MAY 24 1967

U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

IN THE MATTER OF

DUKE POWER COMPANY

CONSTRUCTION PERMIT APPLICANTION FOR

OCONEE UNITS 1, 2 AND 3

REPORT NO. 1

Note by the Director, Division of Reactor Licensing

The attached report has been prepared by the Division of Reactor Licensing for consideration by the ACRS at its June, 1967 meeting.

ABSTRACT

Duke Power Company has submitted an application for Construction Permits for Units 1, 2 and 3 at its Oconee Nuclear Station, Oconee County, South Carolina. The nuclear steam supply system will be supplied by the Babcock and Wilcox Company and the containment, a Bechtel design, will be built by Duke's construction division.

This report is a review of subjects on which the staff is satisfied and includes the site, core design, reactor coolant system, containment and engineered safeguards. The emergency power source is unique to this site, consisting of two on-site hydro electric plants with provision for external power separate from the Duke power grid during hydro outages.

We conclude that sufficient supporting information has been presented for the purposes of a provisional construction permit in the following areas:

(1) Site (exclusive of accident meteorology)

(2) Core Design

(3) Reactor Coolant System

(4) Containment

(5) Engineered Safeguards

(6) Sharing of auxiliary components between units.

We conclude that, although the following areas have not been completely resolved, sufficient information has been provided for the purposes of a provisional construction permit.

(1) An acceptable value of the moderator coefficient will have to be set at the operating license stage based on the final design of the core and more refined accident calculations. Since the applicant has demonstrated the feasibility of reducing or eliminating the positive coefficient, we believe that the proposed design is acceptable.

(2) The applicant has stated that if further analysis substantiates that xenon oscillations will occur, a method for controlling the oscillations will be developed during the detailed design. We believe this to be acceptable.

(3) The irradiation surveillance program including 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 will be further evaluated prior to issuance of an operating license.

(4) We will require a report of the steam generator test data and an analysis of their significance before final approval of the design.

(5) We believe that B&W in making the final design of the core should design those internals whose failure could lead to gross core disruption to remain within code allowable stresses under loss-of-coolant accident conditions, or that if they do not meet these conditions, a detailed engineering justification should be provided.

(6) Core cooling analyses have not been completed for the full spectrum of line break sizes and locations. We will require that these be completed prior to the issuance of an operating license. These analyses will include possibilities of emergency cooling water bypassing the core, including the possibility of a water leg remaining in the steam generator and trapping a steam bubble.

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1.0 INTRODUCTION

The Duke Power Company has submitted an application dated November 28, 1966, for construction permits and facility licenses for two pressurized water reactors to be Units 1 and 2 at its Oconee Nuclear Station located in Oconee County, South Carolina. The application was amended to include an identical Unit 3 by Amendment No. 3 dated April 29, 1967. Each of the three proposed reactors would operate initially at core power levels up to 2452 MW thermal and each has an expected ultimate core power rating of 2568 MW thermal.

The nuclear steam supply system and the first core for each unit will be supplied by the Babcock and Wilcox Company. Construction of the station will be by Duke Power Company which has retained the Bechtel Corporation as a general consultant.

The containment proposed is of the same basic design as the pre-stressed containments used in the Turkey Point and Palisades plants. The reactors are between the Turkey Point and Indian Point 2 reactors in power level. (Table 1-2, Vol. I, PSAR)

A chronology of the action taken on the application to date follows below.

Item	Date	Comments
Application for Units 1 & 2	November 28, 1966	
Staff Question list #1	March 23, 1967	
Amendment No. 1 Amendment No. 2	April 1, 1967 April 18, 1967	These amendments included information on site, thermal analysis, instrumentation, containment design and a revised core cooling system in response to Question list #1
Amendment No. 3	April 29, 1967	Application for Unit 3 and change in rod drives
ACRS Subcommittee Meeting	May 2, 1967	Visit to site
Staff Question list #2	May 11, 1967	

This report includes those topics on which the staff has taken a position with respect to the acceptability of the proposed design. Subjects which will be covered in a subsequent staff report to the ACRS are listed in the Conclusion (Section 10.0, this report).

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 one mile radius (from the center of Unit 2), the low population distance is at least six miles and the nearest population center is Anderson, South Carolina, population 41,000, located 21 miles southeast of

the site. The table below gives the total population at various distance.

Distance (miles)	1965	2010
0- 5	2,163	2,966
5-10	34,171	46,560
10-20	52,864	72,519

By 1985 when the shoreline of Lake Keowee will be fully developed, a transient population of about 7500 on a summer week-end is estimated within the 20 mile radius.

All land within the exclusion boundary will be either owned by Duke Power 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 residents 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 this is an acceptable arrangement since the applicant has evacuation control of these residences and we expect that the dose calculation will show there will be adequate time to evacuate these residences.

The reactors will take cooling water from the future Lake Keowee to be formed by the Keowee Dam, on the site, and the Little River Dam about four miles south of the site. The earthen fill dams will be designed to withstand the maximum 0.1g earthquake (on bedrock) postulated at the site. The seismic analysis of the dams was carried out using Newmark's "N" method as in the Carolina Power and Light application. (Question 8.6, Suppl. 1)

Subject to the submittal of the data obtained from the foundation investitation of the dams, which will be checked by our consultants to insure that no zones of poor material exist in the foundation rock and that no strata of unsuitable material will be present in the unremoved overburden, we believe that the dam design is adequate. As discussed in the emergency power section of this report, provision will be made to provide water and power sources for reactor shutdown even in the case of dam failure.

2.2 Meteorology

The staff has reached oral agreement with the applicant on the meteorological model to be used in the accident calculations and the model and dose calculations will be presented in the second staff report.

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 except that our consultants recommend that no critical structures be located on fill or cross cut-fill interfaces. We understand that no Class I structures will be so located.

2.4 Seismology

The applicant has proposed a design earthquake resulting in a maximum ground acceleration of 0.05g. In addition, for a ground acceleration of 0.1g on bedrock and 0.15g on overburden, the plant will be designed such

that there will be no impairment of function of critical structures and components. (Question 2.7, Suppl. 1) Based on our discussion of the seismology aspects of the site with the USC&GS we believe the above criteria are acceptable for the seismic design of the facilities.

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 airborne particulate material, water, soil and silt, vegetation, milk, and fish and animal life. (Question 2.6, Suppl. 1) 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 & Wildlife Service. We feel the scope of the program is adequate.

3.0 CORE DESIGN

3.1 Description

As presented in Table 1-2 of Volume I of the PSAR, the physical core parameters of the Duke plants, designed by Babcock and Wilcox, are not unlike recent Westinghouse designs. The principal variations are given in the table below:

	Oconee (B&W)	Turkey Point (Westinghouse)
clad thickness, in	0.026	0.0243
pellet diameter, in	0.362	0.367
UO_2 density, % theoretical	95%	93-94%
H_2O/U (volume ratio)	3.7	3.48

	Oconee (B&W)	Turkey Point (Westinghouse)
No. of control clusters	69	41
No. of control pins per cluster	16	20
Total control pins	1104	820
Fuel Rods per assembly	208	204

The larger number of control pins in the Oconee design is reflected in the larger reactivity increment held by control rods (10% versus 7%) and the lower operational boron concentrations.

A variation in UO_2 enrichment between Units 1 and 2 has been indicated by the applicant. This results from the possibility of using part of the Unit 1 fuel in the initial core loading for Unit 2, schedule permitting. We have not reviewed the use of irradiated fuel in the startup of a new unit and the applicant recognizes that restrictions might be placed on this mode of operation at the operating license stage.

3.2 Positive Moderator Coefficient

This core, as others of this size and type, is predicted to have a positive moderator coefficient under first cycle operating conditions. The positive moderator coefficient has been calculated by the applicant to be about 0.9×10^{-4} $(\Delta k/k)/^{o}F$ at the beginning of life. Present calculations indicate only 2 full power seconds of energy would be added under the worst loss of coolant condition which would result from inserting 0.5% in reactivity (based on the above moderator coefficient) after a hot leg

break. We are continuing to explore the adequacy of the applicant's calculation. However, the applicant has the capability to reduce the coefficient by the use of shims if needed. The applicant has calculated that addition of stainless steel shims would reduce the coefficient to 0.44×10^{-4} $(\Delta k/k)/^{o}F$ and addition of 2000 ppm natural boron in the shims would eliminate the positive coefficient. An acceptable value of the moderator coefficient will have to be set at the operating license stage based on the final design of the core and more refined accident calculations. Since the applicant has demonstrated the feasibility of reducing or eliminating the positive coefficient, and since we are continuing to evaluate the magnitude of the energy added, we believe that the proposed design is acceptable. (Question 6.1, Suppl. 2)

3.3 Xenon Oscillations

The applicant's calculations indicate that xenon oscillations might occur in the axial direction, that azimuthal oscillations are unlikely and that raidal oscillations will not occur. Calculations have been made to illustrate the ability of partial control rods, having a three foot poison section, to control a divergent oscillation. The applicant has stated that if further analysis substantiates the assumption that oscillations will occur, a method for controlling the oscillations will be developed during the detail design. We believe this to be acceptable. (Question 1.2, Suppl. 1)

4.0 REACTOR COOLANT SYSTEM

4.1 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. This, as in most other components of the primary system, is of similar material to 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. In addition the shell of the once-through steam generator is a Class A vessel. The only other classification difference from current designs is that the letdown coolers are Class C rather than Class A on the primary coolant side. The applicant has defended this choice on the basis that the heat exchangers are non-regenerative and therefore not subject to the thermal shocks that the regenerative heat exchangers used in other designs experience. In addition, the heat exchangers can be isolated by valve closure if necessary. We believe that this classification is acceptable.

4.2 Neutron Irradiation

We have reviewed the calculated fast neutron exposure of the Oconee reactor vessel and the corresponding shift in the NDT temperature. The reported time-integrated fast ($E > 1$Mev) neutron exposure of the vessel of 3×10^{19} nvt, and the estimated 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} nvt 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. We do not consider the calculational method employed to be fully satisfactory mainly because only four neutron energy groups were used to describe the neutron energy spectrum, and only four intervals were used in the S_n calculations to describe the angular segmentation of the flux. Nevertheless, a sufficiently high safety factor has been applied to the calculations to make them conservative. We reached this conclusion after consideration of the Oconee plant core size, power density and the inner diameter of the reactor vessel.

The irradiation surveillance porgram including 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 will be further evaluated prior to the operating license stage.

4.3 Steam Generators

The steam generators are of unique design, providing slightly superheated steam (35 degrees) at the exit of the generator. Our analysis to date indicates that the applicant has a sound design basis and that stresses imposed on the tubes during transients, including a steam line break, are low compared to the yield strength of the materials. B&W has indicated that a development program, including vibration and blowdown tests, is underway 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 so that the transient thermal analysis can be substantiated. (Appendix 4A, Vol II, PSAR)

4.4 Aluminum Components

The use of aluminum components was proposed in the original application for use in the axiliary systems at design pressures below 300 psig and design temperatures below 300°F. We have been informed by the applicant that these will be replaced by stainless steel piping systems. The only remaining aluminum components are storage tanks.

5.0 Containment

5.1 Description

The containment structures proposed for Duke Power Company's Nuclear Generating Units 1, 2 and 3 are of the same basic design as the containments used for Turkey Point and Palisades. They are structural concrete containments prestressed across the dome and throughout the side walls and employ reinforced concrete for the base slab. Likewise, most structural details are basically the same as in the Palisades and Turkey Point designs.

The reactor containment structure, which encloses the primary system, steam generators, and related auxiliaries, consists of a concrete shell in the form of a vertical right circular cylinder with a shallow spherical sector dome and flat slab base. A one-fourth inch welded mild steel (A-36) liner is attached to the inside face of the concrete shell as a provision to ensure leaktightness.

The base slab is of reinforced concrete construction using a high strength grade (60,000 psi yield) mild steel. The cylinder walls are prestressed circumferentially against hoop stress by three staggered systems of prestressing tendons anchored at vertical buttresses. The cylinder walls are, likewise, prestressed vertically with a series of uniformly spaced tendons extending from the top of the ring girder (thickened section at cylinder-dome intersection) to the bottom of the base slab. Local base moment is carried by 14S and 18S reinforcing bars on approximately one foot centers which are extended around the corner and up the wall about ten feet.

The dome is prestressed by a three way tendon system extending across the dome and anchored on a horizontal plane on the dome ring girder. A grid of supplemental reinforcement consisting of reinforcing bars on eighteen inch centers is provided on the exterior face of the cylinder and dome.

Additional reinforcement is provided on the interior face at the dome line and in the anchorage zones. Backing strips are provided at liner plate splices. Rigid shear "T" and "L" connectors are provided on the liner exterior face on, typically, fifteen inch centers.

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. Base shear is carried by the concrete section, by radial stirrup reinforcing, by vertical mild steel reinforcing and by the mild steel liner participating through composite action.

5.2 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. The manner of load combination is considered to consider realisitically all significant load combinations and is acceptable.

As a result of discussions with the applicant's consultant, Dames and Moore, the applicant has agreed to apply earthquake loadings corresponding to those derived from the spectrum presented in TID-7024 appropriately scaled. On the basis of discussion with our consultants, Drs. Newmark & Hall, we believe that the above treatment of seismic considerations is appropriate for this site.

5.3 Structural Design Details

The applicants overall design concept remains unchanged from Bechtel's criteria for Turkey Point and Palisades. Several changes in details are noteworthy. however. At our suggestion, the applicant has reviewed data relating to strength

in shear under combined loading and, as a result, has revised the design criteria. We consider that the revised criteria have clarified the design approach. We also believe that the use of Mattock's equations in consideration of radial shear represents a rational and more conservative approach to design than reliance on ACI 318-63 provisions. The criterion with respect to lateral shear is the same as the final criterion for Turkey Point and Palisades.

The applicant's base-to-cylinder liner detail has been improved over the Turkey Point and Palisades designs. In previous submittals a rather rigid transition was proposed whereas the design for Duke has incorporated a flexible liner transition section. It is our judgment that the present method for this junction should perform considerably better than that previously proposed with respect to potential leakage under design basis accident loading. The design of penetrations is, also, considerably improved over previous designs. The Oconee design indicates use of sizeable 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 nineteen feet in diameter for Oconee whereas for previous sister containments it is between eleven and fifteen 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 nineteen feet in diameter is not a major perturbation in the design of the structure. 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.

5.4 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 Turkey Point and Palisades. These are high quality, proven materials. Use of a system for cathodic proteciton of the structure is indicated. Likewise, liberal cover allowances on reinforcing steel have been specified to provide assurance that deterioration of the structure during its operating life will not be significant.

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 plant is constructed in accord with the requirements of the design. User testing of the materials of construction is indicated. The constuction 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.

5.5 Testing and Inservice Surveillance

An extensive program of acceptance testing for both structural and leak-tightness has been indicated. The program to establish structural acceptance will require extensive instrumentation around the large opening, at the discontinuities and on the liner to provide a high degree of assurance that anomalous

structural behavior will be detected. Likewise, extensive pre-operational integrated leakage tests are proposed to establish the structure's leakage characteristics.

Detailed inservice surveillance programs have not been established. However, the design will have adequate capability for a suitable program and review of these areas can be left for the operating stage.

5.6 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 equipment 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.

The advantage which the applicant hopes to gain by installing this system is a longer containment leak rate testing interval associated with a higher containment leak rate than otherwise possible. The initial proposal was that the containment leak rate be 0.5%/day and that credit be given 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 a means to test this division of leakage, the applicant has modified his 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% per day ($L_t \leq 0.25$) or

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

$L_t = L_c + L_p \leq 0.5$ and

$L_c = L_t - L_p \leq 0.25$

where L_t = total measured containment leakage

L_p = measured leakage through testable penetrations

L_c = leakage from all other sources

We believe that the above approach is acceptable and that the advantage of filtering the most likely source of containment leakage will justify the longer testing frequency interval associated with 0.5%/day. (Question 1.5, Suppl. 1)

The containment design as proposed for Duke's Oconee units has without question been presented in as much detail as any unit yet considered. The design details have evidenced a high degree of conservatism appropriate to a structure serving as the last barrier to fission product release. It is concluded that the design, as presently proposed, and the construction, as indicated, will result in a structure adequate for its intended purpose.

6.0 Engineered Safeguards

6.1 Core Cooling

The applicant's design basis for the 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 2000°F, the design basis implies that no clad melting will take place.

The criterion for maintenance of mechanical integirty during the blowdown is that deformation of reactor internals shall be limited to insure the capability to insert control rods and also to cool the core. The applicant has proposed that this be accomplished by limiting the direct membrane stress in the cylindrical support shells to values less than the yield strength of the material. The applicant also proposes to permit stresses up to 1-1/2 times the unirradiated yield strength for other basic load carrying members.

We believe that B&W in making the final design of the core should design those internals whose failure could lead to gross core disruption to remain within code allowable stresses under loss-of-coolant accident conditions, or that if they do not meet these conditions, a detailed engineering justification should be provided.

High pressure injection pumps, low pressure injection pumps and core flooding tanks (accumulators) will be provided to cool the core for any coolant break location and any size coolant line break up to the double ended rupture of a recirculation pipe.

The core flooding system is composed of two tanks separated by check valves from the primary system and maintained at 600 psi by compressed nitrogen. Injection into the primary system is initiated 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 designs. The water flows between the reactor vessel wall and the thermal shield and enters the bottom of the core.

An analysis 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 sufficient to cover the core hot spot 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 rupture of a 36 inch reactor outlet line. The hot spot temperature is limited to less than 2000°F for the largest line break

An analysis has not been completed on the possibility of a water leg remaining in the steam generator and trapping a steam bubble which would cause injection water to bypass the core. We believe that the analysis of this area and all other possible means of causing core bypass flow must be completed before the flooding tanks can be accepted at the operating license stage. This subject will pursued in future applications for which B&W is the steam supply vendor.

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 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 scram the decay heat level is such that one of the two low pressure injection coolers can handle the decay heat load. The low pressure injection system delivers water to the same nozzles as the core flooding tanks Under shutdown conditions these pumps serve as decay heat removal pumps.

During the period while the water source is the borated water storage tanks, the three high pressure injection pumps can also deliver water to the reactor.

Each high pressure pump will deliver 400 gpm at 1500 psig and 500 gpm at 470 psig. One pump will be used continuously during plant operation to provide seal water to the reactor coolant pumps. These pumps provide makeup for small line breaks for which the reactor is calculated to 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 tanks through a test line and allow extended operation with the high pressure pumps.

In conclusion, we believe that there is an adequate basis on which to grant a construction permit in regard to emergency core cooling systems provided:

(1) That the core be designed so that reactor internals whose failure could lead to gross core disruption will remain within code allowable stresses or otherwise shown acceptable.

(2) That core cooling analyses for the full spectrum of line break sizes and locations be completed and that all possibilities that could lead to emergency cooling water bypassing the core be considered, including the possibility of a water lag remaining in the steam generator and trapping a steam bubble are examined.

Of course we will have to follow the detailed design of the system as it progresses, as we are on all other pressurized water reactors.

6.2 Containment Cooling Systems

Two differently designed containment cooling systems are provided: (1) containment spray pumps which take water intially from the borated water storage tank and then from the containment sump and deliver it to the containment atmosphere

through redundant spray headers, (2) three emergency cooling units which each consist of a fan and a cooler. The heat sink for the tube coolers is 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 x 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 both of the systems are each degraded by the failure of a single active component. We believe that this system provides adequate redundancy for containment cooling for this reactor.

7.0 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 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 available) which remove core heat. 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 including the metal water reaction (Baker's parabolic rate equation).

(5) During blowdown a core surface heat transfer coefficient of 1000 Btu/hr-ft^2-^{o}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-^{o}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 and greatest heat transfer. The highest blowdown pressure peak (52.9 psi at 30 seconds) was found to result from a 3 ft^2 break. The highest post-blowdown pressure (52.7 psig at 200 seconds) resulted from the 14.1 ft^2 break. The second pressure peak results from the transfer of decay and metal-water reaction heat to the containment. The design pressure of the containment is 55 psig.

An analysis was also performed to illustrate compliance with Criterion 17 of the General Design Criteria. No injection flow was assumed and the analysis was terminated when the reactor vessel boiled dry. This gave a peak pressure of 53.6 psig at about 200 seconds.

The zirconium-water reaction capability of the containment was calculated assuming 3 emergency cooling units in operation (the design heat removal capability). The capability of the containment under these conditions (including hydrogen recombination) is as follows:

Time (sec)	% metal-water reaction	
	design containment cooling	all containment cooling
200	13	24
400	22	38
600	28	55
1200	45	100
2400	75	
3400	100	

Our comparison of the containment capability (without consideration of steam generator leakage) with the Turkey Point and Palisades capabilities indicate that it is equivalent and therefore acceptable.

8.0 Emergency Power and Water

8.1 Accident Conditions

To cope with the postulated loss of coolant accident coincident with loss of network power, the applicant has proposed that two hydro-electric 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 of 87.5 MVA corresponding to about 70 MWe. 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 witstand the maximum earthquake. We understand that the hydro equipment was ordered on the basis of withstanding a 0.2g earthquake before the present maximum ground acceleration of 0.1g was established.

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 seven to ten 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 two hours during an inspection and within six hours during repairs to the penstock.

During the periods of hydro plant maintenance emergency power can be fed to the site through a 100 kv transmission line which can be made separate from the external grid and which is designed for seismic loadings in excess of earthquake requirements. Power would be supplied by one of three 30 MWe gas turbines located at Duke's Lee Station thirty 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 proposal constitutes a satisfactory power source during the brief periods of hydro outage.

A more detailed analysis of the means provided to meet the single failure criterion in the hydro plant system will be a subject in the second staff report to the committee.

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 to test this feature on each unit.

were to fail and the lake and hydro units were lost. This would be accomplished 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 steam to the atmosphere.

The proposed power source for the electrically driven pumps is the 100 kv line fed by a gas turbine off-site as previously described. If not in operation, the gas turbine could be started in about 15 minutes. The 100 kv line will be located so as to not be affected by a dam failure. Any required repair of the lines could be performed in the available 20 hour period before power is required.

If the failure was not of the two major dams but of the intake canal dike, the lake would be retained behind the underwater weir and a pipe or hose could be run to the intake canal behind the weir in the available time period.

9.0 Plant Interactions

Units 1 and 2 share a number of auxiliary systems although no engineered safeguards systems except service water pumps are shared. In Supplement 3 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 the 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	Unit 1 and 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.

In addition, in Unit 3 the Chemical Addition and Sampling System (Table 9-5) will be sized for a single unit (a single system is shared between Unit 1 and

Unit 2), the spent fuel storage pool will be separate for Unit 3, and a separate rad waste system is presently envisioned for Unit 3.

We believe that sharing of the systems described above between units is acceptable. It should be noted that sharing of the engineered safeguards, as originally proposed, was revised to provide separate systems by Amendment No. 1 except that the service water which removes heat from the containment coolers is shared between Units 1 and 2 as indicated above.

10.0 Conclusions

We conclude that sufficient supporting information has been presented for the pruposes of a provisional construction permit in the following areas:

(1) Site (exclusive of accident meteorology)

(2) Core Design

(3) Reactor Coolant System

(4) Containment

(5) Engineered Safeguards

(6) Sharing of auxiliary components between units.

We conclude that, although the following areas have not been completely resolved, sufficient information has been provided for the purposes of a provisional construction permit.

(1) An acceptable value of the moderator coefficient will have to be set at the operating license stage based on the final design of the core and more refined accident calculations. Since the applicant has demonstrated the feasibility of reducing or eliminating the positive coefficient, we believe that the proposed design is acceptable.

(2) The applicant has stated that if further analysis substantiates that xenon oscillations will occur, a method for controlling the oscillations will be developed during the detailed design. We believe this to be acceptable.

(3) The irradiation surveillance program including 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 will be further evaluated prior to issuance of an operating license.

(4) We will require a report of the steam generator test data and an analysis of their significance before final approval of the design.

(5) We believe that B&W in making the final design of the core should design those internals whose failure could lead to gross core disruption to remain within code allowable stresses under loss-of-coolant accident conditions, or that if they do not meet these conditions, a detailed engineering justification should be provided.

(6) Core cooling analyses have not been completed for the full spectrum of line break sizes and locations. We will require that these be completed prior to the issuance of an operating license. These analyses will include possibilities of emergency cooling water bypassing the core, including the possibility of a water leg remaining in the steam generator and trapping a steam bubble.

Our evaluation of the following areas will be presented in a subsequent report to the ACRS on the Oconee Nuclear Station Units 1, 2 and 3:

(1) Thermal Design

(2) Instrumentation

(3) Rod Drives

(4) Accident meteorology and dose calculations

(5) Steam line isolation valves

(6) Liquid effluent release (accidental)

(7) Turbine missile analysis

(8) Evaluation of emergency power

The staff will be prepared to discuss the above subjects orally at the June ACRS meeting.