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SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

UNITED STATES

ATOMIC ENERGY COMMISSION

IN THE MATTER OF

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

UNITS 1, 2 & 3

OCONEE COUNTY, SOUTH CAROLINA

DOCKET NOS. 50-269, 270 AND 287

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1.0 INTRODUCTION

The Duke Power Company (applicant), by applicatio dated November 28, 1966, and subsequent amendments, has 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.

Each of the three proposed reactors would operate at core power levels up to 2452 Mw thermal. The nuclear steam supply system is, however, designed for 2568 Mw(t) and the applicant anticipates that the reactor will ultimately prove capable of operating at that power level. For this reason, the design of the major systems and components of the proposed facility, including the emergency cooling systems and the containment structure, which bear significantly on the acceptability of the facility under the site criteria guidelines identified in 10 CFR Part 100, have been analyzed and evaluated by the applicant and the regulatory staff at a power level of 2568 Mw(t). The thermal and hydraulic characteristics of the reactor core were analyzed and evaluated at 2452 Mw(t) even though the applicant believes that eventually the core will also prove to be capable of operating at higher power levels. Before operation at any power level above 2452 Mw(t) is authorized by the Commission, the Commission must perform a safety evaluation to assure that the core can be operated safely at the higher power level.

The technical safety review of the proposed plant which has been performed by the Commission's regulatory staff has been based on the applicant's Preliminary Safety Analysis Report (PSAR) and five subsequent amendments all of which are contained in the application. In the course of our review of the material submitted, we held a number of meetings with representatives of the applicant and the nuclear steam system supplier, the Babcock and Wilcox Company (B&W), to discuss the proposed plant and to clarify the technical material submitted. In addition, the Commission's Advisory Committee on Reactor Safeguards (ACRS) has also considered this project and has met and discussed it with both the applicant and us. The report of the ACRS is included as Appendix A. A chronology of the meetings and principal correspondence is given in Appendix B. Reports by our consultants on meteorology, hydrology, geology, seismicity, seismic design and environmental considerations are included in Appendices C through G.

The review and evaluation of the proposed design and construction plans of the applicant at this, the construction permit stage, is only the first stage of a continuing review of the design, construction and operation of the nuclear power plants. Prior to issuance of an operating license for each facility, we will review the final design thoroughly to determine that all the Commission's safety requirements have been met. The units would then be operated only in accordance with the terms of the operating licenses and the Commission's regulations and under the continued scrutiny of the Commission's regulatory staff.

The issues to be considered, and on which findings must be made by an atomic safety and licensing board before the requested license may be issued, are set forth in the Notice of Hearing issued by the Commission and published in the Federal Register on July 27, 1967, 32 F.R. 10996.

2.0 SITE

2.1 Description

The site for the proposed units is in eastern Oconee County, South Carolina, about 8 miles northeast of Seneca, South Carolina. The exclusion area will have a 1 mile radius (from the center of Unit 2), the low population distance is at least 6 miles and the nearest population center is Anderson, South Carolina, population 41,000, located 21 miles southeast of the site. The following table gives the 1965 population figures at various distances from the proposed site as well as a projected estimate for the year 2010:

Distance (miles)	1965	2010
0-5	2,200	3,000
5-10	34,000	46,000
10-20	53,000	73,000

By 1985 when the shoreline of the future Lake Keowee, which will be formed by the Keowee Dam, at the site, and the Little River Dam about 4 miles south of the site, will be fully developed, a transient population of about 7500 on a summer weekend is estimated in the vicinity of the lake.

All land within the exclusion area boundary will be either owned by Duke or controlled by contractual arrangement. Three residences within the exclusion area will be owned by Duke but leased as single family residences with the provision that the occupants will immediately evacuate the exclusion area upon notification by Duke. The nearest residence is 4100 feet from the center of the Unit 2 reactor building. We believe that these occupied residences may be permitted on site because of their distance from the reactor structures and because the applicant has control over the evacuation of these residents.

The reactors will take cooling water from the future Lake Keowee. The earthen fill Keowee and Little River Dams will be designed to withstand the maximum hypothetical earthquake acceleration of 0.1 g (on bedrock) postulated for the site. An earthen dam will also be required to complete one side of the intake canal. Our seismic design consultants, Drs. N. M. Newmark and W. J. Hall, have reviewed the design of the dams and conclude that the dams will withstand the maximum hypothetical earthquake postulated for the site. Their report is attached as Appendix F.

An underwater earthen weir will be located in the intake canal to provide a cooling pond between the weir and the intake canal dike in the event that Lake Keowee should be drawn down excessively. The ACRS has recommended that careful attention be paid to the design and construction of this weir to avoid soil instability and hydraulic erosion and our consultants will examine the final design of the earthen weir to this end.

The topography of the site is used to advantage by using the hydroelectric plants in the Keowee Dam as an emergency power source and by using gravity flow from the intake canal to the tailrace of the hydro plants for emergency condenser cooling.

2.2 Meteorology

The valley below the three units and below the Keowee Dam gives rise to a possible drainage flow of air into the valley. The meteorological model used for accident dose calculations is based on the drainage of air down the river valley with no loss from the valley and the assumption that Pasquill Type F meteorology prevails in conjunction with low wind speeds. As discussed in Section 8.0 of this report, this model leads to doses within the 10 CFR Part 100 guidelines during the maximum hypothetical accident. The Environmental Meteorology Branch of the Institute for Atmospheric Sciences has reviewed the proposed meteorological assumptions and has indicated in its reports, attached as Appendix C, that the model is appropriately conservative. The applicant also proposes to conduct an on-site meteorological program to verify the meteorological assumptions utilized.

2.3 Geology and Hydrology

The reactor structures will be founded on Piedmont granite gneisses. The information submitted by the applicant on geologic conditions indicates no unusual design or construction considerations. The US Geological Survey consultant recommends that critical structures should not be located so that they cross a saprolite cut-fill interface and that fill should be so located as to avoid impingement on critical structures in the event of slope failure. Class I structures, which are the critical structures in these units, will conform to the above recommendations.

Flood studies were made by the applicant on the reservoirs and a maximum hypothetical precipitation of 26.6 inches in 48 hours was used to establish a maximum reservoir flood elevation. The maximum flood level calculated for Lake Keowee was 808 ft msl, 8 ft above the full pond level. The crest of the dams on Lake Keowee will be 815 feet. Our U. S. Geological Survey consultant has stated that flooding of the site from this calculated maximum probable flood does not appear to be possible because of the location of high ground between the containment structures and the lake. The hydrology and geology reports referenced above are attached as Appendix D.

2.4 Seismology

The applicant has proposed a design earthquake resulting in a maximum ground acceleration of 0.05 g. In addition, for a ground acceleration of 0.1 g on bedrock and 0.15 g on overburden, the plant will be designed so that there will be no impairment of function of critical structures and components. The above values are in accord with the recommendations of the US Coast and Geodetic Survey in their report on the seismicity of the site, attached as Appendix E.

2.5 Environmental Monitoring

The applicant has described the scope of an environmental monitoring program to be conducted during construction and operation of the plant. The program will include the monitoring of airborne particulate material, water, soil and silt, vegetation, milk, and fish and animal life. The applicant has cooperated with the Fish and Wildlife Service in developing the monitoring program as indicated by the report dated April 24, 1967, by the Fish and Wildlife Service attached as Appendix G. We believe the scope of the program is adequate.

On the basis of the discussion in Section 2.0, we conclude that the site is acceptable for the proposed units.

3.0 NUCLEAR STEAM SYSTEM DESIGN*

3.1 Summary Description

Each nuclear steam supply system consists of a light water moderated and cooled pressurized water reactor (PWR) which transfers reactor heat to two once-through steam generators from which steam passes to a turbine-generator unit. The low-enrichment UO_2 pellet fuel is held in zirconium rods 0.4 inch in diameter and about 12 feet in length. The fuel rods are held in place by perforated can fuel assemblies which have eight lateral grid spacers over the 12-foot length in addition to the two end fittings. Each assembly contains 208 fuel pins, 16 control pin guide tubes and one in-core instrument guide tube (a 15 x 15 array).

The core is comprised of 177 of these fuel assemblies which rest on the lower grid plate which is attached to the core support barrel which is in turn attached to the reactor vessel wall near the top of the vessel. The core obtains lateral support from the center grid plate, located at the top of the fuel assemblies. An upper grid plate above the core provides lateral guidance for the control rod assemblies.

Reactivity control is accomplished by 69 control rod cluster assemblies and by liquid poison (boric acid) in the reactor coolant. Each control rod cluster assembly consists of 16 stainless steel tubes containing a silver-indium-cadmium alloy which are connected to a "spider" assembly at the top so that the 16 poison filled tubes act as a unit. The control cluster assembly

*Since the three units proposed to be built by the applicant are essentially identical, descriptive information contained in this Safety Evaluation is for the most part stated in terms of a single unit. Where there are differences between the units, these will be identified.

is withdrawn and inserted by rack and pinion drive assembly mounted on the reactor vessel head and driven through a magnetic clutch by a synchronous motor. If a rapid reactor shutdown is desired, the control assembly may be dropped by gravity into the core by releasing the magnetic clutch. As the fuel is depleted criticality is maintained by removing the liquid poison from the system by a chemical addition and control system.

The nuclear flux level is monitored by neutron detectors external to the reactor vessel and by 51 in-core chambers which are inserted through the bottom head of the vessel and into the fuel assembly guide tubes in selected locations. Either the nuclear flux level, high or low reactor system pressure, high coolant temperature, or low coolant flow can initiate a reactor trip through the reactor protection instrumentation which deenergizes the magnetic clutches on the control rods and scrams the reactor.

Water heated (from about 555° F to about 600° F at 2200 psi) while passing upward through the reactor core exits from the reactor vessel through two 36-inch diameter lines near the top of the vessel. Each "hot leg" enters the top of a once-through steam generator. The primary coolant passes downward through the steam gnerator within a bank of tubes where it is cooled by water and steam (at about 530° F and 900 psi) on the shell side. The coolant is returned to the reactor vessel from the bottom of the steam generators through four "cold legs" (two from each steam generator). Each cold leg contains a reactor coolant pump which provides the circulatory driving force.

Steam generated on the shell side of the steam generators is superheated by about 35° F before passing through steam lines to a turbine-generator unit outside the containment building. After passing through the turbine, the low-pressure steam is condensed in the turbine condenser and returned as feedwater to the steam generators by electrically driven condensate booster pumps in series with steam driven feedwater pumps.

The pressure vessel and primary system piping, steam generators, control rod drives, instrumentation, core internals and the first core fuel for each unit will be supplied by the Babcock and Wilcox Company (B&W). The steam turbine will be purchased from the General Electric Company.

The B&W system design is, on the whole, not unlike other recent pressurized water reactor designs. The fuel enrichment, fuel design and arrangement of the core internals are similar while differing in design detail. The proposed design is founded on proven concepts and its similarity to other current designs for pressurized water nuclear plants provides a degree of assurance that a reactor of this type can be successfully built and operated.

The only subsystem of the nuclear steam system which differs substantially in design concept from current practice and experience is the once-through steam generator which provides slightly superheated steam to the turbine-generator. Other subsystems such as the rack and pinion control rod drives and the instrumentation are new designs but are based on experience with similar concepts. These systems will be discussed in more detail in following sections of this report.

3.2 Nuclear Design

The light water moderated and cooled core has been designed to allow operation at 2452 Mw thermal to an average fuel burnup of 28,000 megawatt days per metric ton of uranium. The total clean cold excess reactivity is about 30% $\Delta k/k$. About 10% $\Delta k/k$ is held by the control cluster assemblies and the remainder by soluble poison. The reactor can be made subcritical by 1% $\Delta k/k$ with the highest worth control cluster stuck out of the core at hot conditions by inserting the other 68 control assemblies. A similar margin can be obtained at cold conditions by insertion of soluble poison. The reactivity worth of the control cluster assemblies and the rate at which reactivity can be added by the rods or by the soluble poison system is limited to ensure that credible reactivity accidents cannot cause damage to the system or cause extensive fuel failure.

The nuclear design objectives and limits are similar to other pressurized water reactors now under construction. The control clusters have fewer poison pins per cluster but a larger total number of control poison pins than other designs which is reflected in the larger reactivity increment held by control rods (10% versus about 7%) and lower operational boron concentrations.

A slightly different first-core UO_2 enrichment has been indicated for Unit 2 than for Unit 1. This is a consequence of the applicant's plans to use part of the irradiated Unit 1 fuel in the initial core loading for Unit 2, schedule permitting. We will review this proposal at the operating license stage. It is mentioned here to indicate that restrictions might be placed on the startup procedures of a new unit with irradiated fuel in the initial core. Unit 3 will have a fuel loading schedule independent of Units 1 and 2.

This core, as others of this size and type, is predicted to have a positive moderator temperature coefficient of reactivity under first cycle operating conditions. The positive coefficient has been calculated by the applicant to be about $0.9 \times 10^{-4}\ \Delta k/k/^{o}F$ at the beginning of core life. This is calculated to correspond to a maximum 0.5% $\Delta k/k$ in reactivity which could be inserted by a reduction in moderator density. If this reactivity were inserted during a loss-of-coolant accident caused by the break of the largest system pipe, about 2 full power seconds of energy would be released. The resulting increase in peak fuel temperature caused by such a transient would be acceptable, based on the present calculations. An acceptable value of the positive moderator temperature coefficient will be set at the operating license stage, based on the final design and more refined accident calculations. The applicant has stated that the reduction of this coefficient is feasible. The addition of stainless-steel shims would reduce the coefficient to about $0.44 \times 10^{-4}\ \Delta k/k/^{o}F$ and addition of 2000 ppm natural boron to the shims would eliminate the positive coefficient.

Although we are continuing to evaluate the magnitude of the energy added during a loss-of-coolant accident, we believe that the proposed core design can be accepted at this time since the applicant has demonstrated that, if necessary, the positive coefficient can be reduced or eliminated to bring the consequences of the applicable accident within acceptable limits.

The applicant's calculations indicate the stability margin with respect to xenon oscillations is least for the axial direction and that azimuthal and radial oscillations are unlikely. The applicant has stated that although axial xenon oscillations are not expected, further analysis will be made on final core parameters. If it is found that oscillations could occur, a method for controlling the oscillations will be developed. Calculations have been made to illustrate the ability of control rods with a short poison section to control a divergent xenon oscillation. Since xenon oscillations are relatively slow changes and since the flux imbalance could be detected on the proposed instrumentation, we believe that this method of control is feasible and that analytical and, if necessary, control techniques can be developed prior to the operating stage. Manipulation of the normal control cluster assemblies or power reduction can also be used to prevent or correct, to some extent, the undesirable effects of xenon oscillations.

3.3 Mechanical Design

The reactor internals are designed to withstand steady-state and anticipated operational transients and in addition are designed to resist the effects of seismic disturbances and blowdown forces resulting from a primary system pipe break.

Reactor internals will be fabricated from SA-240 (Type 304) stainless steel and will be designed within the allowable stress levels permitted by the ASME Code Section III, Table N-421 except that the allowable stress levels have been increased by about 10% for the one-time loading which might be imposed by blowdown forces. Radiographic requirements on welds will also be in accordance with Section III of the code. We believe that the proposed design stress levels are conservative and that the higher allowable stress levels specified for the blowdown transient are acceptable in the light of the maximum one-time, if ever, blowdown load and the conservative stress margins specified by the code.

The fuel assemblies are designed for steady-state and transient conditions under the combined effects of flow-induced vibration, reactor pressure, fission gas pressure, fuel growth and thermal strain. The cold-worked Zircaloy-4 cladding is designed to be free-standing. The fuel rod spacers are designed to maintain spacing between the fuel rods but to permit thermal expansion of the rod. Structural stability is obtained from a perforated can assembly around the 15 by 15 array (which includes 16 stainless-steel control pin guide tubes and one in-core instrument guide tube as well as the Zircaloy-clad UO_2 pellet fuel).

The control cluster travel is designed so that the control pins are always engaged in the fuel assembly control pin guide tubes, ensuring that the control assembly can be dropped into the core when required. Each pin of the cluster is also guided above the core by tubes slotted to allow passage of the spider connection. The internals are designed to ensure that the dynamic loading resulting from a loss-of-coolant blowdown will not prevent insertion of the control cluster assemblies. The stresses imposed on the control cluster during scram are minimized by a snubbing mechanism in the rod drive housing and by designing the assembly for the deceleration loads.

We believe that the loads considered for the design of the reactor internals and the stress combination considered in the fuel design form appropriate design bases for these components.

3.4 Thermal and Hydraulic Design

The reactor core is designed to operate at a steady state power level of 2452 megawatts thermal corresponding to an average linear heat generation rate of 5.4 kw per foot of fuel rod and a peak of 17.5 kw per foot. The calculated maximum fuel temperature is about 4160° F and the average fuel temperature about 1385° F.

Although the turbine-generator unit and other equipment is sized for a higher core power level (2568 Mwt) and the fission product release studies are based on this higher power level, the application is for a core power level of 2452 Mwt and we have reviewed the thermal-hydraulic characteristics of the core at this power level.

The reactor core is designed (1) to prevent fuel melting at the design overpower of 114% (2680 Mwt), (2) to provide a high degree of assurance that no departure from nucleate boiling (DNB) will be experienced in the core, and (3) to maintain steam voids in the hottest channel at a level well below the threshold of flow instability. The design overpower is the highest credible reactor power which could result from foreseeable reactor operational transients which are terminated by reactor protective system action (which is initiated at 107.5% full power).

The thermal and hydraulic design evaluation presented in the PSAR made use of the BAW-168 heat transfer relationship to establish that DNB would not be reached at the 114% overpower condition. A probability study was included in the analysis as a means of demonstrating the sensitivity of the analysis to the various input parameters and to allow an expression of the fraction of the core endangered when at various hot channel DNB ratios.

The corner and side channels in each fuel assembly present a unique design problem since the flow in these channels will be significantly reduced by the friction drag of the perforated fuel assembly cans. To compensate for the reduced mass flow rate (lb/hr/ft^2) in these channels B&W has increased the flow area to give approximately the same total mass flow (lb/hr) as in the unit cells within the bundle. B&W further substantiated the design by the results of rod bundle burnout tests of similar geometry but with axially uniform heating. These results were corrected to fit the actual non-uniform case by use of a correction factor obtained from single rod burnout data. The applicant further stated that axially non-uniform bundle tests, similar in geometry to the proposed design, are being run as part of the research and development program at B&W and that the results of these tests will be applied to the final thermal design.

The nuclear and engineering hot channel factors used in the analysis were "design" values - worse than the expected (nominal) values but were not maximized to give an extreme worst case. In response to our request, an analysis was submitted for the unit, corner and wall flow channels using extreme worst case values for the hot channel factors and cell dimensions. This analysis included a design check with the Westinghouse W-3 correlation as well as confirmation of the design by the rod bundle burnout data described above.

We agree that the allowable design heat flux should be designated as a research and development item. On the basis of the preliminary research results submitted it appears that B&W will be able to justify the chosen physical parameters and design limits on the basis of its program of rod bundle burnout tests. We have the added assurance, however, that the design could be approved on the basis of the W-3 correlation if necessary.

The present proposal differs in design from previous plants of this type in that it will have fewer outlet than inlet loops and only two outlet pipes on a large core. The coolant distribution within the reactor vessel must therefore be investigated and the associated pressure drops established.

The applicant has stated that a research and development program is underway to measure flow distribution in the core, fluid mixing in the vessel and core, and the distribution of pressure drop within the vessel. These tests will be conducted on a 1/6 scale model of the vessel and internals. In addition, flow distribution, pressure drop, and mixing data will be obtained with a full scale fuel bundle test assembly and on various models of reactor flow cells.

We have reviewed the development program as described above and believe that the scale model testing and the full scale fuel bundle testing are adequate to provide the necessary information and we therefore believe that the proposed program is acceptable.

3.5 Rod Drive Design

The control rod drives originally proposed for the Oconee units were to be a new design using a nutating disk drive system. Due to development problems related to materials used in the nutating drive, the application was amended to provide a drive mechanism utilizing conventional components.

The drive mechanism now proposed is a rack and pinion device driven by a synchronous stepping motor through a worm gear reducer, unidirectional clutch and magnetic clutch, drive shaft and miter gear set. The drive is operated in primary coolant up to the magnetic clutch where a buffer seal and rotary seal prevent leakage of primary coolant.

The mechanism is housed in two pressure housings: (1) The rack housing is connected to the reactor vessel head and provides guides for the rack, hydraulic snubber and spring stops for the rack, and support for the drive shaft housing and drive motor assembly; (2) The drive shaft housing is vertical and parallel to the rack housing and provides alignment and support for the rotating drive shaft and miter gear set and the buffer seal assembly. These housings, one set for each of the 69 control assemblies, are designed to the ASME Code, Section III, for 2500 psig and 650° F. The two housings are joined at the pinion shaft, near the reactor vessel head, by a bolted, double sealed joint. All gasketed joints are sealed by two Conoseal-type gaskets and pressure testing taps are provided between the gaskets.

The rack is directly connected to the spider of the cluster control assembly by a ball and socket positive latch mechanism. The rack is decelerated during scram by a hydraulic snubber assembly contained in the rack housing above the pinion shaft. The snubber causes deceleration of all of the moving parts in the drive mechanism up to the magnetic and unidirectional clutches. A bottoming spring washer assembly is provided at the bottom of the snubber to absorb and end-of-scram stroke impact.

The drive shaft assembly including the miter gear set rotates to drive the pinion shaft. The drive shaft is actually two splined shafts supported by bearings at the ends and at the center connection to prevent shaft whipping and to assure that the critical speed of the drive shaft is not reached during a scram.

The buffer seal assembly design is similar to that for seals now in use for Consolidated Edison Unit 1, the NS Savannah, the SM Army, the Elk River and BONUS reactors. The major difference is that the seal will work in a vertical orientation with a rotating shaft.

The drive motor assembly utilizes a worm gear reducer to prevent torque from being transferred to the drive motor in the event an upward force is applied to the rack. A unidirectional clutch will be provided within the magnetic clutch to provide for drive rundown after scram and to prevent upward movement of the rack without a rod withdrawal signal from the control system. This type of device has been employed in the drive mechanisms of the LaCrosse Boiling Water Reactor to serve the same function.

Normal rod withdrawal and insertion requires that the magnetic clutch be energized. Scram is accomplished by deenergizing the clutch.

The components of the drive that operate in reactor coolant will be capable of performing their function at 650° F. The seal water injection to the buffer seal is expected to maintain the drive components at a lower temperature.

As indicated in the above description, the rod drives are an assemblage of components of known characteristics. Duke has proposed a development program to fully test the proposed design to demonstrate that the design objectives are met.

Our review of the proposed design indicates that no unusual problems are apparent. We agree with the applicant's design objectives and believe that the development program will provide an acceptable control rod drive mechanism.

3.6 Instrumentation and Control

3.6.1 Reactor Protection System

The reactor protection system monitors vital reactor parameters and automatically causes reactor shutdown when predetermined conditions established for each parameter have been exceeded. The parameters monitored include (1) high reactor power, as measured by neutron flux, (2) low reactor coolant flow, (3) high reactor outlet temperature, and (4) high or low reactor pressure.

The system consists of four identical and independent protection channels, each terminating in a bistable and trip relay. Each of the above parameters is monitored by four channels which are coincident and redundant. The output of each channel of a monitored parameter controls one of four logic channels. The outputs of the logic channel trip relays are combined in a two-out-of-four configuration to operate four circuit breakers which deenergize the two a.c. input circuits feeding the rod drive (d.c.) power supplies. A trip results if one of the two circuit breakers in one a.c. line and one in the other line are opened. Each a.c. line furnishes power to one of the clutch power supplies. Diodes at the d.c. outputs permit testing of the final trip circuits during reactor operation.

The nuclear instrumentation has eight channels of neutron information divided into three ranges of sensitivity: source range, intermediate range, and power range. The three ranges combine to give a continuous measurement of reactor power from source level to approximately 125% of full power, or ten decades of information. A minimum of one decade of overlapping information is provided.

The source range instrumentation channels consist of two redundant count rate channels, each using proportional counters as sensors. These channels are not associated with a protection function; however, they do provide an interlock function (a control rod withdrawal hold and alarm on high startup rate).

The intermediate range instrumentation has two log-N channels, each using identical gamma-compensated ion chambers as sensors. Reactor trip initiation is provided by these channels.

The power range instrumentation consists of four linear level channels using three uncompensated ion chambers per channel. The gain of each channel is adjustable, providing a means for calibrating the output against a reactor heat balance. Protective action consists of reactor trip initiation at preset flux levels.

Primary loop flow information is measured as a function of pressure drop by four independent sensors in each of the two hot legs. The outputs of the eight sensors are combined as pairs such that four independent total flow signals are derived. Each total-flow signal is fed to one of the four power range channels, thus creating four independent power/flow channels. In addition, each pump motor breaker has four contacts which are respectively connected to the four power/flow channels. The logic of the power/flow channels is two-out-of-four, and the channels are independently connected to the reactor protection system logic channels in the same manner as the power range channels.

The power/flow channels will initiate a reactor trip if the reactor power exceeds 107.5% full power or if a mismatch exists between power and coolant flow. The mismatch conditions which will initiate reactor trip will include (1) high power to flow ratio, (2) loss of one pump while the reactor is operating above a predetermined power level, or (3) loss of more than one pump if the ratio of power (at the instant of pump loss) to the steady state flow corresponding to the remaining pumps is greater than 107.5%. An automatic servo action, calling for a reduction in power to achieve a proper power-to-flow ratio will allow the downward adjustment of reactor power to a level commensurate with the remaining pumps unless it is certain (as "determined" by the various comparator circuits) that the impending loss-of-flow transient is sufficiently severe to warrant immediate trip.

There is one set of four pressure sensors and one set of four temperature sensors which respectively trip the reactor on high and low primary system pressure, and high coolant outlet temperature. The logic is two-out-of-four, and the instrument channels are independently connected to the four logic channels in the same manner as the power range channels. One pressure channel also provides a signal to the pressurizer pressure controller. The other three channels will provide trip action on a redundant basis should a failure disable the one common channel and simultaneously initiate a pressure transient.

The nuclear and process instrument channels, by virtue of being redundant, can withstand any single failure without loss of protective function. The coincident logic permits testing during reactor operation. In addition, all instrument channels initiate a trip signal in the event of a.c. voltage loss. Control and safety functions are combined within individual instrument channels only to the extent allowed by criteria governing the design of reactor protection systems.

The four logic channels have been analyzed by us and found to be "fail safe" in the event of voltage loss, immune to single failures, and testable for credible faults. The "fail-safety" is inherent since the channels are tripped when deenergized. A partially or completely failed channel will disable only one relay. Action of the three remaining channels will open all four circuit breakers at the rod drive clutch power supplies. Action of only two of these is required, and they will open at least one circuit breaker at each power supply. Faults within a logic channel will be revealed when the bypassed contacts do not trip their relay when tested. Open circuits are self-revealing. Short circuits between channels can be detected by tripping, one at a time, the "high pressure" contacts located farthest upstream.

Our analysis of the final trip circuits shows that they are "fail safe," immune to single failure, and testable. The loss of one breaker in each a.c. line can be tolerated and scram will not be impeded. Diode failure, open or shorted, will not prevent trip action. A "hot" short at the d.c. line will have no effect since the d.c. system is ungrounded. The system will be equipped with ground-fault detectors. Loss of a.c. and/or d.c. will cause, or tend to cause, reactor trip. Testing at power is accomplished by tripping the circuit breakers one at a time and noting the absence of d.c. voltage at the appropriate power supply output just upstream of its isolating diode.

The manual trip switch contacts are in series with the four circuit breaker undervoltage coils. There is no dependence on instrumentation.

The in-core instrumentation system, consisting of 51 in-core chambers which are inserted through the bottom head of the vessel and into the fuel assebmly guide tubes, provides no automatic control or protection function. The system is located entirely within containment, thereby precluding the need for isolation of penetrations associated with the system.

The engineered safety features are automatically initiated as follows: (1) operation of the core emergency injection systems upon detection of low reactor coolant pressure, (2) operation of the reactor building cooling systems upon detection of high reactor building pressure, (3) containment isolation upon detection of high reactor building pressure, and (4) isolation of valves which are directly open to the reactor building on a high radiation signal.

Five sets of pressure sensing channels initiate the engineered safety features. Each set is coincident and redundant (two-out-of-three logic). Two sets respectively initiate the high and low pressure coolant injection systems. These channels operate through amplifiers and bistable devices and are "fail safe" in terms of voltage loss. Two other sets of three channels actuate the reactor building spray system. In these channels, pressure switches are operated directly; there is no dependence on electrical power for switch operation. The remaining set of pressure sensor channels initiates reactor building emergency fan coolers and containment isolation.

Contacts controlled by these channels are respectively combined into pairs of redundant logic chains which, in turn, control the safety feature systems. These chains may be tested at power by means of two lights wired across the contacts of each chain such that the tripping of a channel produces a unique response from its lights.

Each redundant logic chain is energized from an independent d.c. power source. Should a power source be lost, the downstream circuits fail "as-is." We believe that, with the system redundancy provided, this condition is acceptable.

The engineered safety features' instrument channels do not control the parameters which they measure; i.e., there is separation of control and safety. Manual actuation capability, independent of the instrument channels, is provided.

The ACRS has indicated in its report, and we concur, that the high and low pressure injection systems could be made more reliable by providing diversification in the system actuation signal; that is, by choosing a parameter in addition to low system pressure which would supply an actuation signal. This recommendation is made even though the present instrumentation meets a single failure criterion. We believe that the feasibility of actuating the safeguards on yet a second parameter should be investigated before the present system is accepted at the operating license stage. With this exception, we believe that the operational and engineered safety features protection system instrumentation is acceptable.

3.6.2 Reactivity Control

Reactivity control is maintained by movable control rods and by soluble poison (boric acid) dissolved in the reactor coolant.

The control rod dirves will be designed so that (1) no single failure can cause an uncontrolled withdrawal of any rod, (2) no more than two control groups can be withdrawn at one time, (3) the withdrawal speed will be limited so as not to exceed 25 percent overspeed in the event of speed control fault, and (4) continuous position indication will be provided.

In order to determine the worst effect of "single failures" which might not be confined to a single rod drive, we asked the applicant to perform "startup accident" analyses covering the entire spectrum of initial power levels. This accident assumes the uncontrolled simultaneous withdrawal of all rods at maximum design speed, and further assumes that the excursion is terminated only by Doppler feedback and trip action of the power range nuclear channels. The applicant concluded: "No fuel damage would result from simultaneous all-rod withdrawal from any initial power level." On this basis we have concluded that a single failure which allowed an extra rod group to be withdrawn, a situation less severe than the accident analyzed, would not cause fuel damage.

There will be two "speed limiting" features in the rod drive power system. One is the pulser (or clock) which will be designed not to exceed a certain maximum frequency. The other is a "speed saturating circuit" downstream of the pulser which has the inherent property of not responding to a frequency greater than 125% of rated frequency.

There are two independent analog rod-position sensors at each rod drive, a potentiometer and a linear variable differential transformer (LVDT). There are two independent limit switches. In addition, the LVDTs will also generate limit signals. Thus, there are redundant analog and limit position indicating systems at each rod drive. Each analog signal at a rod drive can be fed into the individual rod position indicator.

Based on our analysis, we believe that the applicant's rod drive system criteria are acceptable, that no single failure in the control instrumentation can produce an excursion which will cause fuel damage and that the proposed rod drive designs can be built in accordance with these criteria.

Reactivity is also controlled by a permissive system which allows manual dilution of the primary system coolant boron concentration when a particular control rod group reaches the fully withdrawn position. Dilution is automatically terminated when the rod group, driven down by the servo, reaches a prescribed position, or when the integrated dilution flow has reached a preset maximum. We understand that these circuits will be designed in accordance with protection system standards and no single failure will prevent automatic termination of dilution. On this basis we believe that the proposed design is acceptable.

In summary, we conclude that the applicant's design criteria relating to instrumentation and controls are satisfactory and that the proposed preliminary designs conform to these criteria.

3.7 Reactor Coolant System

3.7.1 Primary System

The reactor coolant is transferred to the top of the two once-through steam generators through two 36-inch lines from the upper reactor vessel plenum. Water is returned from the bottom of the steam generator to the vessel via four 28-inch lines. Circulation is provided by a single-speed, shaft-sealed pump in each of the four cold legs.

The reactor vessel plate material has been specified as SA-302 Grade B clad internally with stainless steel which is similar to that used in previous designs. The major exception to previous designs in the primary system is that the 36-inch and 28-inch ID recirculation piping will be A-212 or A-106 carbon steel internally clad with stainless steel and designed to the ASA Code. The pump casings are designed to ASME, Section III. The primary system vessel classifications are ASME Section III Class A. The letdown coolers, which lower primary coolant temperatures and pressure before it enters the purification and chemical addition systems, differ in code classification from current designs in that they will be Class C rather than Class A on the primary coolant side. We believe that this classification is acceptable since the heat exchangers are nonregenerative and not subject to the thermal transients that regenerative heat exchangers would experience. In addition, the heat exchangers can be isolated by valve closure if necessary. We believe the material specified for use in the primary system, the choice of design codes, and the fabrication procedures indicated will assure high primary system quality commensurate with the reliance to be placed upon the system.

We have reviewed the calculated fast neutron exposure of the Oconee reactor vessels and the corresponding shift in the NDT temperature. The reported time-integrated fast (energy greater than 1 Mev) neutron exposure of the vessel of 3×10^{19} neutrons per square centimeter, and the estimated nil ductility transition (NDT) temperature shift of 260° F, should not cause any significant operational restrictions during the proposed life of the plant.

The neutron exposure of the vessel of 3×10^{19} neutrons per square centimeter was calculated over a 40-year life of the vessel using an 80 per cent load factor and the maximum axial peak-to-average power ratio of 1.7. The calculations were performed using the transport code TOPIC, which is an S_n code designed to solve the one-dimensional transport equation in cylindrical coordinates.

Although the calculational method employed does not describe experimental irradiation data with a high degree of accuracy, we believe a sufficient factor of safety has been applied to the calculations to make them conservative. This conclusion is based on consideration of the Oconee core size, power density, and the inner diameter of the reactor vessel.

The type of neutron flux monitors to be used and the method which will be employed to determine the neutron flux at the sample locations in the irradiation surveillance program will be further evaluated prior to issuance of an operating license.

The applicant has stated that access for inspection can be gained to all internal surfaces of the primary vessel by removing vessel internals and that it will be possible to gain access to the external vessel surfaces although this would require the removal of thermal insulation. The scope and frequency of the inspection program will be reviewed at the operating license stage with the recommendation of the ACRS in mind that the program should reflect the available technology.

The applicant presented the results of an analysis of the thermal transient experienced by the hot reactor vessel wall when deluged with cold safety injection water after a loss-of-coolant accident. Ductile yielding, brittle fracture and fatigue failure were considered in the analysis. The results of the analysis indicate that no loss of vessel integrity would be experienced even if large flaws were presumed to exist in the vessel wall at the beginning of the quenching.

As recommended by the report of the ACRS, we will further review the details of the calculational procedure to ensure that conservative assumptions have been made in this analysis and that the calculational models are supported by experimental information.

3.7.2 Once-Through Steam Generator

Unlike other recent pressurized water plants which have used a U-tube steam generator, in which the primary coolant enters and exits from the bottom of the generator, the Duke design is a single pass or once-through heat exchanger. In this design the primary water enters the top of the steam generator, is cooled while passing downward through the Inconel tubes and exits from the bottom head. The secondary feedwater is sprayed into an annulus in which there are no tubes near the generator carbon steel shell. The feedwater is heated as it falls by steam which is allowed to bypass from the heated region back to the annulus.

When the feedwater reaches the bottom of the annulus it is near the saturation temperature and is boiled as it passes upward through baffling around the tubes which contain the primary fluid. When the steam exits from the generator, all the water has been evaporated and the steam is dry with about 35° F of superheat.

At full power the feedwater to the steam generator is controlled by a combination of megawatt demand, system frequency and secondary steam pressure. In addition to these parameters, maximum and minimum demand limits and a rate limit control the feedwater flow. This integrated controller is similar in concept to the controllers used on conventional steam plants and will be further reviewed at the operating license stage.

Since the tubes are welded to the tube sheets which are in turn fixed to the generator shell, differential expansion and stresses can be experienced when the tube and shell temperatures are different. During startup and shutdown when the temperature difference is greatest (about 40° F) the stresses are compressive and small; only about 25% of the code allowable stress for the Inconel material. Buckling of the tubes is avoided by lateral support at 40-inch intervals.

An analysis has also been performed on the effects of continuing feedwater to the steam generator after a steam line break and the stresses imposed on the tubes were found to be below the yield strength of the material.

A development program for the steam generator has been proposed by the applicant, including vibration and blowdown tests and we will require a report of the test data and an analysis of their significance before final approval of the design at the operating license stage. We believe that both primary and secondary side blowdown tests should be performed during the developmental program. The applicant has indicated that both types of blowdown tests will be performed. Our analysis to date indicates that the applicant has a sound design basis for the steam generators.

3.8 Secondary System

Steam passes from the steam generator at about 530° F and 900 psi through steam lines (one for each generator) through the containment wall and to the turbine building. Safety valves and the automatic dump valves are mounted on each line outside the containment. Each steam line passes through two turbine stop valves before reaching a cross-tie with the other steam generator line. The steam then passes through control valves and to the turbine steam chest. After passing through the turbine, the low energy steam is condensed in the main condenser and returned through feedwater heaters and two half-capacity steam turbine driven feedwater pumps to the steam generator. A low capacity (5% full power) emergency steam turbine driven pump is provided for decay heat removal during normal or emergency shutdown.

Secondary system water quality is maintained at a high level by full-flow demineralizers which will minimize stress-corrosion problems in the steam generators.

The secondary system is designed to reduce load automatically to station auxiliary loads in case of a blackout or other transient on the external power grid. This would be accomplished by briefly venting secondary steam to the atmosphere while feedwater flow is reduced to the generators. This feature will be tested by Duke during the startup of each unit.

Steam line isolation valves in addition to the turbine stop valves have not been proposed in the Duke system. We believe that isolation valves are not required in this case since the proposed system accomplishes the same objectives without the valves and is therefore equivalent to other designs. The objectives in installing steam line isolation valves are: (1) to prevent blowdown of more than one steam generator after a steam line break; (2) to provide a leak-tight barrier after a loss-of-coolant accident when there has been leakage through the steam generator tubes; and (3) to prevent secondary criticality after a steam line break accident.

The blowdown of both steam generators is prevented for the proposed units because the turbine used (General Electric) has stop valves on each steam line and does not require a cross-tie before the stop valves to attain a constant steam temperature. In the proposed design a heat sink would be provided even if both steam generators were blown down. This would be accomplished by supplying feedwater at 700 psig by electrically driven condensate booster pumps feeding through the normal steam driven feedwater pumps.

A leak-tight barrier in case of steam generator tube leakage after a loss-of-coolant accident is maintained because of the leakage characteristics of the turbine stop valves. We believe that the stop valves in this system can be considered a leakage barrier if the secondary system is tested for leakage integrity in conjunction with containment leak rate testing. The applicant has indicated that the leakage integrity of the valves will be demonstrated.

As discussed in Section 8.2 of this report, the applicant's calculations indicate that a return to criticality due to cooldown of the primary coolant will not be experienced after a steam line break.

The proposed system provides protection equivalent to previous designs in these three areas. We consider that the present proposal is acceptable.

4.0 CONTAINMENT

4.1 Description

The containments proposed for the Oconee Units 1, 2, and 3 are concrete structures prestressed across the dome and throughout the side walls and employ reinforced concrete for the base slab. The containment structures are of the same basic design as those of the Florida Power and Light Company Turkey Point reactors and the Consumers Power Company Palisades reactor containments and most structural details are similar. These containments were designed by the Bechtel Corporation and the applicant has retained Bechtel as a consultant for this design.

Each containment building has the shape of a right circular cylinder with a shallow spherical sector dome and flat slab base. A mild steel liner is attached to the inner face of the concrete shell to provide leak-tightness. The cylinder walls are prestressed both circumferentially and vertically and the dome is prestressed in a three-way tendon system. The tendons are bundles of wires which are stretched in tension to exert a compressive force on the concrete walls. Under accident conditions the pressure within the containment relieves part of the compressive force on the concrete and creates a slight increase in tensile stress in the tendons. Prestressing is used because the concrete is strong in compression but weak in tension.

The prestressing tendon pattern is deflected around the major cylinder penetrations (personnel and equipment access hatches) and additional mild steel reinforcement is provided for local moment and shear loads. Shear loads in the base are carried by the concrete section, by radial stirrup reinforcing, by vertical mild steel reinforcing and by the mild steel liner participating through composite action.

4.2 Loadings

The major loadings considered by the applicant include dead load, accident pressure, accident temperature, seismic, and wind. The applicant has also indicated consideration of external pressure, buoyant water force, tornado and missile loadings. The loadings considered and their manner of combination are the same as previously used for Turkey Point and Palisades containments. The manner of load combination considers all significant loads and we believe that the manner of load combination is acceptable.

As noted by our seismic design consultants in Appendix F to this report, the applicant will use earthquake loadings derived from response spectra similar to those presented in TID-7024, appropriately scaled to the design and maximum hypothetical earthquakes postulated for this site. We believe that the treatment of seismic considerations is acceptable.

4.3 Structural Design Details

Several changes in design detail for these units are noteworthy although the overall design concept is the same as the Turkey Point and Palisades structures.

The applicant has revised the design criteria relating to strength of the concrete in shear under combined loading. We consider that the revised criteria have clearly defined the design approach.

The base-to-cylinder liner detail has been improved. In previous submittals a rather rigid transition was proposed whereas the design for the Duke structures has incorporated a flexible liner transition section.

It is our judgment that the present method for this junction should result in considerably better performance than that of the previously proposed method with respect to potential leakage under design basis accident loading.

The design of penetrations is, also, considerably improved over previous designs. The Duke containment design indicates use of sizable rigid shear keys as additional assurance of adequate shear resistance at penetrations. It also indicates use of increased strength piping sections at penetrations to preclude a pipe failure from jeopardizing liner leakage integrity at the liner-penetration junction.

The equipment access hatch is 19 feet in diameter for the Duke units whereas for previous similar containments it was between 11 and 15 feet. This represents a considerable increase in overall hatch size and, to an extent, increases the designer's problems with regard to tendon deflection around the opening and proper reinforcement for local stresses. However, an opening 19 feet in diameter is not a major perturbation in the design of the structure and the method of analysis that the applicant proposes to use to analyze this opening should not be invalidated by the increased size. In addition, the use of extensive instrumentation has been proposed around the opening to provide confirmation of the design during structural acceptance testing.

4.4 Materials and Construction

The materials of construction, i.e., the prestressing system, tendon protective grease, concrete, reinforcing steel, and liner plate materials are essentially the same materials used for the Turkey Point and Palisades facilities. These are high quality, proven materials. The applicant has indicated that, based on extensive resistivity tests at the site, no cathodic protection system will be required.

The existence of a well established, experienced construction department in the Duke Power Company organization which will handle the construction, lightens considerably the task of the quality control organization in ensuring that the plants are constructed in accord with the requirements of the design. User testing of the materials of construction will be performed. The construction quality control program provides an adequate separation of construction and inspection functions, adequate authority for the quality control personnel to perform properly, and design group review of the construction progress.

4.5 Testing and In-service Surveillance

An extensive program of acceptance testing has been indicated which we believe will provide a high degree of assurance that anomalous structural behavior will be detected.

Detailed attention is being given to liner inspection during construction. Use of vacuum boxes, leak chase channels and pressurized penetrations have been proposed and these measures should effectively identify and allow correction of potential sources of containment leakage as construction proceeds. The provisions of the ASME Boiler and Pressure Vessel Code, Section VIII, are being followed. These provisions, while intended primarily as a quality control measure for strength welds and not directly applicable to strain-following, leakage-control membranes, do provide an additional check on weld quality. The amount of radiography specified is considered adequate to identify poor welding and, as such a measure, is considered both desirable and acceptable. In addition, a final integrated leakage test is planned as a final check on the containment's capability to meet its leakage performance requirements prior to operation.

We have been informed that the applicant will increase the amount of nondestructive liner weld inspection in response to the concern expressed by the report of the ACRS by increasing the amount of nondestructive inspection from 10% to 20% of the liner welds. This is in addition to the previously specified 100% visual inspection of each of the two weld passes and essentially 100% vacuum box leakage testing of all welds. We believe the inspection outlined will ensure a leak-tight membrane.

Detailed in-service surveillance programs have not been established. However, the design will have adequate capability for a suitable program and review of these areas will be made at the operating stage.

The containment design proposed has a high degree of conservatism. It is concluded that the design, as presently proposed, and the construction, as indicated, will result in structures adequate for the intended purpose.

4.6 Containment Leakage

The applicant has proposed a penetration room confinement system which would process leakage from most containment penetrations through a filter system external to the containment. During an accident, the penetration room would be maintained at a slight negative pressure by blowers which would take suction from the room through filters designed to retain iodine. All penetrations except equipment hatches and steam lines pass through the penetration room. The steam lines are welded to the containment liner and therefore leakage should be negligible. The applicant has proposed that the space between the gaskets on the outer doors of the equipment and personnel hatches be routed to the penetration room by small tubes, thus providing filtration of leakage from these penetrations also. We believe that the filtration scheme as proposed is acceptable and can be considered an engineered safety feature which is operable during an accident as discussed in Section 6.4 of this report.

The containment leak rate was specified as 0.5%/day in the initial application and credit was requested for filtration of 50% of the total leakage since it was reasoned that at least this fraction would be due to penetration leakage. In response to our concern for means to test this division of leakage, the applicant has modified its proposal to the following:

(1) the total containment leak rate at the peak accident pressure will either be shown to be less than 0.25%/day, or

(2) the total leakage at the peak accident pressure shall be less than 0.5%/day and the difference between the total leakage and the measured leakage from testable penetrations shall be less than 0.25%/day.

We believe that the above approach is acceptable since the testable penetration leakage will be filtered and that the advantage of filtering the most likely source of containment leakage justifies a testing frequency interval associated with 0.5%/day.

4.7 Isolation Systems

Lines which penetrate the containment have provision for isolation. The degree of redundancy depends on the function and configuration of each system. In general, lines which are (1) connected to the primary system,

(2) normally open to the containment atmosphere, or (3) likely to be ruptured during an accident are protected by redundant automatic valves. Lines which must remain open to allow functioning of engineered safety features during an accident must have provision for manual isolation.

Lines which vent the containment atmosphere are closed both on an engineered safety feature and a high containment radiation signal. Closed systems which have a low probability of rupture during an accident are provided with at least one automatic valve external to the containment. The isolation system, including instrumentation, is designed so that no single failure can preclude containment isolation.

We have reviewed the instrumentation and valve arrangements proposed and have found that they conform to current design standards and are acceptable.

4.8 Containment Design Pressure

A parametric analysis has been performed by the applicant to establish the peak containment pressures during a loss-of-coolant accident and to size the containment cooling systems. A spectrum of pipe break sizes between 0.4 ft^2 and 14.1 ft^2 has been evaluated to determine the response of the reactor building pressure.

Assumptions used in the analysis were as follows:

(1) One of three high pressure pumps operate, two of three low pressure pumps operate (with a starting delay of 25 seconds) and no core flooding tanks are available. Including the core flooding tanks would decrease the peak blowdown pressure by about 3 psi.

(2) Reactor building structures were assumed to serve as heat sinks.

(3) The FLASH code was used to determine mass and energy releases to the reactor building.

(4) Following blowdown a 20-region SLUMP code was used to calculate the core thermal transient. The metal-water reaction was included in this calculation using a parabolic rate equation.

(5) During blowdown, a core surface heat transfer coefficient of 1000 Btu/hr-ft^2-$^\circ$F was used to maximize heat transferred to the containment.

(6) Heat removal from the core after blowdown was calculated by assuming a heat transfer coefficient of 100 Btu/hr-ft^2-$^\circ$F. As any core segment reached 4800° F it was assumed to drop to the bottom of the reactor vessel and undergo an additional 10% metal-water reaction and release all heat to the containment by steam generation.

The complete spectrum of breaks was analyzed only for the hot leg since this gave the longest blowdown times, greatest heat transfer and highest containment pressure. The highest blowdown pressure peak (56.8 psi at 30 seconds) was found to result from a 3 ft^2 break. The highest post-blowdown pressure (55.9 psig at 200 seconds) resulted from the 14.1 ft^2 break. The second pressure peak is a consequence of the assumed transfer of decay and metal-water reaction heat to the containment and is limited by the operation of the containment cooling systems. The calculated peak pressures are below the containment design pressure of 59 psig.

An analysis was also performed by the applicant to illustrate that the containment will withstand the metal-water reaction associated with inoperability of core quenching systems. No injection flow was assumed and the analysis was terminated when the reactor vessel boiled dry. This gave a peak pressure of 56.7 psig at about 220 seconds, also less than the containment design pressure.

The zirconium-water reaction capability of the containment was calculated assuming three emergency fan cooling units in operation (the design heat removal capability). The capability of the containment under these conditions (including hydrogen recombination) is about 30% metal-water reaction at 600 seconds and 100% at 3400 seconds. If all containment cooling were in operation, the containment could withstand 100% metal-water reaction at 1200 seconds. This capability is similar to that found in the Florida Power and Light Turkey Point reactors and the Consumers Power Palisades reactor.

Our evaluation of the containment design pressure analysis and the containment's capability to withstand metal-water reaction indicates that it is acceptable.

5.0 ELECTRICAL SYSTEMS

Upon completion of Unit 1, off-site power will be available from the 100 kilovolt (kv) system and from the 230 kv system which feed power into Oconee over separate transmission lines from Duke's Jocasse and Central power stations. An additional 230 kv tie to Duke's Tiger station will be installed upon completion of Unit 2; and, upon completion of Unit 3, a tie to Duke's 500 kv system will be installed. All off-site lines will be energized from several power generating stations, and the Duke system is designed to withstand the step-loss of any single generating unit within its network.

Each reactor unit will generate electric power at 19 kv which will be fed through an isolated phase bus to a unit step-up transformer where it will be raised to 230 kv for Units 1 and 2, and 500 kv for Unit 3. Two 230 kv overhead transmission lines will carry power between Units 1 and 2 and the station switchyard which will be connected to the existing Duke 230 kv transmission line. From Unit 3, an overhead transmission line will carry power between the station and the switchyard which will be connected to Duke's 500 kv transmission network. An autotransformer will tie together the 230 and 500 kv systems at the station switchyard. In addition, a 100 kv line will be run from the gas-turbines at the Lee Power Station. Each unit will have its own startup transformer. The 100 kv line will terminate at a transformer separate from the switchyard which will serve all three units, as required.

Normally, each unit will supply its own auxiliary loads directly from the generator via the station auxiliary transformer. Since each unit is being designed to accept a 100% load rejection, the primary source of power for the auxiliary loads in the event of system loss will be the unit generators themselves. In the event of a unit trip, the power sources will be automatically switched onto the auxiliary busses in the preferential sequence of (1) the startup transformer bus (includes the Keowee Hydro Station overhead line), (2) the other units' auxiliary electrical system when available, (3) the 100 kv transmission line from Lee Station, and (4) the Keowee Hydro Station 13.8 kv underground line.

Upon loss of the external grid, redundant voltage and frequency sensing devices on each of the 230 kv switching station busses will initiate, through separate and redundant channels, tripping of all 230 kv switching station isolation breakers, closing of all 230 kv switching station power supply breakers and startup of both Keowee Hydro units. The hydro units will synchronize and be connected to the 230 kv lines. One unit will also feed the 13.8 kv underground line. Shedding of non-essential loads (a requirement because of the limited capacity of the 10 MVA emergency power transformer connected to the underground line) will be accomplished by circuit breakers with duplicate trip coils energized from different d.c. busses. Upon tripping of a given Oconee unit (caused, for example, by a loss-of-coolant accident) the emergency power sources will be automatically switched onto the emergency (4.16 kv) busses of the affected unit in the sequence stated previously.

Our analysis indicates that the sequencing system is essential to plant safety since its failure could leave the emergency busses with no power. We have been assured that this system will meet the single failure criterion. This will be reviewed in detail at the operating stage.

Four 125 volt direct current (v.d.c.) batteries and six battery chargers will be supplied for Unit 1. One pair of batteries and one set of three chargers will feed one 250/125 volt bus, and the remaining pair of batteries and set of chargers will feed a redundant 250/125 volt bus. Upon completion of Unit 2, this d.c. system will serve both units. A third three-wire system will be installed upon completion of Unit 3. Switching circuits will permit any d.c. system to serve any unit.

Initially, there will be six 125 v.d.c. distribution panels, each of which will receive d.c. power from both three-wire d.c. sources through isolating diodes. Two more panels will be installed with Unit 3 and will be similarly powered.

Our review of the station battery system indicates that it is redundant and can be tested. Voltage at each of the panel-boards is derived from redundant sources feeding through isolation diodes such that failure of one source does not affect the voltage at the panel board bus. Loss of voltage at a panel board bus will not negate the d.c. system function.

Four vital instrument busses (single phase) will be provided for Units 1 and 2, and will be independently energized from static inverters connected to one of the six d.c. distribution panels. Two more vital instrument busses will be added with Unit 3. These will be powered, through static inverters, from the two Unit 3 d.c. panels.

In addition, there will be three single phase 120 volt alternating current (v.a.c.) regulated instrument busses. These will normally be connected to the 600 v.a.c. busses of their own units through regulating equipment. Provision will be made to switch over to the vital instrument busses, if necessary.

The engineered safety features auxiliaries are provided with redundancy. To maintain this redundancy, the applicant has stated that these auxiliaries will be connected to redundant busses such that safety feature auxiliaries performing the same function are connected to different busses. Each of these busses is supplied from the redundant 4160 volt main feeder busses which are, in turn, supplied from the redundant sources described previously. We believe this design approach is acceptable since it is an effective and simple way of implementing the single failure criterion.

We believe that the vital instrument busses can be designed so that no single failure can cause a loss of voltage at all vital instrument busses and so that continuity of normal or emergency operation can be maintained under a single failure condition.

We believe that the external power sources available for use at the plant provide a high degree of assurance that power will be available when required. As with previous applicants, however, we have required that the available on-site power which is directly under the control of the applicant meet a single-failure criterion. The design and utilization of the Keowee Hydro Station Units as emergency on-site power sources which meet a single-failure criterion is discussed in Section 6.3 of this report.

6.0 ENGINEERED SAFETY FEATURES

6.1 Core Cooling

The applicant's design basis for the emergency core cooling systems is that mechanical integrity of the core shall be maintained to prevent damage that would interfere with core cooling and that metal-water reaction shall be limited to less than approximately 1% after a loss-of-coolant accident. Since the analyses show that the clad hot spot maximum temperature is about 2000°F, the design basis implies that no clad melting will take place.

The applicant's criterion for maintenance of mechanical integrity during the blowdown is that deformation of reactor internals shall be limited to ensure the capability to insert control rods and also to cool the core. The applicant has proposed that the stress levels to be met in the analysis of blowdown forces on reactor internals correspond to the minimum specification yield strength value specified in Section III of the ASME Code. As discussed in Section 3.3 of this report, we believe that the applicant's proposal for maintenance of mechanical integrity during blowdown is acceptable. As recommended by the ACRS, we will review the results of the detailed blowdown calculations more fully when these become available.

Core cooling for any location and size of primary coolant pipe break up to the double-ended rupture of a recirculation pipe will be provided by high pressure injection pumps, low pressure injection pumps and core flooding tanks (accumulators).

The core flooding tank system is composed of two tanks separated by check valves from the primary system. Borated coolant is maintained in the tanks at 600 psi by compressed nitrogen. Injection of the borated coolant into the primary system is initiated by the stored energy when the reactor pressure drops below 600 psi. The tanks discharge directly to the reactor vessel rather than into a reactor recirculation line as in other pressurized water reactor designs. The water flows between the reactor vessel wall and the thermal shield and enters the bottom of the core.

An analysis by the applicant was presented which provided the basis for the choice of the flooding tank pressure, size of the discharge line and the fraction of nitrogen in the tank volume. The combined coolant content of the two tanks is more than sufficient to cover the midplane of the core assuming no liquid is initially in the reactor vessel. The design values chosen for the flooding system are calculated to accomplish this within 25 seconds after the double-ended rupture of a 36-inch reactor outlet line. The hot spot temperature is limited to less than $2000^{\circ}F$ for the largest line break.

Although the sizing analysis of the tanks has been completed, we will require that an analysis of all possible means of causing core bypass flow be completed before the flooding tanks are accepted at the operating license stage.

The applicant has not provided extra core flooding tanks beyond those required to meet, with some margin, the cooling requirements imposed by a double-ended break of the largest primary system coolant line. No extra component was provided since (1) the accumulators are passive and require no initiating signal, (2) the break of a primary system pipe would not cause loss of the flow from an accumulator since the two accumulators have separate penetrations into the reactor vessel, and (3) in case of the break of an accumulator line, the core could be cooled by the remaining accumulator. When allowance is made for the loss of an accumulator as a result of a line break in designs in which the accumulators discharge to coolant lines rather than the reactor vessel, the presently proposed system is equivalent in capacity to other designs. The reactor will not be operated unless the above conditions are satisfied. That is, an accumulator could not be valved off for maintenance during reactor operation as noted in the ACRS letter.

In addition to the flooding tanks, coolant injection is also provided for each reactor by three low pressure pumps which will each deliver 3000 gpm at a vessel pressure of 100 psig. These pumps initially take suction from the 350,000-gallon borated water storage tank provided for each reactor unit and are converted to a recirculation mode by operator action in 25 to 40 minutes, depending on the number of pumps in operation. At 25 minutes after reactor scram the decay heat level of the core is such that one of the two low pressure injection coolers can remove the decay heat from the spilled coolant contained in the bottom of the containment building. The low pressure injection system delivers water to the same nozzles as the core flooding tanks. Under normal shutdown conditions these pumps serve as decay heat removal pumps.

During the period while the water source is the borated water storage tank, the three high pressure injection pumps can also deliver water to the reactor. Each high pressure pump will deliver about 350 gpm at 1800 psig and about 500 gpm at 470 psig. These pumps provide makeup for small breaks for which the reactor would remain at a high pressure. In the unlikely case that reactor pressure should remain high over a long period of time so that the low pressure injection pumps could not operate, water could be returned from the containment to the borated water storage tank through a test line and permit extended operation with the high pressure pumps.

One high pressure pump will be used continuously during plant operation to provide seal water to the reactor coolant pumps. Since only one high pressure pump is required (as a design specification) to supply water during emergency service, we believe that adequate redundancy exists in the two pumps not used for normal service. In addition, the normal use of one pump provides assurance that an operable pump will be available if required for emergency service.

The design objectives of the emergency core cooling systems include providing core cooling for all break sizes in the primary system piping. The core cooling analyses for all break sizes has not been completed but the analysis of the double-ended rupture of the largest coolant pipe (14.1 ft^2) has been presented and a spectrum of hot leg breaks (including 8.55, 3, 2, 1 and 0.4 ft^2) was analyzed with respect to system pressure history and coolant mass release during blowdown. The analyses presented for the spectrum of break sizes, although not complete with respect to core temperature history, do contain sufficient information to provide assurance that all break sizes can be accommodated by the proposed systems. As recommended by the ACRS, we will review the detailed design of the emergency core cooling systems and the performance analysis for the entire spectrum of break sizes as soon as the information is available.

We conclude that the proposed emergency core cooling systems meet the intent of the Commission's General Design Criteria which have been published for comment in that part of the core cooling systems relied on are passive in nature and adequate redundancy is provided in other systems to assure reliable functioning of the systems. This is true even though the "two core cooling systems" mentioned in Criterion 44 are not provided. The final design will be evaluated in light of the Commission's criteria as they are formulated at the time of the operating license review.

The applicant has 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. Two methods of relieving hot leg pressure to the cold leg have been proposed as solutions to this problem: (1) check valves located on the Core Support Shield which would be held closed by higher pressure in the outer annulus during pump operation or natural circulation, or (2) a rupture disk which would be designed to blow out under internal steam pressure, but which would withstand the external operating pressure differential. Either method would be designed to open on an internal pressure less than 3.5 psi. We understand that the applicant intends to utilize the check valves at present but that alternate means will continue to be studied as the design progresses. The check valves would be designed and supplied by a valve manufacturer with experience in the fabrication of check valves requiring similar specifications.

We believe that the check valves proposed could provide an acceptable solution to the steam bubble problem. Other potential problems arise, however, because of the use of these valves which we believe must be considered in the final design:

(1) The core support shield must be locally strengthened to compensate for the removal of material.

(2) The forceful opening of the valves against the reactor vessel during blowdown must be considered.

(3) The consequences of loss of a valve must be evaluated or the design must provide assurance against such loss.

(4) The effect on normal operation must be considered, particularly any possibility of bypassing or short-circuiting the core during pump operation or natural circulation.

(5) The valves must be capable of being tested, inspected, and maintained.

We believe that the above problems are all capable of solution and that the present proposal could resolve the steam bubble problem. As recommended by the ACRS we will further review the final design of this feature.

In summary, we believe that there is an adequate basis for concluding that the emergency core cooling systems are acceptable. We will continue to review (1) the core cooling analyses for the full spectrum of line break sizes and locations, (2) the final design of the cooling systems when available, (3) the possibilities that could lead to emergency cooling water bypassing the core, (4) the blowdown forces on reactor internals, and (5) the solution to the steam bubble problem to assure that the final design will perform its intended function.

6.2 Containment Cooling Systems

Two differently designed containment cooling systems are provided: (1) containment spray pumps which take water initially from the borated water storage tank and then from the containment sump and deliver it to the containment atmosphere through redundant spray headers and (2) three emergency cooling units each consisting of a fan and a tube cooler which will remove heat from the containment atmosphere and transfer it to the low pressure service water system.

The containment cooling requirement is that the post-blowdown reactor building pressure be maintained below the design containment pressure. This requires an initial heat removal capacity of 240×10^6 Btu/hr. This requirement can be satisfied by either: (1) 2 of 2 spray pumps, (2) 3 of 3 fan coolers or (3) 2 of 3 fan coolers and 1 of 2 spray pumps. Adequate containment cooling is supplied if either system is assumed to be completely inoperative or if each system is degraded by a single failure. We believe that these systems provide adequate redundancy for containment cooling and have sufficient capacity to reduce the containment pressure (and thereby reduce leakage) after the design basis accident.

6.3 Emergency Power and Water

6.3.1 Design Basis Accident Conditions

To cope with the postulated loss of coolant accident coincident with loss of network power, the applicant has proposed that two hydroelectric plants, located on-site in the Keowee Dam be used as the emergency power source. The hydro plants would be controlled by the reactor operator and designed against a single failure. Each hydro unit would have a rating corresponding to about 70 Mw electric. The hydro station power would be delivered to the reactors by either an overhead 230 kv line through the switchyard or by a 13.8 kv underground line, either hydro feeding either line. Power transmitted by the underground line would be limited to about 10 Mw by the transformer. This would be enough to handle minimum safeguards on all units simultaneously but will require that reliable load shedding equipment be incorporated in the design. The hydro plant equipment and dams are designed to withstand the maximum hypothetical earthquake ground acceleration of o.1g.

Each unit is essentially independent of the other and is provided with its own startup equipment located within separate cubicles within the Keowee control room. The initiation of startup is accomplished by control signals from the reactor unit control room areas. Normal startup of either unit is by operator action while emergency startup is automatic. Both units are started automatically and simultaneously if the external transmission system is lost or if engineered safety features action is required.

The Keowee hydro units can pick up emergency loads from dead start in 23 seconds, which is adequate under design basis accident conditions. If tripped off line at full power due to a system disturbance, each unit can pick up full load in 7 seconds. Each hydro unit's voltage regulator is equipped with a volt-per-cycle limiting feature which permits it to accept load at the outset and thus drag the loads up to full speed in synchronsim with its own acceleration. This serves to reduce the time required for the initiation of engineered safety features action.

The hydro plants are started by opening gates which are powered by hydraulic accumulators. Stored hydraulic energy is sufficient for three full opening and closing cycles. Control circuits for emergency actuation of the accumulators will be redundant. A shear pin arrangement within the mechanical portion of the gate drive will release a jammed or otherwise fouled gate from the others. The protection system on the hydro plant will be limited to only those parameters that will prevent generation of power, such as generator insulation breakdown or loss of field.

We believe that the proposed hydro plant design will fulfill the requirement for a reliable and redundant source of on-site emergency power.

The applicant has estimated that the hydro plants will be dewatered and out for maintenance for a brief period of inspection of the hydro waterwheels each year and that major repairs are expected on a 7 to 10 year frequency. Since the penstock, a concrete lined rock tunnel, is common to both units, both units will be simultaneously unavailable for use during these periods. The hydro plants can be restored to operation within 2 hours during an inspection and within 6 hours during repairs to the penstock.

During the periods of hydro plant maintenance emergency power can be fed to the site through the 100 kv transmission line which can be made separate from the external grid and which is designed for loadings in excess of earthquake requirements. Power would be supplied by one of three 30 Mwe gas turbines located at Duke's Lee Station 30 miles from the site. Since the line could be separated from the external grid and a gas turbine run continuously in a no-load condition, we believe that this constitutes a satisfactory power source during the brief periods of hydro outage.

After the second and third reactor units have been added to the site each unit can serve as an additional power source since 100% load rejection capability will be provided in each unit by venting of secondary steam to the atmosphere in case of loss of the external grid. The applicant has stated its intention t .est this feature on each unit.

The water source for cooling after a loss of coolant accident, in addition to the core flooding tanks (accumulators), will be the 350,000 gallon borated water storage tank provided for each unit. After about 45 minutes, water in the containment is recirculated and cooled by service water taken from the intake structure.

We believe that the proposed emergency power and water sources will provide assurance that a loss-of-coolant accident could be coped with even in the event of loss of the system grid coincident with the accident.

6.3.2 Blackout of AC Power

Because of the topography of the site, heat can be removed from the condensers by gravity flow from the intake canal through the condenser to the tailrace of the hydro plant. Decay heat removal is thus possible for an extended period of time without reliance on off-site or on-site a.c. power.

Decay heat is removed from the core to the steam generators by natural circulation in the primary loop. The water in the secondary side of the steam generator is boiled and transferred to the turbine condensers. Additional feedwater is supplied to the steam generator by an emergency steam driven turbine pump which draws from the water in the hotwell of the main condenser. Heat is removed from the main condenser by the gravity flow scheme discussed above. Controls and auxiliary systems for the emergency steam driven feedwater pumps are operated from the station batteries.

This capability provides an added flexibility in the plant shutdown equipment and we agree with the applicant that it adds to the safety of the reactor units.

6.3.3 Rapid Drawdown of Lake Keowee

The applicant has agreed to provide alternate water and power sources sufficient to ensure an orderly shutdown of the plant in case of a rapid drawdown of Lake Keowee. A water source would be assured by constructing an underwater weir in the intake canal which would retain a large amount of water to serve as a cooling pond.

Heat transferred by the emergency steam driven feedwater pump to the condenser would be removed from the condenser by electrically driven pumps supplying water from and returning it to the cooling pond. The above mode of operation is not required immediately since enough condensate storage is available to remove decay heat for about 20 hours by venting secondary system steam to the atmosphere.

The power source for the electrically driven pumps would be either the normal system network or the 100 kv line fed by a gas turbine at the Lee Station. In the event of loss of system network power the gas turbine could be started and the

100 kv line separated from the station grid in the available 20-hour period before a.c. power is required.

We believe that the proposed alternate water and power sources will assure an orderly shutdown of the plant under the condition of a rapid drawdown of Lake Keowee.

6.3.4 Equipment Failure During Normal Shutdown

Decay heat after a reactor scram will normally be removed by natural circulation in the primary system transferring core heat to the boiling secondary system. Feedwater is supplied to the secondary by a single steam-driven emergency feedwater pump. After the second and third units have been added at the site, additional reliability will be obtained by cross-connecting the outlets of the feedwater pumps in all units. Any two of the three pumps, each sized at 5% of full flow, will supply enough feedwater to all six steam generators.

In addition, feedwater can be supplied through the emergency feedwater line by electrically driven condensate booster pumps. The secondary side would be depressurized manually to less than 700 psi through the turbine bypass system to allow this mode of operation. Sufficient power is available from the emergency power source (the hydro units) to operate the condensate booster pumps on emergency power.

Heat can be removed from the main condensers by either the normal intake pumps, the emergency electrical pump or by gravity flow.

After system depressurization, either from an accident or normal shutdown, decay heat is removed through the low pressure injection system to the decay heat removal coolers. Any one of the three low pressure injection pumps in conjunction with either of the heat exchangers in each unit has enough capacity to remove decay heat from the core. The secondary side of each heat exchanger is cooled by low pressure service water. Three low pressure service water pumps will be shared between Units 1 and 2 each capable of supplying the normal operational service water requirement. In addition, the two full capacity pumps in Unit 3 will be cross-connected to provide added flexibility. A single equipment failure will therefore not impair the decay heat removal capability of any unit.

As discussed above, we believe that the proposed design for the Oconee Nuclear Station will provide reliable and flexible emergency sources of water and power to cope with a wide range of abnormal conditions.

6.4 Penetration Room Ventilation System

The penetration room ventilation system is provided to maintain a small negative pressure in the penetration room under accident conditions and to filter fission product leakage from penetrations. This is accomplished by two blower units in series with two filter banks which are valved so that either blower can be used with either filter. The charcoal filters operate in a dry atmosphere and we believe that the 90% filter efficiency claimed by the applicant can be attained by the final design.

All penetrations except the steam lines and the personnel and equipment hatches pass through the penetration room. Leakage around the steam lines is not expected since they are welded to the containment liner. The outer door of each hatch will be double-gasketed and a small line run from between the gaskets to the penetration room.

The influence of this system on the total containment leakage is discussed in Section 4.6 of this report. We believe that adequate redundancy and reliability can be provided in the final design of this system to allow credit for its use as an engineered safety feature which is operable during an accident.

7.0 Radioactive Waste Control

The sizing of the waste handling and storage equipment has been performed on the basis of continued reactor operation with clad defects in 1% of the fuel rods. The primary system is maintained at high water purity and radioactive wastes removed by the chemical purification system. A small stream is bled from the primary system, reduced in pressure and temperature by the letdown coolers and passed through the demineralizer as necessary and then routed to the letdown storage tank. Makeup to the primary system is provided by pumping the water in the letdown storage tank through the seal water or high pressure injection system. Addition or dilution of borated water is also accomplished by this system by feeding the letdown storage tank from the chemical addition system.

Liquid wastes are collected from the demineralized sluice or other miscellaneous sources, monitored and, if necessary, held for decay. Low concentration wastes are discharged to the Keowee hydro tailrace. The applicant has stated that additional dilution may be obtained by opening the gates on the dam. We will review the amount of dilution to be allowed in the tailrace at the operating license stage.

Solid wastes will be temporarily pending shipment from the site in containers approved for the purpose.

Gaseous wastes will be monitored and diluted and release or stored in waste gas holdup tanks until decayed.

The applicant has indicated monitoring of all likely sources of effluent release and has performed an analysis on the liquid waste disposal systems to show that multiple equipment failures and operator errors would be required to allow undetected discharge of radioactive wastes. An analysis was performed, as described in

Section 8.1 of this report, to show that even if the wastes stored at the site under failed-fuel conditions were discharged, the public drinking supplies would not be endangered.

We believe that the waste disposal system described by the applicant will effectively control radioactive wastes generated on the site. The release limits to be set will be reviewed by the staff at the operating license stage.

8.0 ACCIDENT ANALYSIS

8.1 Incidents

A number of operational transients were considered by the applicant including rod withdrawal during startup and from power, moderator dilution, and loss-of-coolant flow, and no radiological hazard was found to result. The ACRS has recommended, and we concur, that further evidence should be obtained concerning the ability of the fuel to withstand expected transients at the end of its design lifetime.

A "guillotine" rupture of a steam generator tube was postulated and fission product release from primary system water with fission product inventory corresponding to 1% failed fuel through the turbine main condenser resulted in doses less than 10 CFR Part 20 limits at the site boundary. The release of activity from a waste gas tank failure after operation with one percent failed fuel is calculated to be within 10 CFR Part 20 limits. Limits on radioactive waste concentration will be set at the operating license stage.

An accidental discharge of 20,000 gallons of liquid waste at activity levels corresponding to continued operation with 1% failed fuel was postulated from the waste holdup tank. The spill was assumed even though multiple equipment failures and operator errors would be required before radioactive effluent could be released. The calculations, which utilize conservative dilution factors, show that accidental discharge of operational stored wastes would result in doses below 10 CFR Part 20 limits.

An analysis was also performed to illustrate that an extended (and undetected) release of wastes collected after the maximum hypothetical accident must be postulated before 10 CFR Part 100 guideline doses would be exceeded. The doses calculated assume no corrective action at the public water intakes.

The only significant hazard to the public drinking supply would be an accidental release of stored wastes after a major accident when (it is expected) comprehensive monitoring programs would be undertaken and any accidental release would be detected. We believe that the analysis presented illustrates the potential magnitude of the problem and the corrective measures which are available and that the accidental release of liquid waste would not result in excessive exposure to the public.

8.2 Steam Line Break

A steam line failure was analyzed which resulted in the release of the fission products contained in the secondary system (which are accumulated due to a minor tube leakage in the steam generator). The doses from this accident were calculated to be within 10 CFR Part 20 limits. A steam line break coincident with multiple tube failures was also analyzed.

Break of a main steam line during operation would cause cooldown of the primary system due to flashing of the secondary system inventory. The flashing of the relatively low feedwater inventory would cause a decrease in primary coolant temperature of about 40°F at the end-of-life conditions when the maximum negative moderator temperature coefficient is present. This, combined with the large increment of reactivity held by the control rods prevents an immediate secondary criticality. Injection of boron from the high pressure injection system and core flooding tanks is calculated to maintain the core in a shutdown condition.

The applicant's calculations indicate a maximum clad temperature of about 750°F during the steam line break transient and no clad failure is postulated which would result in additional release of fission products to the primary system water. Dose calculations were therefore based on release of fission products contained in the primary system water (at a concentration based on extended operation with 1% failed fuel) until the primary system temperature reached 200°F. The applicant has calculated that a thyroid dose of about 30 rem would result at the site boundary for a complete break of one tube, assuming no plateout of halogens. Our calculations indicate a thyroid dose less than 10 CFR Part 100 guidelines even for a complete blowdown of the primary system.

8.3 Rod Ejection Accident

The ejection of a control rod from the core is postulated to occur as a result of a break in the pressure housing of the control rod drive. The maximum reactivity increment that could be inserted corresponds to the worth of the ejected rod in the core prior to the accident. The applicant has stated that the maximum worth of a control rod at full power is 0.2% $\Delta k/k$ and the maximum worth at source level, 0.5% $\Delta k/k$. The parametric study presented showed the effect of ejected rods worth 0.1% to 0.7% $\Delta k/k$ for both the full power and source level cases.

For the ejection of a 0.2% rod from full power the maximum enthalpy in the hottest fuel rod was calculated to be 157 calories per gram (cal/gm). The applicant's sensitivity analysis, which arbitrarily increased the worth of the ejected rod, indicates that ejection of a rod worth 0.6% from full power would result in a hot spot enthalpy of about 200 cal/gm. This is still below the fuel melting temperature and no significant rapid energy release to the water is expected.

An ejection of a 0.5% $\Delta k/k$ rod at source power was calculated by ejecting a 1% rod with the core initially 0.5% $\Delta k/k$ subcritical. The results of the analysis indicate a resultant peak power level of about 39% full power.

Parameters varied in the sensitivity analysis included rod worth, Doppler coefficient, moderator coefficient and trip delay time. An analysis was also performed to obtain an estimate of the margin to failure of the vessel. Vessel failure was estimated to occur for ejection of a rod worth of 2% $\Delta k/k$. The applicant also stated that core internals would not be damaged by ejection of a 1% rod since no fuel melting was calculated for that rod worth.

An environmental analysis was also performed and resulted in doses well within Part 100 guidelines.

We believe that the results of the applicant's analyses show that vessel failure would not occur as a result of an ejected rod of the worths calculated for this design and that damage to core internals would not be expected at the peak enthalpy values calculated.

8.4 Loss of Coolant Accident

The applicant has proposed an emergency core cooling system (including core flooding tanks) which is designed to protect the core for the full spectrum of primary system break sizes which would result in a loss of coolant up to the double-ended rupture of the largest pipe in the system. The applicant plans core cooling analyses for the spectrum of break sizes. All of these have not yet been completed and the calculation of blowdown forces has not been completed but analyses are underway in both cases and will be looked at before the operating license stage. As discussed in Section 6.1 of this report, we believe that the representative analyses already done provide assurance to support the construction permits for these units. As discussed in Section 4.8 of this report the applicant performed parametric analyses to establish the peak accident pressure in the containment which we believe to be satisfactory.

As recommended by the ACRS, further evidence will be required at the operating license stage that fuel clad failure will not affect significantly the ability of the injection systems to cool the core.

The applicant has calculated the environmental consequences of this accident (and we have duplicated the analysis of the data) for the expected course of the accident and for a "design basis accident" in which 100% of the noble gases and 50% of the halogens and 1% of the particulates are assumed to be released to the reactor containment. The reactor building leak rate was assumed constant for 24 hours at 0.5%/day and 0.25%/day for the remaining duration of the accident. As discussed in Section 4.6 of this report, one-half of this leakage is assumed to pass through the penetration room filters where the halogens are removed with a conservatively estimated 90% filter efficiency.

The meteorological model used as a basis for dose calculations is based on the drainage of air down the river valley with no loss from the valley. It is assumed that Pasquill Type F diffusion conditions prevail during such times, and that the wind speed is low.

For the first 2 hours the wind speed is taken to be 1 m/sec, and the diffusion is calculated at the site boundary. At this point, there are two hills which confine the valley so that the total cross-sectional area below their tops (775 feet mean sea level) is about 13,500 square meters. For a plume uniformly distributed in such a space, the equivalent diffusion factor (X/Q) is 7.4×10^{-5}.

Doses for the duration of the accident were determined at the low population distance of about 6 miles. A narrow point in the river valley exists 5.8 river miles from the site boundary in the vicinity of Clemson, S. C., and the valley cross section at this point is about 101,400 square meters below elevation 820 feet. It is noted that on the way to this point, the Keowee River is joined by a similar sized stream, the Little River. By comparing the valley cross-section at this confluence, it can be shown that the air flowing down the Keowee valley is diluted by a factor of about 0.48 at this point.

Using a wind speed of 1.5 meters/sec for the first 24 hours, the value of X/Q in the vicinity of Clemson is 3.16×10^{-6}. For the remaining 29 days over which the dose was calculated, the same conditions are assumed to prevail 35% of the time, giving a value of X/Q of 1.10×10^{-6}. These values of diffusion factors were used as appropriate in the dose calculations for all accidents.

The two hour dose at the site boundary for the "design basis accident" using the above meteorological model is about 250 rem to the thyroid and 2 rem whole body. The thirty day dose at the low population distance was calculated to be 150 rem to the thyroid by the applicant. However, this did not include the initial 24 hour dose. The total of the "thirty day" and "24 hour" doses would be 220 rem to the thyroid and about 1 rem whole body.

The above doses are all within the guidelines listed in 10 CFR Part 100 which recommends less than 300 rem to the thyroid and 25 rem whole body at the exclusion radius for the two hour dose and less than 300 rem to the thyroid and 25 rem whole body at the low population distance over the course of the accident.

We believe that the reactor site conforms to the Commission's guidelines and therefore is acceptable.

9.0 MULTIPLE UNIT INSTALLATION

Units 1 and 2 share a number of auxiliary systems although no engineered safety features components, except service water pumps, are shared. The applicant has described Unit 3 as being separate from the other reactors except for mutual sharing of conventional plant utility systems. Each unit has two battery banks which feed a bus for that unit. The battery buses are cross-connected between units by a breaker system.

Systems shared between Unit 1 and Unit 2 are listed in table below along with similar components which exclusively serve Units 1, 2 or 3.

Component or System	No. of Components		
	Unit 1 (or Unit 2) Exclusively	Units 1 & 2 Shared	Unit 3 Exclusively
(1) Purification demineralizers	1	1	2
(2) Component coolers	1	1	2
(3) High pressure service water pumps	-	3	2
(4) Low pressure service water pumps	-	3	2
(5) Recirculated cooling water pumps	-	2	2
(6) Recirculated cooling water heat exchangers	-	2	2

In items (1) through (4) in the above table, one component is sized to handle one unit. In items (5) and (6) one component is sized to handle two units in the shared systems. We are also informed that a cross-tie will be installed between the outlets of the low pressure service water pumps of all three units to provide added flexibility.

In Unit 3 the Chemical Addition and Sampling System, Spent Fuel Storage Pool and Radioactive Waste Disposal System will be sized for a single unit while these systems are shared between Unit 1 and Unit 2 and sized accordingly.

The vital bus system will be shared between the three units but will be designed such that no single failure will interrupt protective electrical systems as discussed in Section 5.0 of this report.

We believe that sharing of the systems described above between units is acceptable and will not compromise the safety of the three units by increasing the probability or consequences of an accident.

10.0 RESEARCH AND DEVELOPMENT

The applicant has identified a number of areas in which research and development is required as listed in items 1 through 4 below. We believe that item 5, core cooling, and item 6, xenon oscillations, should also be included in the development program. Items 1 and 4 also contain considerations beyond those initially specified by the applicant.

(1) Once-through steam generator

Steady-state conditions and operational transients will be investigated in conjunction with the control system to be used. We believe that vibration tests, including steam generator response to primary system blowdown, should be investigated and the thermal response to both primary and secondary blowdowns determined.

(2) Control rod drive unit test

The prototype tests outlined by the applicant to be conducted under operating temperature, pressure, flow and water chemistry should provide information on the operability and reliability of the system.

(3) In-core neutron detectors

The self-powered units are currently under test in the Big Rock Point Nuclear Power Plant.

(4) Thermal and Hydraulic Programs

The applicant has proposed scaled flow distribution tests on the vessel and internals and rod bundle tests to determine local mixing and flow effects as discussed in Section 3.4 of this report. We believe that further experimental and analytical work must be done to determine the limiting heat fluxes at various positions within the fuel bundle if the design is to be based on the B&W heat transfer data.

(5) We also believe that the applicant should include core cooling in the development program. Specifically (a) the completion of the analysis of the spectrum of break sizes in the loss-of-coolant accident, (b) the development of the analytical techniques for determining blowdown forces on reactor internals, and (c) demonstration that the injection coolant will cool the core including core bypass or formation of a vapor lock.

(6) Xenon Oscillations

The applicant should further develop analytical techniques to determine whether xenon oscillations can occur. If oscillations are possible a system for controlling the oscillations will also have to be developed.

11.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards, by letter to Chairman Seaborg dated July 11, 1967, reported on the Oconee Nuclear Station, Units 1, 2 and 3. A copy of this letter is attached as Appendix A. The letter contained a number of comments and recommendations which we are implementing as noted in the appropriate sections of this report. The items mentioned will be resolved prior to the issuance of an operating license to the satisfaction of the staff and the ACRS.

The report concluded "...the Committee believes that the proposed Oconee Nuclear Station can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public."

12.0 TECHNICAL QUALIFICATIONS

The applicant, Duke Power Company, has extensive experience in the design, construction and operation of electric generating plants. Duke personnel have been involved with nuclear power generation since the early 1950's, culminating in company ownership of 34% of the Carolinas-Virginia Tube Reactor at Parr, South Carolina. Detailed qualifications of Duke and its principal nuclear personnel can be found in Appendix 1A of the PSAR.

The nuclear steam system supplier, Babcock & Wilcox, designed and constructed the N.S. SAVANNAH and Indian Point 1 reactors, as well as the ATR and several research reactors. In addition, B&W is one of two companies presently supplying large reactor pressure vessels.

The Bechtel Corporation has served as the architect-engineer on many nuclear projects over the last few years. The latest of these include the San Onofre, Turkey Point, Palisades, and Point Beach reactors.

On the basis of the above considerations, and based upon our evaluation of the responsible personnel, we believe that the applicant and its contractors, B&W and Bechtel are suitably qualified to design and construct the proposed facility.

13.0 CONFORMANCE TO THE GENERAL DESIGN CRITERIA

In November 1965, the Commission published its General Design Criteria for Nuclear Power Plant Construction Permits. In the PSAR, Duke has evaluated the units considering these criteria. However, on July 11, 1967, the Commission published in the Federal Register its revised General Design Criteria taking into account comments received on the initial criteria and further development of the criteria by the regulatory staff. Anticipating the publication of revised criteria and because we were involved in the formulation and development of the revisions, we have evaluated the Duke application against the revised criteria and have concluded that the proposed units conform to the intent of the revised criteria. Recognizing that the proposed revised criteria may be modified as a result of comments by interested parties during the 60 day period provided for this purpose, we intend to review the proposed units at the operating license stage in light of the criteria as formulated at that time.

14.0 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. We find nothing in the application or otherwise to suggest that the applicant is owned, controlled or dominated by an alien, a foreign corporation or a foreign Government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with paragraph 50.33(j) of 10 CFR Part 50. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material from military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

15.0 CONCLUSIONS

Based on the proposed design of the Duke Power Company's Oconee Nuclear Station, Units 1, 2 and 3, on the criteria, principles and design arrangements for systems and components thus far described, which include all of the important safety items, on the calculated potential consequences of routine and accidental release of radioactive materials to the environs, on the scope of the development program which will be conducted, and on the technical competence of the applicant and the principal contractors, we have concluded that, in accordance with the provisions of paragraph 50.35(a), 10 CFR Part 50 and paragraph 2.104(b) 10 CFR Part 2:

1. The applicant has described the proposed design of the facilities, including, the principal architectural and engineering criteria for the design and has identified the major features or components for the protection of the health and safety of the public;
2. Such further technical or design information as may be required to complete the safety analysis and which can reasonably be left for later consideration, will be supplied in the final safety analysis reports;
3. Safety features or components, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components;

4. On the basis of the foregoing, there is reasonable assurance that (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facilities and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, the proposed facilities can be constructed and operated at the proposed location without undue risk to the health and safety of the public;
5. The applicant is technically qualified to design and construct the proposed facilities; and
6. The issuance of permits for the construction of the facilities will not be inimical to the common defense and security or to the health and safety of the public.

APPENDIX A

(78)

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

UNITED STATES ATOMIC ENERGY COMMISSION

WASHINGTON, D.C. 20545

JUL 11 1967

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

Subject: REPORT ON OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

Dear Dr. Seaborg:

At its eighty-sixth meeting, on June 8-10, 1967, and its eighty-seventh meeting, on July 6-8, 1967, the Advisory Committee on Reactor Safeguards reviewed the proposal of the Duke Power Company to construct the Oconee Nuclear Station, Units 1, 2, and 3, at a site near Clemson, South Carolina. This project was reviewed by an ACRS Subcommittee on May 2, 1967, at the site and at Clemson, and on May 31 and June 23, 1967, in Washington, D. C. The Committee had the benefit of discussions with representatives of the Duke Power Company and its consultants, The Babcock and Wilcox Company, Bechtel Corporation, and the AEC Regulatory Staff, and of the documents listed.

Each unit of the Oconee Station includes a pressurized-water reactor rated at 2452 MWt. Each unit is to be provided with an emergency core cooling system (ECCS), including two core flooding tanks, three high-pressure injection pumps, and three low-pressure injection and recirculation pumps. The applicant proposes not to operate a unit with a core flooding tank valved off. The Committee recommends that the Regulatory Staff review the detailed design of the ECCS and the analysis of its performance for the entire spectrum of break sizes, as soon as this information is available. In this respect:

1. The Regulatory Staff should review analyses of possible effects, upon pressure-vessel integrity, arising from thermal shock induced by ECCS operation.*

2. The effects of blowdown forces on core and other primary system components should be analyzed more fully as detailed design proceeds.*

3. Further evidence should be obtained to show that fuel-rod failure in loss-of-coolant accidents will not affect significantly the ability of the ECCS to prevent clad melting.*

4. The applicant has proposed adding swing-check valves in the core barrel to ensure obtaining adequate height of cooling water in the core under all circumstances of ECCS operation. This feature should be further reviewed to ensure that no new problems are introduced.

5. The applicant will explore further possibilities for improvement, particularly by diversification, of the instrumentation that initiates ECCS action.

Emergency power sources for the ECCS and other safeguards are: (a) the other Oconee units (each unit can withstand and will be tested to withstand instantaneous loss of load without a reactor trip or a turbine trip); (b) two hydroelectric units at Keowee station less than one mile away, with independent overhead and underground transmission lines; and (c) a gas-turbine unit thirty miles away with independent transmission line, transformer, and switchyard -- all in addition to the usual multiple ties to the power transmission grid. The applicant stated that switching and sequencing of sources, buses, and loads would be such that no single failure would impair system availability.

The applicant stated that the entire primary system of each unit, including the inside and outside of the reactor vessel, will be accessible for inspection over the life of the plant.

The Committee continues to emphasize the importance of quality assurance in fabrication of the primary system as well as inspection during service life, and recommends that the applicant implement those improvements in primary system quality that are practical with current technology.*

The moderator coefficient of reactivity is calculated to be positive at the beginning of core life, for the first core. The applicant is making detailed studies of the effect of this coefficient on the course of postulated accidents; if necessary, the coefficient will be made more negative by the addition of solid poison shims to the core.

Further evidence should be obtained concerning the ability of the fuel to withstand expected transients at the end of its anticipated lifetime.*

The applicant is investigating further the stability margin for xenon oscillations.

The containment structures are similar to those for the Turkey Point reactors previously reviewed. Consideration should be given to improved inspection of welds in the steel liner of such containments, because an acceptance pressurization test does not stress the liner to postulated accident conditions.

JUL 11 1967

Power for the reactor protection systems and the safeguards protection systems for all three units is provided by a system of six batteries, static inverters, and six buses. The same batteries, via other inverters and buses, provide power to the control systems for all three units. The Committee urges the applicant to review the design of these systems with respect to independence of each unit from troubles in the others.

The applicant proposes to construct a submerged earthen weir in the intake canal to assure a heat sink in the event Keowee Reservoir is drawn down excessively. The Committee believes that careful attention is necessary in the design and construction of this weir to avoid hydraulic erosion and soil instability, particularly in case of rapid drawdown.

The Advisory Committee on Reactor Safeguards believes that the items mentioned above can be resolved by the applicant and the Regulatory Staff during construction of the reactors. On the basis of the foregoing comments, the Committee believes that the proposed Oconee Nuclear Station can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

ORIGINAL SIGNED BY
N. J. PALLADINO

N. J. Palladino
Chairman

*The Committee believes that these matters are significant for all large water-cooled power reactors, and warrant careful attention.

References:
1. Duke Power Company, Oconee Nuclear Station, Units 1 and 2, Preliminary Safety Analysis Report, Volumes I and II, undated, received December 5, 1966.
2. Amendment No. 1, dated April 1, 1967.
3. Amendment No. 2, dated April 18, 1967.
4. Amendment No. 3, dated April 29, 1967.
5. Amendment No. 4, dated May 25, 1967.
6. Amendment No. 5, dated June 16, 1967.

APPENDIX B

CHRONOLOGY - REGULATORY REVIEW OF THE

OCONEE NUCLEAR STATION

1.	November 28, 1966	Submittal of Preliminary Safety Analysis Report.
2.	January 18, 1967	Meeting with applicant to review greater detail required.
3.	February 14 and 15, 1967	Meeting with applicant to discuss technical aspects of plant design.
4.	March 23, 1967	Questions issued to the applicant requesting plant design and safety features information.
5.	April 1, 1967	Submittal of Amendment No. 1: partial answer to staff questions of March 23, 1967.
6.	April 18, 1967	Submittal of Amendment No. 2: complete answer to staff questions of March 23, 1967.
7.	April 27 and 28, 1967	Meeting with applicant to discuss information submitted in Amendments Nos. 1 and 2.
8.	April 29, 1967	Submittal of Amendment No. 3: includes application for construction permit for Unit No. 3 at Oconee site.
9.	May 2, 1967	ACRS Subcommittee meeting with staff and applicant at the site.
10.	May 8, 1967	Meeting with applicant to discuss accident meteorology and use of hydro plants as emergency power units.
11.	May 11, 1967	Questions issued to the applicant.
12.	May 25, 1967	Submittal of Amendment No. 4 answers staff questions of May 11, 1967
13.	May 31, 1967	ACRS Subcommittee meeting with staff and applicant to discuss technical aspects of design.
14.	June 8, 1967	ACRS meeting discusses technical aspects of design.

15. June 13, 1967 — Staff meeting at Cowan's Ford Hydroelectric Station to discuss reliability of hydro-power.

16. June 16, 1967 — Submittal of Amendment No. 5 supplies information on topics raised at June 8 at the ACRS meeting.

17. June 23, 1967 — ACRS Subcommittee meeting with staff and applicant to discuss technical aspects of design.

18. June 29, 1967 — Staff meeting with applicant to discuss loss-of-coolant analysis.

19. July 7, 1967 — ACRS meeting discusses technical aspects of design.

20. July 11, 1967 — ACRS Report on Oconee Nuclear Station, Units 1, 2, and 3.

Harold L. Price
Director of Regulation

MAY 11 1967

Original signed by
Milton Shaw

Milton Shaw, Director
Division of Reactor Development & Technology

HAZARDS SUMMARY REPORT

RDT:NS:S146

Reference is made to the letters of March 6 and April 19, 1967, from the Division of Reactor Licensing, to the Environmental Science Services Administration requesting comments on the following safety analysis reports respectively:

Peach Bottom Atomic Power Station Units No. 2 & 3
Philadelphia Electric Company
Preliminary Safety Analysis Report

Oconee Nuclear Station Units 1 & 2
Duke Power Company
Preliminary Safety Analysis Report
Amendment #2 dated April 13, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and by copy of this memorandum, we are transmitting their comments to Mr. P. Morris, Director, DRL.

cc: P. Morris, Director, DRL, w/attach. (Orig. & 1 cy.)

1677-19

Comments on

Oconee Nuclear Station Units 1 and 2
Duke Power Company
Preliminary Safety Analysis Report
Amendment #2 dated April 18, 1967

Prepared by

Environmental Meteorology Branch
Institute for Atmospheric Sciences
May 3, 1967

The critical off-site location with regard to high concentrations would appear to be the Keowee River valley to the east and southeast of the site. As indicated by a statement in the applicant's revision, the terrain in this direction will modify the drainage flow direction to that following the Keowee River. At the 1 mile site boundary the valley is confined by two hills at a height of 778 and 773 ft Mean Sea Level and separated by a horizontal distance of about 2000 feet. Thus, with the valley floor at 660 ft MSL, the cross-sectional area at the height of these hills is about 1×10^5 square feet. The valley remains restricted in a similar fashion farther downstream.

Depending upon the assumptions used, the following concentrations could be attained at the site boundary of 1600 meters:

Assumption	χ/Q (sec m^{-3})
Type F, 1 m/sec, no bldg. effect	3.4×10^{-4}
Type F, 1 m/sec, bldg. effect C = .5, A = 5180 m^2	1.6×10^{-4}
Type F, 1 m/sec, bldg. effect C = 1.0, A = 5180 m^2	1.1×10^{-4}
Valley Confinement, u = 1 m/sec, Area = 10^5 ft^2	1.0×10^{-4}
Type F, u = 1.9 m/sec, bldg. effect C = 1.0, A = 5180 m^2	6.0×10^{-5}

Since very little site meteorological data are available, it remains to be seen whether a wind speed of 1.9 as opposed to 1.0 m/sec is more appropriately conservative. The applicant has chosen to use the least conservative assumption listed above, resulting in a concentration a factor of 5 lower than T.I.D. 14844 meteorology, which does not give credit for building turbulence effects.

In addition to the meteorological measurements planned for the microwave tower on a hill to the west of the reactor building complex, it is also necessary to measure air flow in the valley to the east if information on

1477-19

the drainage flow is to be obtained. However, any measurement program started now will not truly reflect the conditions which will exist when Keowee Dam is completed and Lake Keowee to the west and north of the site has reached its full pond elevation of 800 feet MSL, which is 4 feet above plant grade.

In summary, atmospheric diffusion rates in the general area of the site are expected to be somewhat lower on the average when compared to other locations in the United States. With the construction of Keowee Dam and its resevoir, the primary nighttime, inversion transport is expected to be down the Keowee River valley. Assuming an effluent will be confined to the valley to a height of about 100 feet at the site boundary, a concentration of 1×10^{-4} sec m^{-3} would result with a wind speed of 1 m/sec and uniform mixing within the valley. This would be our best, estimate at the moment, of a controlling concentration.

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA FPMR (41 CFR) 101-11.6

(86)

UNITED STATES GOVERNMENT

Memorandum

TO : Peter A. Morris, Director
Division of Reactor Licensing

DATE: JUN 16 1967

FROM : Milton Shaw, Director
Division of Reactor Development & Technology

SUBJECT: HAZARDS SUMMARY REPORT

RDT:NS:S198

Reference is made to the letter of May 31, 1967, from your Division, to the Environmental Science Services Administration requesting comments on the following safety analysis report:

Oconee Nuclear Station Units 1, 2 and 3
Duke Power Company
Preliminary Safety Analysis Report
Amendment #4 dated May 25, 1967

Review by the Environmental Meteorology Branch, Air Resources Laboratory, ESSA, has now been completed and their comments are attached.

Attachment:
Comments (Orig. and 1 cy.)

Comments on

Oconee Nuclear Station Units 1, 2 and 3
Duke Power Company
Preliminary Safety Analysis Report
Amendment #4 dated May 25, 1967

Prepared by

Environmental Meteorology Branch
Institute for Atmospheric Sciences
June 9, 1967

It is noted that the amendment offers a second meteorological diffusion model based on confined valley drainage under inversion conditions. It is our opinion that this second model, which is identical to the valley confinement assumption in our comments of May 3, 1967, is more realistic and is appropriately conservative.

APPENDIX D

(88)

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

JUN 19 1967

Mr. Harold L. Price
Director of Regulations
U. S. Atomic Energy Commission
4915 St. Elmo Avenue
Bethesda, Maryland 20545

Dear Mr. Price:

Transmitted herewith in response to the request of Edson G. Case, dated December 21, 1966, is a review of the geologic and hydrologic aspects of the license application of the Duke Power and Light Company Oconee Nuclear Station.

This review prepared by Henry W. Coulter and Eric L. Meyer of the U. S. Geological Survey has been discussed with members of your staff and we have no objections to your making it a part of the public record.

Sincerely yours,

W T Pecora

Director

Enclosure

- v-1094.

Oconee Nuclear Station, Duke Power Company
Units 1, 2, and 3
Oconee County, South Carolina
AEC Dockets 50-269, 50-270

HYDROLOGY

This review is based on information provided by the applicant in the Preliminary Safety Analysis Report and Supplements 1 though 4.

The proposed plant is approximately half a mile west of the Keowee River at an altitude of 794 feet above mean sea level (msl), and 129 feet above the flood pool elevation of the Hartwell Reservoir, ponded by a dam about 35 miles downstream from the site. Cooling water for the nuclear plant will be taken from Keowee Reservoir to be ponded by Keowee Dam (proposed) on the Keowee River and another dam on the Little River. This pool will cover the drainage divide between the two streams at one point upstream from the plant site; normal pool level is 800 feet msl. The low point of the ridge that separates Keowee Reservoir and the site is shown on topographic maps to be between 820 to 827 feet msl. The surcharge on the reservoir due to a maximum probable flood on Keowee River is given as 808 feet msl. Flooding of the site either by topping the ridge or by a rise of the river below Keowee Dam does not appear to be possible.

Liquid radioactive wastes from the reactor are to be discharged into the tailrace of the proposed Keowee Dam hydro-power units. Concentrations of waste radionuclides are computed in section 11 of the Preliminary Safety Analysis Report assuming dilution by the average discharge past Keowee dam. In determining the radioactive effluent limits set by 10 CFR 20, credit for this dilution should be allowed only if complete mixing of the wastes and river water occurs prior to entry of the combined discharge into unrestricted areas. Consideration should also be given to avoiding radioactive waste releases during periods when flow is low, as it would be when the hydro-power units are not in use.

GEOLOGY

The applicants geologic analysis of the Oconee Nuclear Station site presented in the Atomic Energy Commission Docket (50-269-270) was examined and compared with the available literature.

Because little is known concerning geological details of structural elements within the piedmont crystalline zone and most epicentral locations there are inexact, attempts to relate individual seismic events to specific structures within the zone cannot be made with any great degree of confidence. Hence it must be assumed that an earthquake equal in intensity to the largest earthquake that has been recorded anywhere throughout the zone may occur at any given locality within the zone.

In general, when dealing with weathered crystalline rocks throughout the Piedmont Province the following considerations apply. Where the interlocking texture of the saprolite is undisturbed it will stand in steep slopes in cut faces and will support considerable loads. However, when saprolite is reworked and used as fill it may be subject to failure even in moderate slopes. Thus any saprolite fill should be so located as to avoid the possibility of impingement on critical structures in the event of slope failure and the location of structures across a saprolite cut-fill interface should be avoided.

Boring data indicate that adequate foundation conditions on firm bedrock beneath the plant site should be encountered at anticipated elevations. Because of the irregularity of the weathered zone, the requirement for detailed modifications of footing design as is usual in standard engineering practice to ensure bearing on sound rock at all localities may be anticipated.

The applicants responses to questions 12.1 and 12.2 contained in amendment #4 to Docket Nos. 50-269, 50-270, and 50-287 indicate that foundation investigations and stability analyses comparable to those undertaken for other critical plant components have been, or will be, undertaken for the dams and intake structures necessary to provide cooling water supply to the plant.

APPENDIX E
(91)

U.S. DEPARTMENT OF COMMERCE
ENVIRONMENTAL SCIENCE SERVICES ADMINISTRATION
COAST AND GEODETIC SURVEY
~~WASHINGTON, D.C. 20235~~
ROCKVILLE, MD. 20852

JUN 16 1967

IN REPLY REFER TO: C23

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

Dear Mr. Price:

In accordance with your request, we are forwarding 10 copies of our report on the seismicity of the Seneca (Oconee County), South Carolina, area. The Coast and Geodetic Survey has reviewed and evaluated the information on the seismicity of the area presented by the applicant in their "Duke Power Company Oconee Nuclear Station Preliminary Safety Analysis Report," and find that it is satisfactory with respect to both distant and nearby earthquakes.

If we may be of further assistance to you, please do not hesitate to contact us.

Sincerely yours,

Raymond M. Stone
for James C. Tison, Jr.
Rear Admiral, USESSA
Director

Enclosure

REPORT ON THE SEISMICITY OF THE SENECA (OCONEE COUNTY), SOUTH CAROLINA AREA

At the request of the Division of Reactor Licensing of the Atomic Energy Commission, the Seismology Division of the Coast and Geodetic Survey has evaluated the seismicity of the area around the proposed reactor site near Seneca (Oconee County), South Carolina, and has reviewed the similar analysis by the applicant in their "Duke Power Company Oconee Nuclear Station Preliminary Safety Analysis Report." The applicant's seismicity report contains a complete listing of the earthquakes both distant and nearby, which may have affected the proposed site, and a detailed review of the geology within a few hundred miles of the site. Little is known, however, concerning the details of the geological structures in the area and the relationship of these structures to earthquakes. Because of this, the Survey believes that the largest earthquake recorded anywhere in the zone may occur along one of the faults near the site.

Based upon the review of the seismic history of the site and the surrounding area and the related geologic considerations, the Coast and Geodetic Survey agrees with the applicant that an acceleration of 0.05g on rock would be adequate for representing the ground motions from earthquake disturbances likely to occur within the lifetime of the facility. In addition, it agrees that an acceleration of 0.10g on rock would represent the ground motions from the maximum earthquake likely to affect the site. We believe this value would provide an adequate basis for designing protection against the loss of function of components important to safety. We also agree that an acceleration of 0.15g is an adequate basis for designing protection against the loss of function of components important to safety that are not located on rock.

U. S. Coast and Geodetic Survey
Rockville, Maryland 20852

June 14, 1967

APPENDIX F
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N A T H A N M. N E W M A R K

Consulting Engineering Services

1114 Civil Engineering Building
Urbana, Illinois 61801

Report to AEC Regulatory Staff

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE
OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

DUKE POWER COMPANY
(Dockets 50-269, 50-270, and 50-287)

by

N. M. Newmark and W. J. Hall

June 1967

ADEQUACY OF THE STRUCTURAL CRITERIA FOR THE OCONEE NUCLEAR STATION UNITS 1, 2, AND 3

by

N. M. Newmark and W. J. Hall

INTRODUCTION

This report concerns the adequacy of the containment structures, components, and dams for the three units of 2452 MWt each (874 MWe, net) for which application for a construction permit and operating license has been made to the U. S. Atomic Energy Commission (Dockets No. 50-269, 50-270, and 50-287) by the Duke Power Company. The facility is to be located on the shore of future Lake Keowee in Oconee County, South Carolina, 8 miles NE of Seneca, South Carolina.

The report is concerned specifically with the evaluation of the design criteria that determine the ability of the containment system to withstand a design earthquake acting simultaneously with other applicable loads forming the basis of the containment design. The facility also is to be designed to withstand a maximum earthquake simultaneously with other applicable loads to the extent of insuring safe shutdown as well as containment. The seismic design criteria for Class I equipment and piping are also reviewed herein, along with a review of the analyses of the dams which are required for impounding the required cooling water supplies. This report is based on information and criteria set forth in the Preliminary Safety Analysis Report (PSAR) and Supplements thereto as listed at the end of this report. We have participated in discussions with the AEC regulatory staff, in which many of the design criteria were discussed in detail.

DESCRIPTION OF THE FACILITY

Oconee Nuclear Station Units 1, 2, and 3 are described in the PSAR as pressurized water reactors for which the nuclear steam system and fuel cores are to be supplied by the Babcock and Wilcox Company, each designed for a power output of 874 MWe (net). The reactor coolant system for each unit consists of two closed reactor coolant loops connected in parallel to the reactor vessel, each provided with reactor coolant pumps and a steam generator. The reactor vessel will have an inside diameter of about 14 ft-3 in., a height of about 41 ft-9 in., and is designed for an internal pressure of 2500 psig, a temperature of 650^{o}F, and is made of SA-302 Grade B steel clad with Type 304 austenitic stainless steel.

Each of the reactor units is contained in a fully reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post-tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post-tensioning system. The flat foundation slab is conventionally reinforced with high-strength reinforcing steel, and the entire structure is lined with a 1/4 in. welded steel plate. The cylindrical part of each of the containment structures is approximately 116 ft inside diameter, has an inside height of 206 ft, vertical wall thickness of 3 ft-9 in., and a dome thickness of about 3 ft-3 in. The foundation slab is about 8-1/2 ft thick.

The PSAR on page 5-1 of Vol. I indicates that the design will in many respects be similar to that for the Florida Power and Light Company's Turkey Point Plant, Consumer Power Company's Palisades Plant, and Wisconsin-Michigan Power Company's Point Beach Plant. Although no stated details are given, we assume, then, that the cylindrical wall is to be provided with a system of hoop tendons which are placed in a 3-120^o system using six buttresses as anchorages with the tendons staggered so that half of the tendons at each buttress terminate at that buttress. In Appendix 5B it is noted that the prestressing will be post-tensioned, and unbonded, with the tendons encased in rigid steel conduit and corrosion protection provided by grease injected into the conduit under pressure. The answer to Question 9.2 of Supplement 1 indicates that the BBRV system of prestressing will be employed.

From Appendix 5E and Figure 5-1, it is noted that the welded steel liner will be at least 1/4 in. thick and made up of ASTM A-442 steel with angle-type anchors. It is noted that the liner plate will be thickened in the vicinity of penetrations.

Appendix 5B indicates that ASTM A-432 reinforcing steel will be used in the base slab, and that ASTM A-15 deformed bars will be employed in the cylinder wall, the domed roof, and around the openings to control shrinkage and tensile cracks. It is further noted in Appendix 5D that for large 14S and 18S reinforcing steel, Cadweld splices will be employed, and the Errata filed with Amendment 3 indicate that the tensile strength of the splices will equal or exceed 125 percent of the minimum yield strength of each grade of reinforcing steel as specified in the appropriate ASTM standard. We recommend that tack welding or other welding not be permitted for the A-432 bars in the foundation slab or elsewhere, to avoid the possiblity of fracture or other difficulties in achieving the required ductility of these reinforcing bars.

The geology is summarized in Appendices 2A and 2E; on page 2-9 of Vol. I of the PSAR it is stated that the structure will be founded on the normal Piedmont granite gneisses.

SOURCES OF STRESSES IN CONTAINMENT STRUCTURE AND TYPE 1 COMPONENTS

The containment structure is to be designed for the following loads: dead load of the structure; live loads (including roof loads, pipe forces, and reactor service crane loads); accident pressure load associated with loss-of-coolant accident of 59 psig; test pressure of 67.9 psig, and external-internal pressure differential of 3 psig corresponding to a drop of barometric pressure associated with a tornado with wind speeds of 300 mph (Supplement 4) as well as wind loading corresponding to 95 mph at 30 ft height.

On the basis of the information presented on page 5-5 of Vol. I of the PSAR, Appendix 5B, page 5B-4, and the answer to Question 8.5 of Supplement 1, and in accord with the USC&GS report (Ref. 3), the design earthquake will be characterized by a maximum horizontal ground acceleration of 0.05g and the maximum earthquake by a 0.10g horizontal ground acceleration. The structure is to be founded on firm basement rock.

COMMENTS ON ADEQUACY OF DESIGN

Seismic Design -- In connection with the selection of the design earthquake and the maximum earthquake, we agree with the values selected, and concurred in by the USC&GS, namely that of a basic design for a design earthquake of 0.05g and design for a maximum earthquake of 0.10g maximum horizontal ground acceleration.

On page 5B-4 of Appendix 5B, for the design earthquake of 0.05g, it is indicated that the horizontal and vertical acceleration will be taken as equal in intensity. We find no mention of this fact for the maximum earthquake but assume that the same situation will obtain there, and assuming that this is the case, we concur in this approach.

The proposed response spectra for various degrees of damping for the maximum earthquake are presented in answer to Question 8.5 of Supplement 1, for the design earthquake as part of Appendix 2B, and as modified in both cases by Supplement 4. We find no explanation for the basis of the selection of the ground motions ("ground motion spectra"), other than for the acceleration values which have already been agreed upon and which control in the high frequency band. We have compared the revised response spectra (Supplement 4) with those presented in report TID-7024 and find them to be substantially in agreement for frequencies above 0.2 cps, the region in which most, if not all, structural elements will fall. We believe that the applicable parts of the spectra are acceptable for design purposes.

The damping values to be employed are listed in answer to Question 8.4 of Supplement 1. We are in agreement with the damping values given therein with the further understanding, however, that the 5 percent damping value to be used for the maximum earthquake will be employed in the design in such a way that there will be a limitation on the deformations of the containment structure and its components. The general dynamic design approach outlined in answer to this same question appears acceptable to us both for the containment structure and for the piping.

The loading combinations for the containment design are presented in Appendix 5A. We are in agreement with the load factor expressions stated there for the case of the design and maximum earthquake. In reply to Question 8.1 of Supplement 1, however, it is noted that "the design criteria which will be applied to the above loading is that the deformation will be limited to values which will permit a safe and orderly shutdown." This statement provides no guide as to what the limitation on deformations will be, but we note that, in connection with the design of the liner, as described in Appendix 5E, a limitation of 0.5 percent strain has been set for the liner. On the assumption that the design will call for a reasonable limitation on ductility, i.e., on the order of not more than two or three times the gross yield deformation, and further that the liner deformation will be restricted as noted, we believe that the design approach for the maximum earthquake will be satisfactory in this respect.

In Appendix 5A it is noted that the polar crane is a Class I structure, and on the assumption that steps will be taken to insure that these cranes cannot be displaced from the rails during a design or maximum earthquake or otherwise topple to create damage which would prevent safe shutdown or impair the containment, we believe that this aspect of the design can be handled properly to make it satisfactory in all respects.

We have reviewed in some detail the design calculations for the dams as given in Supplement 1, and on the basis of the analyses we believe that the safety of the dams is satisfactory for the 0.1g earthquake on the basis that the dams are founded on basement rock as indicated. We call attention to one minor discrepancy in the method of analysis, based on that by N. M. Newmark (1965) given in the Rankine lecture to the Institution of Civil Engineers, in which the analyses given in Supplement 1 used static slip circles for the dynamic analysis. In general the slip circles for the dynamic analysis will be different from those for a static analysis, but in this case it appears that the results will be only slightly different, and that safety is achieved. It was noted in one case that one of the slices was on the verge of movement for the 0.1g earthquake; however, investigation reveals that such a slice would move only a small distance, and we believe that the function of the dam would not be impaired in any serious manner by such a minor slippage, should it occur. In summary, we believe that the dams can withstand the maximum earthquake stipulated, although the margin of safety against slippage, as noted above, is not great for the maximum hypothetical earthquake. As documented in Supplement 4 a natural pool of water will be provided for shutdown cooling in the event of unexpected dam failure.

General Design Considerations -- We have reviewed with care and interest the design criteria for the prestressed concrete reactor building as presented in Appendix 5C, and the elaboration on the development for handling the shear at yield loads as given in answer to Question 8.7 of Supplement 2. We are in agreement with the provisions there for handling principal concrete tension and the new recommendations for handling radial shear. In the event that further data become available on this matter prior to completion of the design stages, we trust that such information can be incorporated into the design, if this appears warranted.

Penetrations -- The design of the penetrations is described briefly in Section 5 of Vol. I of the PSAR, and elaboration is given in answer to Question 8.8 of Supplement 2. On the basis of the discussions presented therein, we concur in the approach that is described for this particular design.

Surveillance -- We find some information on the planned surveillance program in Section 5, and recommend strongly that a reasonable and sensible surveillance program be maintained throughout the life of the structure.

Piping and Other Type 1 Components -- We find discussion of the design of the piping presented in answer to Question 8.1 of Supplement 1 which refers to Appendix 5A as appropriate for the class of piping involved, with further amplification on the dynamic design provision as given in answer to Question 8.4. We are in general agreement with the approach proposed therein, but are still not sure exactly how the piping analysis will be carried out in the sense that is implied in the last paragraph on page 8.4-3 (4-1-67), which states that the stresses from the horizontal and vertical components acting simultaneously will be combined with the stresses due to weight, thermal and mechanical loads, and internal pressure, and in turn these stresses will determine the required yield strength of the piping systems. This does not completely answer the question of what limitations will be placed on the piping in terms of behavior under the maximum earthquake, particularly in terms of limitations on deformation. We recommend, for the specific materials used, that the deformations be limited to reasonable values which will preclude any difficulties with fatigue or fracture. Particular attention should be given to the piping at those places where it penetrates the containment, or to that piping which is required for safe shutdown in this regard. The same provisions apply to piping that will run from intake structures to the plant and which will be required for safe shutdown in the event of an earthquake or an accident.

Conclusions -- On the basis of the information presented, and in accord with the design goal of providing serviceable structures and components with a reserve of strength and ductility and which will provide for containment as well as safe shutdown, we believe that with approapriate attention to the design details as discussed in the body of our report, the design criteria outlined for the containment structures and Type 1 piping can provide an adequate margin of safety for seismic resistance.

1. "Preliminary Safety Analysis Report--Volumes I and II," Oconee Nuclear Station Units 1, 2, and 3, Duke Power Company, 1966.

2. "Preliminary Safety Analysis Report--Supplements 1, 2, 3, and 4," Oconee Nuclear Station Units 1, 2, and 3, Duke Power Company, 1967.

3. "Report on Seismicity of the Oconee Nuclear Station Units 1, 2, and 3," U. S. Coast & Geodetic Survey, Rockville, Maryland, June 16, 1967.

APPENDIX F
(104)

NATHAN M. NEWMARK
CONSULTING ENGINEERING SERVICES

1114 CIVIL ENGINEERING BUILDING
URBANA, ILLINOIS 61801

5 July 1967

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

Re: Submerged Weir in Intake Canal
Duke Power Co.
Docket Nos. 50-269, 50-270, 50-287

Dear Dr. Morris:

The following report is based on the studies made by Dr. A. J. Hendron, Jr. of our staff, and has been approved by Dr. W. J. Hall and myself. Our comments concerning the stability of the proposed submerged weir, as described in Supplement 5 of Amendment 5, by Item No. 11, dated 16 June 1967, follow.

(1) The factor of safety for static behavior under rapid drawdown, within the pressure levels of interest, as stated in the report is consistent with the shear strength of the material equal to or greater than that corresponding to a Mohr failure envelope with a cohesion intercept of 280 psf and an angle of internal friction of 30°, for consolidated-undrained conditions.

These material properties appear to be reasonable. However, the test technique used may not give completely conservative values since there may be a possibility of incomplete saturation of the samples because they were not "back-pressured" to assure 100% saturation, immediately before shearing. Nevertheless, from the results of our calculations, there appears to be no cause for concern.

(2) For combined earthquake and rapid drawdown, the effective shearing resistance of the dam on sloping surfaces is about 0.09W, where W is the weight of the sliding wedge. Although this is slightly less than the maximum earthquake acceleration of 0.10g, for a consistent value of maximum ground velocity of 5 in/sec., the maximum sliding displacement of each of the sloping surfaces is estimated to be less than 0.4 in. Even for a larger earthquake, the amount of motion under earthquake conditions appears to be relatively small or negligible.

(3) It appears that the static stability and the resistance to piping are the major problems in relation to possibilities of instability. Another major concern is the possibility of erosion of the downstream sloping surface if local settlements of the crest could occur, causing high velocity local flows if the weir is overtopped. A special spillway section could avoid this difficulty.

(4) Avoidance of erosion due to overtopping is also possible through use of riprap of adequate thickness and size of stone. This should be placed on a filter layer of thickness adequate to insure that the continuity of the filter will not be interrupted. It may not be possible to have this assurance with only a 12 in. thickness of filter, unless there is careful inspection during construction.

(5) To avoid piping and to insure downstream stability under steady-state seepage, a base drainage filter and toe drain is usually required. Although this is not shown in Amendment No. 5, it is our understanding that such a drain and filter of length about one-third the base width of the weir will be used. This will probably be adequate.

(6) Although no specific foundation treatment is indicated beyond the removal of alluvial materials, it appears to us that the foundation will present no problems from the point of view of large amounts of seepage or of stability against earthquake motions of the intensity considered possible in the application.

Respectfully submitted,

N. M. Newmark

bjw

cc: W. J. Hall
A. J. Hendron, Jr.

APPENDIX G
(106)

IN REPLY REFER TO:

UNITED STATES
DEPARTMENT OF THE INTERIOR
FISH AND WILDLIFE SERVICE
WASHINGTON 25, D. C.

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

APR 24 1967

Dear Mr. Price:

This is in reply to Mr. Case's letter of December 30, 1966, requesting our comments on the application by Duke Power Company for a construction permit and operating license for the proposed Oconee Nuclear Station, Units 1 and 2, Oconee County, South Carolina, Docket Nos. 50-269 and 50-270.

The plant would be located adjacent to the company's proposed Keowee Dam and Hydroelectric Station, on Keowee River just upstream from Hartwell Reservoir. The plant would employ two pressurized water reactors designed for a combined power output of 5,136 mw thermal, or an equivalent net capacity of 1,748 mw electrical. A radioactive waste disposal system, fuel handling system and all auxiliaries, structures, and other onsite facilities required for a complete and operable nuclear power plant would be provided.

Condenser cooling water would be conveyed to the station from the Little River arm of Lake Keowee through an intake conduit by 8 circulating water pumps, with a combined capacity of approximately 2,900 cfs. The intake canal would have a skimmer wall across its mouth with a 20-foot opening located 70 feet below full pool elevation. 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 tail water area. These resources support moderate sport fishing and a minor commercial fishery. The proposed Keowee and Jocassee Reservoirs will support fishery resources very similar to those of Hartwell Reservoir, with a good possibility for a second-layer cold-water fishery in Jocassee Reservoir. These proposed reservoirs will provide additional sport fishing opportunity in this area. Commercial fishing is minor in the project area and is not expected to increase significantly as a result of the company's proposed hydroelectric project.

The application indicates that the release of radioactive wastes would not exceed maximum permissible limits prescribed in Title 10, Part 20, of the Code of Federal Regulations. Although these limits refer to maximum levels of radioactivity that can occur in drinking water for man without resulting in any known harmful effects, operation within the limits may not always guarantee that fish and wildlife will be protected from adverse effects. If the concentration in the receiving water were the only consideration, maximum permissible limits would be adequate criteria for determining the safe rate of discharge. However, radioisotopes of many elements are concentrated and stored by organisms that require these elements for their normal metabolic activities. Some organisms concentrate and store radioisotopes of elements not normally required but which are chemically similar to elements essential for metabolism. In both cases, the radionuclides are transferred from one organism to another through various levels of the food chain just as are the nonradioactive elements. These transfers may result in further concentration of radionuclides and a wide dispersion from the project area particularly by migratory fish, mammals, and birds.

In view of the above, we believe that pre- and post-operational radiological surveys should be conducted by the applicant and include studies of the effects of radionuclides on selected organisms which require the waste elements or similar elements for their metabolic activities. These surveys should be planned in cooperation with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, and the South Carolina Wildlife Resources Department.

If the post-operational surveys establish that the release of radioactive effluent at levels permitted under Title 10, Part 20, Code of Federal Regulations, results in harmful concentrations of radioactivity in fish and wildlife, the data from the radiological surveys should serve as a guide to reduce the discharge of radioactivity to acceptable levels.

In view of the importance of the sport fishery of Hartwell Reservoir and the fishery potential of Lakes Keowee and Jocassee, it is imperative that every possible effort be made to protect these valuable resources from radioactive contamination. Therefore, it is recommended that the Duke Power Company be required to:

1. Cooperate with the Fish and Wildlife Service, the Federal Water Pollution Control Administration, the South Carolina Wildlife Resources Department, and other interested State agencies in developing plans for radiological surveys.

2. Conduct or arrange for the conduct of pre-operational radiological surveys of selected organisms that concentrate and store radioactive isotopes, and of the environment including water and sediment samples. These surveys should be conducted by scientists knowledgeable in the fish and wildlife field.

3. Prepare a report of the pre-operational radiological survey and provide five copies to the Secretary of the Interior for evaluation prior to project operation.

4. Conduct radiological surveys, similar to those specified in recommendation 2 above, analyze the data, and prepare and submit reports every three months during the first year of reactor operation and every six months thereafter or until it has been conclusively demonstrated that no significant adverse conditions exist. Submit five copies of these reports to the Secretary of the Interior for distribution to the appropriate State and Federal agencies for evaluation.

5. Reduce the discharge of radioactive wastes to acceptable levels, if the post-operational surveys establish that the release of radioactive effluent at levels permitted under Title 10, Part 20, Code of Federal Regulations, results in harmful concentrations of radioactivity in fish and wildlife.

We understand it is the Commission's opinion that its regulatory authority involves only those hazards associated with radioactive materials. We have recommended in past applications that thermal pollution and other detrimental effects from plant construction and operation be called to the attention of the applicant. In this case, however, we believe that the applicant is aware of the problem, since an analysis of thermal effects resulting from the operation and of the proposed nuclear plant was conducted by the Fish and Wildlife Service in conjunction with the Duke Power Company application for a license from the Federal Power Commission for the Keowee-Toxaway hydroelectric project, FPC Project No. 2503.

This evaluation was based on condenser cooling water intake from the Keowee River arm of Lake Keowee and discharge into Little River arm. Under these conditions the only anticipated detrimental effects upon the prospective fishery resources within Lake Keowee were the limitation of productivity in a relatively small area around the discharge point. No significant harmful effects were expected in Hartwell Reservoir or the proposed Jocassee Reservoir. The present plans for the proposed nuclear plant contains several modifications to the plan originally evaluated by the Fish and Wildlife Service. This application is for license of Units 1 and 2 of a total of three considered in the prior analysis. The volume of cooling water required will be less, but the condenser's cooling water intake and discharge points have been reversed and discharge outlets have been provided into the tailrace of Keowee Dam for emergency cooling water release and for routing discharge of liquid effluents from the nuclear plant's waste treatment facilities.

With the present plan it is not anticipated that there will be any change in the effects of the project upon fishery resources in Lakes Keowee and Jocassee. However, it is obvious that these project alterations change the consideration that must be given to possible damages to fishery resources in the Keowee tailrace and in Hartwell Reservoir. Duke Power Company has joined with a group of industries which, under the guidance of John Hopkins University, is investigating problems relating to the dissipation of waste heat in the aquatic environment. Preproject surveys of physical and biological conditions are in progress in the Keowee-Jocassee area, and firm plans have been made for their continuance when the project is in operation. The applicant has expressed the desire to cooperate fully with the Fish and Wildlife Service and the South Carolina Wildlife Resource Department in planning and carrying out these studies, and to make their findings available to these organizations.

We commend the applicant for its initiative in planning the pre- and post-operational surveys of the environment, and for their cooperation. If the post-operational surveys establish that the heated water discharged into Lake Keowee or its tailrace results in any changes in the environment of the tailrace or Hartwell Reservoir that are significantly detrimental to fish and wildlife, as determined by the Secretary of the Interior or the South Carolina Wildlife Resources Department, corrective measures should be taken to reduce the temperature of the effluent to an acceptable level.

We request that the Commission urge the Duke Power Company to:

1. Conduct pre- and post-operational surveys of the environment and include sufficient monitoring programs on effluents and receiving waters of Lake Keowee and Hartwell Reservoir, and collect related climatological data necessary for the Secretary of the Interior to evaluate the effects of the operation of the two units, prior to the approval of additional units.

2. Make any modifications in project structure and operations as may be determined necessary as a result of the surveys.

The opportunity for presenting our views is appreciated.

Sincerely yours,

[signature]

Acting Commissioner

APPENDIX G

(110)

IN REPLY REFER TO:

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

JUN 7 1967

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

Dear Mr. Price:

This is in response to Mr. Roger Boyd's letter of May 8, transmitting Amendment No. 3 to the Preliminary Safety Analysis Report, submitted by the Duke Power Company for its proposed Oconee Nuclear Station, Oconee County, South Carolina. Amendment No. 3 includes an application for a third pressurized water reactor to be known as "Oconee Nuclear Station, Unit No. 3". In our letter of April 29, we commented only on the effects that Units 1 and 2 would have upon fish and wildlife resources in the project area. We have the following additional comments concerning Unit No. 3.

The addition of a third unit would raise the capacity of the nuclear station to 2,622 megawatts, thereby increasing the possibility of damage to the fishery in the Keowee tailrace and Hartwell Reservoir from thermal effects. We reaffirm our request that the Duke Power Company: (1) conduct pre- and post-operational surveys, including sufficient monitoring of effluents and receiving waters of Lake Keowee and Hartwell Reservoir and collection of related climatological data, necessary for the Secretary of the Interior to evaluate the effects of the operation of the two units prior to construction of the third or any additional units; and (2) make any modifications in project structures and operations as may be determined necessary as a result of the surveys.

The opportunity for presenting our views is appreciated.

Sincerely yours,

Clarence F. Pautzke

Commissioner

Rec'd Off. Dir. of Reg.
Date 6/9/67
Time 2:40