Benjamin

Director

APPENDIX F

JOHN A. BLUME & ASSOCIATES, ENGINEERS

612 HOWARD STREET • SAN FRANCISCO, CALIFORNIA 94105 • (415) 397-2525

September 4, 1970

RECEIVED SEP 8 1970 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. Edson G. Case, Director
Division of Reactor Standards
U.S. Atomic Energy Commission
Washington, D.C. 20343

Contract No: AT(49-5)-4011
Blume Project No: 2085511
Subject: Oconee Nuclear Station Units 1, 2, and 3
Duke Power Company
Docket Nos. 50-269, 270, 287

Dear Mr. Case:

In accordance with your request we have performed a general review of the FSAR Volumes 1, 2, and 3, Amendments 8 through 18, and BAW Topical Reports, BAW-10008, Part 1 and BAW-10008, Part 2 for the Oconee Nuclear Station Units 1, 2, and 3.

We are enclosing herewith five copies of our report, "Review of the Seismic Design of the Oconee Nuclear Station Units 1, 2, and 3" and the signature page. Based upon our review we feel that the applicant has provided sufficient evidence showing that the Oconee Nuclear Power Station has been adequately designed for the postulated seismic conditions.

Very truly yours,

JOHN A. BLUME & ASSOCIATES, ENGINEERS

Roland L. Sharpe

Roland L. Sharpe
Executive Vice President

DDM:aa

Enclosures 6

REVIEW OF THE SEISMIC DESIGN OF THE OCONEE NUCLEAR STATION, UNITS 1, 2, and 3

(Docket Nos. 50-269, 270, 287)

September 4, 1970

JOHN A. BLUME & ASSOCIATES, ENGINEERS
San Francisco, California

REVIEW OF THE SEISMIC DESIGN
OF THE OCONEE NUCLEAR STATION, UNITS 1, 2, and 3
(Docket Nos. 50-269, 50-270, and 50-287)

INTRODUCTION

This report summarizes our review of the engineering factors pertinent to the seismic design of the Oconee Nuclear Station. The plant is located on Lake Keowee in Oconee County about eight miles northeast of Seneca, South Carolina. The design and construction of the plant was performed by the applicant, Duke Power Company. As a general consultant, Bechtel Corporation was assigned responsibility for the design of the plant buildings. The nuclear power system was designed and manufactured by Babcock & Wilcox. Application for an operating permit has been made to the U. S. Atomic Energy Commission by Duke Power Company. A Safety Analysis Report has been submitted in support of the application to show that the plant has been designed and constructed in a manner which will provide for safe and reliable operation. Our review is based on information presented in the Safety Analysis Report and is directed specifically towards an evaluation of the seismic design of Class I structures, systems, and components. The list of reference documents upon which this review has been based is given at the end of this report.

DESCRIPTION OF FACILITY

The Oconee plant is located on normal Piedmont granite gneisses. The rock underlying the site is hard, structurally sound, and of ancient origin. Excavation to unweathered material provides an excellent foundation for major structures.

The three containment structures are identical. Each is a prestressed, post-tensioned, concrete cylinder and dome. The inside diameter of the cylindrical section is 116 feet and the total inside height of the structure is 208 feet. The vertical wall thickness is 3.75 feet and the dome thickness is 3.25 feet. The structure rests on an 8.5 feet thick reinforced concrete foundation mat. The containment structure is lined with a 0.25 inch steel plate to provide leak-tightness. Reinforced concrete construction was used for the Auxiliary Building.

STRUCTURAL DESIGN CRITERIA AND LOADS

All structures, equipment, systems, and piping are classified according to function or consequence of failure as either Class 1, 2, or 3 as defined in Appendix 5A of the Safety Analysis Report. Class I structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function. Class 2 structures are those whose limited damage would not result in a release of radioactivity and would permit a controlled plant shutdown but could interrupt power generation. Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenence of the reactor in a safe condition. They include all structures not included in Class 1 or 2.

The design loads for Oconee Nuclear Station were based upon ultimate strength design criteria as presented in ACI 318-63. Structural design loads are increased by load factors based on the probability and conservatism of the predicted design loads. Yield capacity reduction factors are applied to the stresses allowed by the applicable building codes. The reactor buildings were designed for an internal pressure of 59 psi. The coincident design temperature is 286°F.

Wind loads were determined for a 95 mph wind as described in ASCE Paper No. 3269. The structure was designed for tornado loadings which correspond to a design tornado with a total tangential and forward velocity of 300 mph. Tornado-generated missiles considered in the design were a 12-foot long, 8-inch diameter pole traveling at 250 mph and a 2000 lb automobile traveling at 100 mph. A simultaneous dfferential pressure drop of 3 psi was assumed.

ADEQUACY OF THE SEISMIC DESIGN

We have reviewed the Final Safety Analysis Report, Volumes 1, 2, and 3, and B&W topical report BAW-10008. Our comments regarding the adequacy of the seismic design are as follows:

1. A maximum hypothetical earthquake with a maximum horizontal ground acceleration of 0.10g was specified for those structures founded on rock and of 0.15g was specified for those structures founded on overburden. A response spectrum defining the response of the structures to the ground motion was postulated for the site. A maximum ground acceleration of 0.05g and a similar response spectrum was used for the design earthquake. These criteria were approved prior to the issuance of the construction permit.

2. The applicant has stated that he used the response spectrum method of dynamic analysis for Class I structures, equipment, and piping primarily within the Reactor Building. Other piping was analyzed by an approximate static method. Time-history analyses were performed to develop response spectra at the points of support of piping and equipment. Based on the information presented by the applicant, we concur in general with the seismic design criteria and approach to the seismic design of Class I structures, piping, and equipment. The analytical techniques used by the applicant are satisfactory and should result in a conservative design. We therefore conclude that the applicant has provided assurance of the adequacy of the seismic design of Class I piping systems.

3. When the structural tests are complete, we would like to review a summary of the predicted stresses, strains, and deflections versus the actual recorded values for each increment of pressure testing. The summary should include an evaluation of the results of these tests as related to the adequacy and conservativeness of the design and analysis assumptions.

4. We have evaluated two Babcock and Wilcox Topical Reports, BAW-10008, Part 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake" and BAW-10008, Part 2, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation." We have concluded that these reports demonstrate that the Reactor Internals and Fuel Assembly have been adequately designed to resist seismic and loss-of-coolant forces.

5. An equivalent static load approach has been used in the analysis of all the hydroelectric facilities. The applicant has presented justification that the equivalent static load approach is appropriate for these structures. We therefore conclude that the applicant has provided assurance of the adequacy of the seismic design of the hydroelectric facilities.

6. The applicant has stated that analyses of qualifying tests were performed on all Class I equipment (pumps, tanks, etc.) to verify that this equipment is capable of maintaining its functions when subjected to the design seismic loadings. We therefore conclude that the applicant has provided adequate assurance that Class I equipment will resist seismic forces.

CONCLUSIONS

On the basis of the information presented by the applicant in the Final Safety Analysis Report and Amendments, and on the basis of information contained in two Babcock and Wilcox topical reports, BAW-10008, Part 1 and BAW-10008, Part 2, it is our opinion that the seismic design criteria and approach to seismic design for Oconee Nuclear Station, Units 1, 2, and 3 have resulted in a design that is adequate to resist the earthquake conditions postulated for the site.

JOHN A. BLUME & ASSOCIATES, ENGINEERS

Roland L. Sharpe

Roland L. Sharpe

Garrison Kost

Garrison Kost

REFERENCE DOCUMENTS

OCONEE NUCLEAR STATION, UNITS 1, 2, & 3

DUKE POWER COMPANY

(Docket Nos. 50-269, 50-270, and 50-287)

Final Safety Analysis Report, Volumes 1, 2, and 3.

Amendments Numbered 8 through 18

BAW Topical Reports BAW-10008, Part 1 and BAW-10008, Part 2

APPENDIX G

DUKE POWER COMPANY

DOCKET NO. 50-269

AEC REGULATORY STAFF FINANCIAL ANALYSIS

	(Dollars in millions) Calendar Year Ended Dec. 31 1969	1968	1965
ng-term debt	$ 663.7	$ 515.0	$ 368.8
ility plant (net)	1,284.1	1,048.6	711.2
Ratio - debt to fixed plant	.52	.49	.52
ility plant (net)	1,284.1	1,048.6	711.2
pitalization	1,204.9	989.2	718.8
Ratio - net plant to capitalization	1.07	1.06	.99
ockholders' equity	541.2	474.2	348.7
tal assets	1,399.0	1,153.1	789.9
?roprietary ratio	.39	.41	.44
rnings for common stock	47.4	44.2	39.3
nmon equity	386.2	369.2	315.0
Rate of return on common equity	12.3%	11.9%	12.5%
t income	54.4	49.1	40.8
ockholders' equity	541.2	474.2	350.0
Rate of return on total equity	10.1%	10.4%	11.7%
t income before interest	78.5	65.8	52.9
abilities and capital	1,399.0	1,153.1	789.9
Rate of return on total investment	5.6%	5.7%	6.7%
t income before interest	78.5	65.8	52.9
terest on long-term debt	29.0	21.9	13.8
No. of times fixed charges earned	2.7	3.0	3.8
t income	54.4	49.1	40.8
tal revenues	344.6	315.0	236.3
Net income ratio (percentage)	15.8%	15.6%	17.3%
erating expenses (including taxes)	266.1	249.2	185.8
erating revenues	342.2	312.2	234.4
Operating ratio	.78	.80	.79
tained earnings	66.9	52.8	

	1969		1968	
pitalization:	Amount	% of Total	Amount	% of Total
Long-term debt	$ 663.7	55.1%	$ 515.0	52.1%
Preferred stock	155.0	12.9	105.0	10.6
Common stock	386.2	32.0	369.2	37.3
Total	$1,204.9	100.0%	$ 989.2	100.0%

ody's Bond Rating: First Mortgage Aa
Debentures A

n and Bradstreet Credit Rating AaA1