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Docket Nos. 50-269
50-270
and 50-287

July 24, 1970

Report to the ACRS

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

OPERATING LICENSE

U.S. Atomic Energy Commission
Division of Reactor Licensing

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ABSTRACT

Duke Power Company has requested operating licenses for its Oconee Nuclear Station Units 1, 2, and 3. The plant, located near Clemson, South Carolina, was evaluated at the license application power level of 2568 MWt. The nuclear steam supply system will be the first of the B&W power reactor systems to go into service.

This report presents the results of our evaluation which have been completed at this time.

The results of our completed review will be presented in a subsequent report.

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1.0 INTRODUCTION AND SUMMARY

1.1 General

This report presents the results of our evaluation of safety considerations in connection with a request by Duke Power Company (Duke) for operating licenses for their Oconee Nuclear Station Units 1, 2, and 3. The plant is located near Clemson, South Carolina.

Oconee Unit 1 will be the first of the B&W two-loop, four-pump, pressurized water reactor, nuclear steam supply system designs. The application for operating licenses is for a power level of 2568 MWt which is higher than the 2452 MWt requested in the construction permit application. Our evaluation of both the reactor systems and the engineered safety features has been performed for a maximum power of 2568. The construction permits were issued on November 6, 1967.

The containment design, but not procurement or construction of the containment, was performed by the Bechtel Corporation. The balance of plant design was performed by the Duke Engineering Department. Construction of the plant is being performed by the Duke Construction Department. The design and construction of the Oconee units have been taking place during a period of rapidly evolving design criteria regarding the safety of nuclear power plants.

1.2 Major Areas of Review

Our safety evaluation has emphasized review of the implementation of criteria into designs and of matters related especially to operation of the plant. Where applicable, we have referred to our completed reviews of other current generation nuclear plant designs. Our review also placed special emphasis on those design features unique to the B&W nuclear system and on site features. Because of the changes in regulatory criteria during design and construction of the Oconee Station we have also considered the many "as built" features of the Oconee Units relative to our current requirements.

Our review emphasized consideration of the following:

(1) Meteorology.

(2) Reactor power distribution monitoring and control.

(3) Once-through steam generator design and testing.

(4) Internals vent valve design and testing.

(5) Reactor material surveillance program.

(6) General structural and containment design.

(7) Post-LOCA hydrogen control.

(8) Emergency core cooling system performance.

(9) Instrumentation control and power system.

(10) The emergency provisions related to Lake Keowee (hydro stations, submerged weir and dam construction).

(11) Accident analyses.

(12) Operation with less than rated reactor coolant flows.

(13) Waste disposal.

(14) Conduct of operations.

(15) Shared systems.

(16) Technical specifications.

Although our review of the Oconee plant is not complete, major portions of our evaluation have been finished. The following sections, or major portions thereof, are complete and are submitted to the Committee for review at this time.

2.0 Site and Environment

3.0 Reactor Design

4.0 Reactor Coolant System (Partially Complete)

5.0 Containment and Class I Structures (Partially Complete)

7.0 Instrumentation, Control and Power Systems (Partially Complete)

8.0 Auxiliary Systems (Partially Complete)

9.0 Accident Analyses (Partially Complete)

10.0 Conduct of Operations (Partially Complete)

11.0 Quality Assurance

For the remaining sections of this report we have included brief discussions of the status of our review. We anticipate that a supplemental report will be submitted next month to complete our review.

A chronology of our review is presented in Table 1.2-1.

1.3 Summary

In general, for those portions of the review on which we have concluded favorably, the bases for our acceptance and favorable conclusions are stated in the body of the report.

A number of significant favorable (acceptance) conclusions have been reached involving first-of-a-kind matters which may set precedents or

TABLE 1.2-1

CHRONOLOGY OF OPERATING LICENSE REVIEW FOR OCONEE STATION

	Date	
1.	June 2, 1969	FSAR submitted as Amendment 7 to Duke's Application for licenses for the Oconee Nuclear Station.
2.	August 5, 1969	Meeting with Duke to discuss general aspects of our review.
3.	August 13-14, 1969	Visit to Duke engineering offices and Oconee construction site with our seismic consultant to discuss status of design and observe construction progress at the site.
4.	September 15, 1969	Application Amendment 8 submitted, providing information on quality assurance (QA) and piping system classification and incorporating seven B&W Topical Reports by reference.
5.	September 18, 1969	Meeting with Duke on QA.
6.	September 24, 1969	Meeting with Duke to discuss thermal hydraulics design.
7.	November 17-18, 1969	Meeting with Duke to discuss site instrumentation, electrical systems, reactor physics, steam generators and vent valves, conduct of operations and initial tests.
8.	November 28, 1969	AEC-DRL letter to Duke requesting information on sustained DNB analysis as covered in B&W Topical Report BAW-10[illegible]4.
9.	January 21-22, 1970	Meeting with Duke to discuss B&W Topical Reports, preoperational testing and electrical penetrations.
10.	February 9, 1970	Application Amendment 9 submitted, providing information in response to our QA review and incorporating two B&W Topical Reports by reference.
11.	February 13, 1970	AEC-DRL letter to Duke requesting additional information to continue review (major question list).
12.	February 19, 1970	Site visit with our meteorology consultant to witness gas diffusions testing and discuss Duke's efforts to resolve meteorology problems.

13.	March 3, 1970	AEC-DRL letter to Duke requesting information on core internals and accident analyses.
14.	March 16, 1970	Application Amendment 10 submitted, providing the final stress analysis report on the reactor coolant system.
15.	March 19-20, 1970	Meeting with Duke at B&W, Lynchburg, Virginia facility to discuss thermal hydraulic design analyses.
16.	March 27, 1970	AEC-DRL letter to Duke requesting information on thermal hydraulic design codes.
17.	March 30-31, 1970	Meeting with Duke at B&W, Lynchburg, Virginia facility to discuss core internals design analyses.
18.	April 2, 1970	Meeting with Duke to discuss instrumentation and electrical system drawings.
19.	April 3, 1970	Meeting with Duke at Bechtel's Gaithersburg, Maryland facility to discuss structural and piping design analyses.
20.	April 15, 1970	AEC-DRL letter to Duke requesting additional information on accident analyses.
21.	April 20, 1970	Application Amendment 11 submitted, providing responses to our letters of November 28, 1969 and February 13, 1970, and incorporating one B&W Topical Report by reference.
22.	April 22, 1970	AEC-DRL letter to Duke requesting information on core internals.
23.	May 6, 1970	AEC-DRL letter to Duke requesting information for our environmental policy statement.
24.	May 25, 1970	Application Amendment 12 submitted, providing responses to our letters and incorporating two B&W Topical Reports by reference.
25.	June 18, 1970	Meeting with Duke on Technical Specifications.
26.	June 22, 1970	Application Amendment 13 submitted, providing responses to AEC requests for information and incorporating three B&W Topical Reports by reference.

27. June 23, 1970 — ACRS Subcommittee meeting and site visit.

28. June 25, 1970 — Meeting with Duke to inform B&W in detail on our concerns with reference to potential deficiencies in ECCS analyses.

29. July 9, 1970 — Application Amendment 14 submitted, providing by reference a Duke report containing proprietary answers to our letters of February 13, 1970 and March 3, 1970. Also incorporated by reference, four B&W Topical Reports.

30. July 9, 1970 — Application Amendment 15 submitted, providing answers to several of our letters including instrumentation qualification tests and meteorology measurements.

involving acceptance of design features which do not meet present day regulatory criteria. These are summarized below:

(1) We have allowed a reduction factor of 2.2, as a result of onsite SF_6 gaseous tracer diffusion studies in arriving at our X/Q value.

(2) We have accepted onsite hydro-turbine generators as emergency power sources.

(3) We expect to approve a unique reactor internals vent valve designed to prevent potential steam binding during a LOCA.

(4) We expect to approve a unique once-through steam generator for which there has been no previous operating experience.

(5) There are two unique features related to emergency cooling water. We have accepted an underwater weir in the intake canal to trap water in the event of a dam failure and we have accepted a siphon effect capability for condenser cooling during a complete loss of power.

2.0 SITE AND ENVIRONMENT

2.1 Site Location and Description

Oconee Station is located in Oconee County, South Carolina, about 8 miles northeast of Seneca, South Carolina. The site is adjacent to Lake Keowee which was formed by impounding the Keowee and Little Rivers with separate dams and then joining the lakes by a canal about half a mile north of the site. The nuclear station is about eight-tenths of a mile west of Keowee River at the dam. Anderson, South Carolina, the nearest population center (1960 population of 41,136), is 21 miles south. The applicant proposes a minimum exclusion radius of 1 mile. Based on the 10 CFR 100 definition of the exclusion radius, we conclude that the distance selected by the applicant is acceptable.

The applicant has proposed a 6-mile Low Population Zone (LPZ) which he estimates will contain 3,400 people in 1970 and 8,900 people by 2010. The transient population within the LPZ is estimated by the applicant to be 2,000 in 1970 and, because of the development of recreational and vacation facilities along Lake Keowee, is expected to increase to 19,000 by 2010. Based on the population distribution in the proposed LPZ and the 10 CFR 100 definition of the LPZ, we conclude that the 6-mile LPZ is acceptable.

2.2 Geology and Seismology

Plant structures will be founded on Piedmont granite gneiss rock. According to the applicant and to the AEC Division of Compliance, no unusual problems concerning the foundation material occurred during construction. The geology and seismology of the Oconee site were reviewed in detail during the construction permit (CP) stage, and nothing has occurred to alter our previous conclusions that the geological and seismological conditions are acceptable for the safe operation of the Oconee nuclear units.

The Class I structures founded on bedrock were designed to withstand horizontal ground accelerations of 0.10g and 0.05g for the Design Basis Earthquake (DBE) and Operating Basis Earthquake (OBE), respectively. For Class I structures on overburden a DBE acceleration of 0.15g was used as a design parameter.

2.3 Hydrology

Lake Keowee will be the source of condenser cooling water for the Oconee plant. Cooling water will be withdrawn from the Little River arm of the lake and discharged into Lake Keowee just west of Keowee Dam on the north side of the plant property.

In order to provide a continuous supply of emergency cooling water, a Class I submerged weir, which impounds 9 million cubic feet of water, has been constructed across the lagoon from which condenser cooling water is withdrawn. In the unlikely event that Keowee or Little River Dam should fail (both have been shown by the applicant to be capable of withstanding the DBE accelerations), the water retained by the submerged weir would be circulated through the condensers and back to the intake lagoon providing continuous emergency cooling.

The applicant, using a maximum hypothetical precipitation of 26.6 inches of rainfall within a 48-hour period occurring over the entire affected drainage areas, calculated what we evaluate to be equivalent to the Probable Maximum Flood (PMF). This flood results in a lake stage of 808 ft above MSL (mean sea level). Plant grade and critical plant components, at 796 ft above MSL, are provided flood protection to 815 ft above MSL by the Keowee Dam and intake canal dike. We have made an independent analysis of the the anticipated wave effects and have concluded that the 7 feet of freeboard between the PMF flood stage and the 815 ft above MSL protection level is adequate to protect the nuclear plant against any credible combination of wave effects and PMF stage.

Based on the considerations discussed above, we and our hydrological consultants, the U.S. Geological Survey, conclude that the hydrological conditions at Oconee Station are acceptable relative to the protection of public health and safety during operation of the Oconee nuclear units. A copy of the U.S. Geological Survey report has been forwarded to the Committee.

2.4 Meteorology

During the CP review of the Oconee site, we and our meteorological consultant, ESSA, concluded that a "Valley" diffusion model would best characterize the meteorology of the site because of the complicated, rough topography between the Oconee nuclear units and the nearest site boundary 1 mile to the south. In this postulated model it was assumed that the effluent released as a consequence of a reactor accident would be channeled generally down the Keowee River Valley to the nearest site boundary. Complete mixing of the effleuents was assumed to occur within the confines of the topographic ridges along the Keowee River between the nuclear plants and the exclusion radius. The resultant meteorological diffusion factor was 7.4×10^{-5} sec/m^3. The applicant expected to prove that the "Valley" model was valid for the site from the results of the post-CP meteorological program.

The applicant conducted 15 gas tracer (SF_6) experiments under inversion conditions, and in all cases the centerline concentration was lower than that which would have been predicted by the use of the equivalent Pasquill type diffusion conditions. The best agreement between a measured concentration and the concentration predicted with the use of the Pasquill model was at 680 meters where the measured concentration was a factor of 2.2 lower than what Pasquill categorization would have predicted, including building wake credit. In all other cases the measured concentration was even a smaller fraction of the predicted concentration.

An examination of 1 year's data of the joint frequency tabulation of wind speed, direction, and stability condition (delta T) taken from a 150-foot tower indicated that for 9 percent of the time the diffusion rate was equal to, or worse than, that associated with Pasquill Type F conditions and a wind speed of 1.5 m/sec. These data indicate that a diffusion rate equivalent to, or worse than that associated with Pasquill Type F conditions and a wind speed of 1 m/sec occurs 5 percent of the time.

Because of the height of the trees in the vicinity of the meteorological tower, the wind measurements were made at 150 feet above grade rather than at a level more appropriate for a ground level release (20 or 30 feet elevation). However, based upon a visual examination of the topography of the site, the 150-foot level appears to be a reasonable lower level at which to collect wind data free from topographic interference. In a related matter, a wind speed calibration check was made in October 1969 and indicated that the instrument was measuring low by a factor of 1.4. However, there is no rigorous way to determine how long this situation had persisted and to what extent the data in the joint frequency tabulation were affected. Therefore, the effect of not correcting the 150-foot wind measurements down to the appropriate 20- or 30-foot elevation is probably compensated for by the 1.4 calibration factor. For this reason, we believe that the unmodified 150-foot-elevation wind data are a reasonable representation of the wind speeds to be expected at the 20- to 30-foot level.

In evaluating the radiological consequences of the design basis accidents, we have employed our usual model of Pasquill Type F conditions and a wind speed of 1.0 meter per second (which was shown to be applicable by the onsite data) with building wake credit and with an additional correction factor of 2.2 which is justified by the improved diffusion at Oconee Station due to topographical effects, as demonstrated by the gas tracer (SF_6) measurements discussed above. This results in a diffusion factor of 1.16×10^{-4} sec/m^3. Without the correction credit the staff's diffusion factor would have been 2.18×10^{-4}sec/m^3. ESSA, whose report has been forwarded to the Committee, has concurred in this model with its building wake credit and correction factor.

2.5 Environmental Monitoring

A preoperational environmental monitoring program was initiated in January 1969, so that 2 years of data would be available before startup of Unit 1. (Water samples from private wells and from the Keowee and Little River arms of Lake Keowee have been analyzed since 1966.) The preoperational program included the following samples: water, airborne particulates, rain, settled dust, silt (river and lake), vegatation, aquatic vegetation, algae and plankton, fish, milk, and animals. No environmental radiation anomalies have been indicated by the preoperational data thus far reported.

The operational environmental monitoring program will be an extension of the preoperational program with the following additions in order to provide a more comprehensive program to quantify the environmental effects of operating the nuclear units:

(1) two additional onsite air monitoring stations,

(2) a continuous water sampling station on the Keowee River just within the exclusion radius, and

(3) a thermoluminescent dosimeter network within the 1-mile exclusion radius.

The frequency of sampling and types of analyses of each media are given in Table 2-1a and in Sections 2.7.2 and 2.7.3 of the FSAR. These references will be incorporated in the Oconee technical specifications.

Duke Power Company is cooperating with the South Carolina State Board of Health, South Carolina Pollution Control Authority, South Carolina Wildlife Resources Department, and the U.S. Fish and Wildlife Service in matters concerning the environment.

The Oconee environmental monitoring program encompasses the recommendations of the U.S. Fish and Wildlife Service with the exception of sampling within 500 feet of the liquid effluent outfall.

The applicant has added a commitment to sample aquatic biota, crustaceans or mollusks, benthic organisms, and bottom sediments as near the outfall as they can be found. The applicant has stated that the scouring effect of the Keowee Hydro Plant discharge will prevent the accumulation of these organisms or bottom sediments close to the outfall. The comments of the Fish and Wildlife Service have been forwarded to the Committee.

Based on the description of the environmental monitoring program in the PSAR and FSAR, we conclude that the environmental surveillance for Oconee Station is acceptable.

3.0 REACTOR DESIGN

3.1 General

The design of the B&W reactors for Oconee Units 1, 2, and 3 are similar to currently designed Westinghouse and Combustion Engineering (CE) reactors, in most respects. They employ full- and part-length control rods, a chemical neutron absorber, Zircaloy fuel cladding, and burnable poison rods (for Unit 2). A unique feature of the B&W design is the use of internal vent valves to prevent steam binding for a loss-of-coolant accident (LOCA) resulting from rupture of a pressure vessel main coolant inlet line (cold-leg break).

Tables 3.1-1 and 3.1-2 provide comparisons with the Indian Point (4-loop), H. B. Robinson (3-loop), and Palisades (2-loop - 4-loop) reactors. The initial cores for Oconee Units 1, 2, and 3 will be of the same design except that Unit 2 will employ burnable boron rods and Unit 3 will utilize fuel from Unit 1.

3.2 Nuclear Design

3.2.1 Introduction

We have reviewed the reactor physics related aspects of the Oconee reactor designs based on material provided by the applicant in the FSAR, revisions to the FSAR in response to our questions, discussions in meetings with the applicant and B&W, and some independent calculations obtained through our technical assistance program at Brookhaven National Laboratory. In general, we find that the computer programs, techniques used, and comparison calculations of experimental data provide a high degree of confidence that the nuclear characteristics of the reactors and attendant safety-related provisions have been predicted accurately. We conclude, however, that greater conservatism than the applicant has provided is required in the areas of (1) power distribution monitoring and control and (2) restriction of the beginning of life (BOL) positive moderator coefficient. For these areas, we intend to require technical specifications comparable with those developed for the H. B. Robinson, Point Beach, and Palisades facilities.

TABLE 3.1-1

THERMAL AND HYDRAULIC DESIGN COMPARISON

	Oconee 1, 2, 3	Indian Point 2	H. B. Robinson	Palisades
Thermal Power, MWt	2568	2758	2200	2212
Hot Channel Factors				
Heat Flux				
Nuclear F_q^N	3.03	3.12	3.13	3.62
Engineering F_q^E	1.03	1.03	1.03	1.05
Total	3.12	3.23	3.23	3.80
Enthalpy Rise				
Nuclear F_h^N	1.78	1.75	1.75	1.94
Heat Transfer Surface, ft^2	49,734	52,200	42,460	51,400
Avg Heat Flux, $Btu/hr/ft^2$	171,470	175,600	171,600	142,400
Max Heat Flux, $Btu/hr/ft^2$	534,440	570,800	554,200	541,200
Fuel Central Temperature °F				
Max at 100% Power	4250	4090	4030	4040
Max at Max Overpower	4600 (114%)	4380 (112%)	4300 (112%)	4350 (112%)
Linear Power Density, KW/ft				
Max at 100% Power	17.63	18.4	17.9	17.6
Max at Max Overpower	20.1 (114%)	20.6 (112%)	20.0 (112%)	19.7 (112%)
Avg at 100% Power	5.66			
DNB Ratio at Nominal Conditions (W-3)	2.0	2.00	1.81	2.0
Minimum DNB Ratio for Design Transients	1.3	1.30	1.30	1.3

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TABLE 3.1-2

NUCLEAR CORE DESIGN COMPARISON

	Oconee 1, 2, 3	Indian Point 2	H. B. Robinson	Palisades
Active fuel height, inches	144	144	144.0	132.0
Number of fuel rods	36,816	39,372	32,028	43,168
Fuel Assemblies				
Type	canless	canless	canless	canless
Number	177	193	157	204
Array	15 x 15	15 x 15	15 x 15	16 x 16
Fuel rods/assembly	208	204	204	212
Assembly spacing; inches	8.59	8.47	8.47	8.11
Cladding OD, inch	0.430	0.422	0.422	0.4135
Fuel rod pitch	0.568	0.563	0.563	0.550
Rod type	Unpressurized	Unpressurized	Pressurized & Unpressurized	Unpressurized
Control Assemblies				
Type	Rod cluster	Rod cluster	Rod cluster	Cruciform
Poison material	5%Cd 15%In 80%Ag →	→	→	→
Number full length	61	53	45	41
Number partial length	8	8	8	4
Total number	69	61	53	45
Control rods/cluster	16	20	20	N/A
Burnable Poison Rods				
Number	1088	1156	816	704
Poison material	Boron (Unit 2 only)	Boron	Boron	Boron

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3.2.2 Reactivity Calculations

The applicant has described the computer programs and calculational techniques used by B&W to predict the nuclear characteristics of the reactor designs, and has provided examples to demonstrate the ability of these methods to predict UO_2 and $P_u O_2 - UO_2$ critical assemblies. We concur that these examples demonstrate the validity of the methods used to predict k_{eff} for the large power reactor cores.

Detailed three-dimensional power distribution measurements have been performed by B&W at the Babcock & Wilcox Critical Experiments Laboratory. Results of the applicant's calculations using PDQ07, a three-dimensional program, agree quite well with the measured power distributions. The B&W version of PDQ07 used for the calculations incorporates a thermal feedback option, permitting accurate descriptions to be made of the radial and axial power distributions in analyses of control rod maneuvering, xenon stability status and control, and reactivity coefficients. These distributions are needed to evaluate core thermal margins.

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

We have concluded that the material presented adequately demonstrates B&W's ability to predict the physics characteristics of the reactors.

3.2.3 Reactivity Control Requirements

The applicant has provided substantial information supporting his ability to control the excess reactivity provided in the reactors and maintain a shutdown margin to hot critical of at least 1% Δk/k throughout core life with the most reactive control rod stuck out of the core. Sixty-one full-length control rods are provided. This is greater than the number provided in other PWR designs. The predicted worth of 12.1% Δk/k for Oconee 1, is also greater than the worth usually provided. For 90% of the fuel cycle in Oconee 1, 1.0 to 1.3% of the control rod worth will be used for partial xenon override capability, in one of the four permissible rod groupings. This worth will normally be inserted in the reactor and unavailable for shutdown; but the remaining worth is still more than the usual amount provided. Soluble boron is used for control in the Oconee reactors in a manner generally similar to that employed in other PWR plants.

We have concluded that the applicant's assessment of control requirements over the core lifetime is conservative. It assumes a Doppler deficit based on the maximum expected fuel temperatures at the appropriate point in life. Although the calculated total rod worth was reduced by 10%, the calculated worth of the stuck rod, which is conservative compared to other PWR plants licensed for operation, was not reduced. The assumptions used in the rod worth calculations were conservative. In particular we noted that the calculated increase in the worth of the full-length rods from the presence of the eight part-length rods was not claimed. For these reasons, we have concluded that adequate control rod worth has been provided to assure shutdown. The required Oconee 1 startup program measurements are expected to verify this conclusion. Although some differences in rod worths will exist for Units 2 and 3, because of the differences in fuel loadings, we have concluded that these will not significantly alter our conclusion.

3.2.4 Power Distribution Monitoring and Control

The basic concept for monitoring the nuclear power level and distribution in the Oconee reactors is the same as for all PWR plants recently licensed for operation. Primary reliance is placed on four axially split out-of-core detectors which are located symmetrically and opposite to the core diagonals. Also, 52 assemblies of 7 local and 1 full-length self-powered neutron detectors are available for in-core mapping. Special calibration tubes are provided in the Oconee 1 core. Normal readout will be through the plant computer; however, a backup readout system is provided for selected detectors (this will be a technical specification consideration). The applicant's initial position was that the in-core detectors were provided for fuel management purposes, and initial use in the startup program to obtain correlations with out-of-core detector responses. Our position is that the out-of-core detectors are adequate for detecting power maldistributions originating from axial xenon instability and misplaced control rods, only if a power distribution mapping capability is provided to periodically calibrate the out-of-core detectors and to investigate any power distribution anomalies detected by the out-of-core detectors. We will require that the technical specifications for the in-core detectors take this position into account.

Eight part-length rods are provided for control of the axial power shape. These are required for control of the axial xenon instabilities, that are predicted to be possible in the Oconee 1 core at BOL with the xenon override rods in the core. We intend to require the applicant to establish in the technical specifications, limits on the imbalance of power in the upper and lower halves of the core, in order to prevent unacceptable axial peaking factors. In the Westinghouse-designed reactor protection system provided in H. B. Robinson Unit 2, overtemperature ΔT and overpower ΔT reactor trip setpoints are automatically reduced when this imbalance exceeds a threshold value. We have concluded that normal action to move the part-length rods to control this imbalance, or failure to use these rods, could potentially lead to core damage. We have verbally informed the applicant of our concern in this regard. We are currently evaluating this potential for core damage and will provide our conclusions in a subsequent report to the ACRS.

The applicant has not yet responded to our request for a technical specification and basis for x-y (or radial) power tilt limits. Since the applicant claims x-y xenon stability, and intends to verify this in a startup test, the only known source for such tilts is misplaced control rods. Since single misplaced rods do not result in violation of safety limits, we conclc only alarms are needed to alert the operator that a nuclear rad'al mit is being approached. These alarms would provide a backup to the pon derived from the rod position indicators. Resolution of this matt be discussed in a subsequent report.

3.2.5 Moderator Coeffic

For Oconee plicant has proposed a technical specification limit for the BOL modeefficient of + 0.9 x 10^{-4} $\Delta k/k/°F$ at full power, based upon use ovalue in his accident analyses. On the basis of our evaluation, we coat this is an appropriate limit for rod ejection accidents, and fxenon stability analyses (because the applicant will demonstrate stabiut we have not yet completed our analysis of the maximum possible (of the related moderator density coefficient) in a loss-of-coolantnt (see Section 6.1 of this report). The predicted moderator temperaefficient for Oconee Unit 1 is + 0.27 x 10^{-4} $\Delta k/k/°F$. A slightly less p coefficient is predicted for Units 2 and 3. The Unit 1 value wasd from a power distribution weighted calculation, and is smaller th isothermal coefficient. We believe that this effect results from thee of the xenon override control bank. Our technical assistance consut BNL have shown that the power distribution weighted c. fficient can br or smaller than the isothermal coefficient, depending on the power Further, their calculations indicate that a greater soluble boron contion will be required than the applicant predicts, which would also raise the moderator coefficient. We do not suggest that the BNL calcs are more valid than the applicant's, but they do contribute to ourn that a more positive coefficient might be present in the reactor th predicted. We are therefore working very carefully with the applican technical specification to ensure that the BOL full power coefficientdicted from low power measurements is indeed limited to + 0.9 x 10^{-4} Δ As noted above, this coefficient is an input to the loss-of-cooldent which is currently being re-examined both by us and the applicant results of this examination could further limit the permissible valuds coefficient.

3.3 Thermal and Hydraulic Design

We expect to provide the Committee with our complete review of this subject in our next report. We are currently evaluating additional information requested from the applicant on partial-loop operation.

We have, however, advised the applicant that we have reached a tentative conclusion not to license Oconee Unit 1 for one-loop operation. Our conclusion is based on these points:

(1) Vessel model flow tests did not simulate one-loop operation and thus flow patterns are unknown.

(2) The idle steam generator and associated piping constitute a flow bypass path; bypass flow through an idle hot leg would be indicated by the flow tube in the working hot leg to be core flow. Also colder bypass flows would mix with the hot core flow and be interpreted as lower core temperature.

(3) Instrument criteria (redundancy) are in jeopardy, as cold reverse (bypass) flow in an "idle" hot leg effectively disables one-half of the flow and outlet temperature measurements.

We expect to elaborate on these points in our subsequent report.

We do expect to license Oconee for three-pump operation and for one-pump-in-each-loop operation. However, we are not finished with our discussions on power level and scram setpoints. We have concluded that for such operation, the power-to-flow protection channels constitute one acceptable means of automatically and properly changing the power level at which the reactor will be tripped as reactor coolant flow changes.

Yet to be resolved are:

(1) Burnout correlations on the fringe of W-3 applicability,
(2) Flow instability threshold,
(3) Flow patterns for partial-loop operation,
(4) Use of outlet temperature thermocouples in the fuel assemblies during startup tests,

(5) Use of the power/reactor-coolant-pump-logic trip for core protection (Applicant proposes to delete this trip for three-pump operation),

(6) Reactor trip adjustments for one-pump-in-each-loop operation,

(7) Analysis of backflow in idle loops for partial-loop operation.

We will incorporate our conclusions on these items in a subsequent report.

3.4 Mechanical Design

3.4.1 Reactor Internals

The reactor internals have been designed to operate within the allowable stress intensity limits of Section III of the ASME Boiler and Pressure Vessel Code for normal design loads of mechanical, hydraulic, and thermal origin, including the operational basis earthquake and anticipated transients.

All internals components are designated as Class I (seismic) items, and are designed to withstand loads resulting from a combined design basis earthquake and loss-of-coolant accident. Strain limits for the internals under this combined load are held to less than 20% of the uniform ultimate strain for this material (Type 304 stainless steel) corresponding to an elastically calculated stress limit of not greater than two-thirds of the ultimate tensile strength. Allowable deflection limits are generally within 50% of loss-of-function deformation limits. We consider these design limits to be acceptable.

Topical Report BAW-10008, Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake, Part 1, and Proprietary Part 2, dated June, 1970, describe the methods of analysis employed for the internals and fuel assemblies under concurrent LOCA and design basis earthquake loadings. An earlier version of this report was submitted for our review in mid-1969. We and our seismic design consultant, John A. Blume & Associates, Engineers, are continuing our review of the Topical Report. We will incorporate the conclusions of our review in a subsequent report.

3.4.2 Vibration Control

Flow induced vibration analyses have been made for the reactor internals including the thermal shield, fuel assemblies, fuel rods, surveillance tube and specimen holder assemblies, control rod guide tube assemblies, and the piping for the in-core monitors. The vibration analyses of the thermal shield showed that the flow-induced pressure fluctuations acting on the surface of the shield resulted in modal amplitudes of less than 0.002 inch. These analyses considered inlet flow impingement and turbulent flow, as well as natural frequency calculations, and the results indicated that a factor of at least 2 exists between conditions of possible resonance and excitation frequencies. It has also been determined that the flow-induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc.

The applicant has not yet submitted his plans for vibration monitoring during preoperational testing of the Oconee plant. We will review this information when received and incorporate our conclusions in a subsequent report.

The feasibility of inservice monitoring for vibration and the detection of loose parts is being explored by B&W. They have investigated the application of such sensors as accelerometers, strain gages and load cells to monitor vibrations of internals, and of inertially loaded-force pickups to monitor for loose parts. B&W plans additional discussions with consultants and instrumentation vendors in order to determine the feasibility and practicality of developing acceptably sensitive systems for installation in nuclear plants. The applicant is aware of these programs being pursued by B&W, and of our concerns in this area. We have informed the applicant that when a suitable monitoring system is available, it should be installed in the Oconee plants.

4.0 REACTOR COOLANT SYSTEM

4.1 General

The reactor coolant system has been designed as a Class I (seismic) system to withstand the normal loads of mechanical, hydraulic, and thermal origin including anticipated plant transients and the operational basis earthquake within the stress limits of the appropriate codes given below.

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

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

Nondestructive examination requirements for reactor coolant system pumps and valves are given in Table 4-12 of the FSAR. These examinations include radiography of castings, ultrasonic testing of forgings, dye penetrant inspection of pump and valve body surfaces, and radiography of circumferential weldments. This program upgrades the nondestructive testing of pumps and valves within the reactor coolant pressure boundary to essentially that of the ASME Code for Pumps and Valves for Nuclear Power.

The applicant states that earthquake loads for the OBE and DBE have been determined by dynamic analyses. We and our consultants are continuing our review of additional information recently submitted on this subject. We will report our conclusions in a subsequent report.

Duke has made arrangements to replace the Unit 1 reactor coolant pumps because a current problem on long-term wear cannot be resolved in a timely fashion. The applicant will submit appropriate information on the replacement pumps for our review and we will report our conclusions on this matter in our next report.

4.2 Reactor Vessel

The reactor vessels have been designed and fabricated in accordance with Class A requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, including the Summer 1967 Addenda. Applicable Code Cases are 1332, 1325, and 1336.

The vessels are essentially identical to those intended for the Arkansas Nuclear One, Crystal River 3, Rancho Seco 1, Midland 1 and 2, and Three Mile Island 1 and 2 plants, and have been designed to permit complete removal of the vessel internals. Fabrication materials are low alloy steel plates, Type SA-533, Grade B, Class 1, and forging steel Type SA-508-64, Class 2. The vessel interior is clad with Type 304 austenitic stainless steel applied by weld overlay technique. The applicant has informed us that furnace sensitization of stainless steel vessel material has been limited to the nonpressure-bearing interior cladding. The requirements for nondestructive examinations have been limited to those required by Section III, except that head and shell plate material and flange forgings have been given a 100% volumetric examination using both longitudinal and shear wave UT techniques.

The Oconee reactor vessels are the prototype vessels for the B&W supplied 850-MWe class of reactor vessels. However, no unusual design or fabrication problems either before or during manufacture have been identified. We have concluded that the reactor vessels as designed and fabricated are acceptable.

4.2.1 Reactor Vessel Material Surveillance Program

The estimated end-of-life neutron fluence for each of the three reactor vessels is 2.2×10^{19} nvt, based on a 40-year service lifetime and a load factor of 0.80; however, B&W has selected a design value of 3.0×10^{19} nvt. B&W has verified their calculational model, the NRN Code, through comparison of predicted results with those obtained from three separate nuclear experiements. On the basis of our past reviews of other plants and the experimental verification obtained on the calculational model, we have concluded that the neutron fluence value used for the design of the reactor vessels is acceptable.

B&W Topical Report, Reactor Vessel Material Surveillance Program, BAW-10006, Revision 1, dated May 1970 has been incorporated in the FSAR by reference. The program described in the report meets or exceeds our requirements in regard to compliance with ASTM E-185, number of capsules and contents of capsules. The program does not meet the requirements of the AEC proposed Fracture Toughness Criteria, Reactor Material Surveillance Program Requirements. The AEC criteria specify a scheduled withdrawal program of four capsules from each of the three vessels; the proposed program schedules a withdrawal of three capsules from Unit 3 as part of an integrated plan tying all 11 B&W plants into a single program. We informed the applicant and B&W at a meeting on June 18, 1970 that we will require individual, four capsule, scheduled withdrawal programs for each vessel. A suitable program will be required by the technical specifications.

4.3 Once-Through Steam Generators

The Oconee Nuclear Station will be the first plant to utilize the B&W once-through steam generator. The design of this steam generator has several unique features which require careful review:

(1) It has two horizontal tube sheets (top and bottom) into which are welded approximately 15,000 straight vertical tubes. This is in contrast to the single tube sheet, U-tube design used in other PWR steam generators.

(2) The tube bundle is exposed to steam. The steam/water level interface varies with load.

(3) It incorporates a feedwater heater by spray-mixing feedwater with steam extracted above the steam/water level interface.

(4) It makes use of an adjustable (fixed at installation) feedwater orifice to provide means for eliminating oscillations in reactor coolant temperature should they develop during the operating life of the steam generators.

(5) There is no secondary blowdown. Instead reliance is placed on "full flow" Powdex polishing demineralizers upstream of the feedwater train. These demineralizers treat 70% of feedwater flow at full power.

(6) Auxiliary feedwater is injected directly into the "dry" top of the tube bundle.

We are presently evaluating additional information made available to us in July, 1970 in response to concerns raised during our review. We expect to include our conclusions on this matter in a subsequent report.

4.4 Internals Vent Valves

In order to prevent steam binding of the core for cold leg pipe ruptures, B&W has incorporated eight 14-inch diameter hinged disc vent valves in the core support shield which separates the upper plenum reactor outlet region from the annular reactor coolant inlet region. In the event that steam pressure buildup in the upper plenum region exceeds the pressure in the annular region by less than 1 psi during a cold leg break, the valves will open, relieving the steam pressure and permitting emergency core coolant to flow upward to recover the core.

The B&W evaluation of these valves was initially presented in a proprietary Topical Report, Internals Vent Valve Evaluation, BAW 10005, dated July 1969, which the applicant incorporated in the FSAR by reference. We reviewed that report and requested that additional information be submitted to better document the development and testing performed on these valves. A revised version of the Topical Report was submitted on June 23, 1970. The report describes the results of extensive analytical and experimental evaluations performed to demonstrate the capability of the valves to perform as intended under accident conditions. On the basis of our review of this information and assuming that our present discussions on ECCS codes do not affect the required venting capacity, we have concluded (1) that the vent valves will perform their intended function without significant vibration during normal operation; (2) that they have adequate capacity even with one valve failed closed to permit the core to be recovered; (3) that the plastic deformation predicted upon impact with the vessel wall when initially opening under LOCA conditions is acceptable; (4) that it can and will be demonstrated that the valves can be inspected, removed, and replaced after installation; (5) that during normal operation, core bypass flow caused by valve seat leakage, would not be significant with all valves in place; and (6) that core bypass (4.6% reduction in core flow) with one valve disc completely removed is acceptable. (Page 3-56 of the FSAR shows DNBR reduced from 1.55 to 1.40 at 114% of rated power by this flow reduction.)

4.5 Missile Protection and Flywheel Integrity

The applicant has described the evaluations made to assess potential internal missiles and the measures taken to protect the plant against missiles. We have reviewed the information provided by the applicant, including that given in Amendment No. 12 in response to request 4.13.2. We have concluded that the design features incorporated in the plant will provide adequate protection to the primary system, other vital systems, and the containment liner from missile hazards.

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

4.6 Inservice Inspection

The applicant will apply the rules of Section XI of the ASME Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as the basis for the proposed inservice inspection program. We informed the applicant at a meeting on June 18, 1970, that we will require that the inservice inspection program be upgraded, where practicable, to the level of the programs recently approved for the Palisades and H. B. Robinson plants. We will require this subject to be addressed in an acceptable manner by the technical specifications, and, on this basis, we have concluded that the Oconee inservice inspection program will meet our requirements.

4.7 Leak Detection

The reactor coolant pressure boundary leak detection systems for Oconee include air particulate monitoring, radiogas monitoring, and containment sump level monitoring. Humidity detection is not included. The Oconee array of instrumentation is redundant, diverse, and provides timely alarms. The applicant contends, and we concur, that the system is at least as sensitive as those to be provided in plants recently reviewed for constrution permits such as the Davis-Besse and Trojan facilities.

We will require the technical specifications to address positive surveillance methods, minimum instrument sensitivities that must be maintained, the nature of safety evaluations to be performed upon detection of any leak, and the time permitted to complete these evaluations prior to specific mandatory action.

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

4.8 Sensitized Stainless Steel

We have reviewed the Oconee reactor coolant system to determine the extent to which furnace sensitized stainless steel is present in pressure retaining and structural members.

Stainless steel safe ends and thermal sleeves have been welded to certain carbon steel nozzles only after the vessels were stress relieved. The primary coolant piping is fabricated of low alloy carbon steel and consequently the main coolant nozzles on the reactor pressure vessels and steam generators do not require stainless steel safe ends.

The stainless steel reactor coolant pump casings and all thick wall stainless steel pipe sections are water quenched from solution heat temperatures. The stainless steel flow vanes and internals guide lugs are attached to the reactor vessel after the final stress relief of the vessel.

Shop welding of stainless steel safe ends to Inconel buttered joints was done using a low energy heat input. Field welds were made by the shielded metal arc process which also used low energy heat input.

On the basis of our review we have concluded that shop and field procedures used in fabrication of the primary system have resulted in a system that contains relatively little sensitized stainless steel, and that the system, as fabricated, is acceptable.

4.9 Other Class I (Seismic) Mechanical Equipment

Quality control standards for engineered safety feature equipment are briefly summarized below:

All piping meets the requirements of Class II or III of ANSIB31.7.

All welding procedures and operators concerned with the fabrication of pumps and valves have been qualified to Section IX of the ASME Boiler and Pressure Vessel Code.

Core flooding tanks and the tube side of the low pressure injection heat exchangers are ASME Class III.

Hydrostatic tests of valve bodies and valve seats were conducted in accordance with ANSI B16.5 and MSS SP-61. The ECCS pumps have been hydrostatically tested to the requirements of UG-99 of Section VIII-Division 1 of the ASME Code. Low pressure service water pump ASME Code hydrostatic tests are also performed.

The quality control standards for valves require inspection of raw material and review of material certification in conformance to ANSI B31.7 requirements. In addition, radiography and liquid penetrant tests of valve bodies, valve bonnets are performed to meet ANSI B31.7 acceptance standards. ASME III liquid penetrant tests are performed on all pumps and the pump barrel of the HP injection pump is examined ultrasonically.

These requirements result in a fabrication and inspection program which contains the essential elements of the ASME Code for Pumps and Valves for Nuclear Power. We find these requirements acceptable.

The codes and standards applicable to other Class I (seismic) systems have been reviewed and are considered adequate from a safety standpoint.

All Class I (seismic) equipment has been designed to withstand the design basis earthquake without loss of function. We and our seismic design consultant are continuing our review of the analytical procedures used to calculate the seismic loadings on Class I (seismic) equipment, the adequacy of the applicant's check on the vendors' methods of certification, the design organizations involved in seismic design and their responsibilities, and the documented procedures to provide for the interchange of design information between the involved organizations. We will provide our conclusions on this review in a subsequent report.

5.0 CONTAINMENT AND CLASS I STRUCTURES

5.1 General Structural Design

The containments or reactor buildings are similar to those of other PWR facilities recently reviewed for operating licenses such as the Palisades and Point Beach Units, for which Bechtel was also the architect-engineer. The containment is principally of prestressed concrete construction with design details similar to those of the Point Beach and Palisades containments. The structures interior to the containment are of massive reinforced concrete construction. The auxiliary building is of reinforced concrete column, beam, and slab construction. The turbine building is of structural steel with panel siding.

The loads, their combinations and methods of application are in accord with what is considered to represent current good practice, and are acceptable.

5.2 Containment Structural Design and Design Analysis

The designer has employed analytical techniques that account for thick-walled areas (such as the ring girder region), the influence of the foundation, and the creep and cracking properties of concrete. Special attention has been given to the design of the liner, buttress, and large openings. The stresses and deformations have been kept to very conservative values. (See Table 5-3A in the FSAR.)

5.3 Design and Design Analysis of Other Class I Structures

In the design of Class I (seismic) structures other than the containment, due care has been exercised. The interior shield structure design, as an example, shows consideration of important loads such as differential pressure across compartments and inertia loads resulting from postulated seismic ground motion.

5.4 Seismic Design and Design Analysis

The principal structures, consisting of the containment, the internal structures, the auxiliary building, and the turbine building, have been dynamically analyzed by the response spectrum technique using appropriate shock spectra. The Class I (seismic) piping outside the containment has been analyzed using equivalent static methods. These methods have been conservatively applied.

5.5 Structural Materials, Construction Techniques, and Quality Control

The structural materials, construction techniques, and quality control employed have been indicated in some detail in the FSAR. Based on this information, and the reports from the Division of Compliance describing the manner in which the quality control program is being carried out in the field, we believe that an adequate quality product is being achieved. We have identified instances where controls have failed, resulting in the need to evaluate the consequences of such failure on the final product. Two instances of such failure are construction of the Unit 1 shell with omission of two horizontal tendons and the incorporation of concrete which failed to meet its 28-day break test strengths. Both of these areas are currently being assessed and our conclusions will be given in a subsequent report.

5.6 Structural Acceptance Testing and Surveillance

The applicant has described the structural test program and the containment structural surveillance program. Nine tendons of one containment have been selected for surveillance. The tendon monitoring program consists essentially of monitoring of tendon force by lift-off readings and sampling of selected tendons: first, at 5 years after plant startup and, thereafter, at an interval to be determined when the results of the initial surveillance test are known. Bechtel is in the process of completing a statistical study to substantiate the appropriate frequency, numbers, types and positions of tendons for a containment of this type. We will require that the results of this study be incorporated into the program for the Oconee units and, also, that visual inspection of the containment interior including the liner and penetrations be performed as a part of the structural surveillance program. The program, as proposed and to be modified as indicated above, is considered acceptable.

Monitoring for seismic ground motion will be provded as described on page 5-59a of the FSAR. The monitoring system is completely self-contained, will detect and record three-plane accelerations above a 0.01g threshold, and will have an alarm at 0.05g. The system will be located in the tendon access gallery. We conclude that this system is adequate.

5.7 Reactor Cavity Design

The reactor cavities are designed to hold water up to the level of the reactor nozzles, 10 ft above the top of the core. The ability to withstand a major pipe break within this cavity was investigated at the construction permit review stage. At that time the applicant reported that an 8-ft^2 break would cause a 195 psi differential peak pressure acting outward on the cavity walls. As reported in the FSAR, the cavity is designed to withstand a 208-psi differential pressure without loss of flooding capability. This compares favorably with other current generation reactor cavity designs such as those for the Sequoyah plant which are designed to withstand 120 psi differential pressure without loss of flooding capability (there, a 4.5-ft^2 equivalent pipe break inside the cavity will result in 100 psi differential pressure acting on the cavity walls). Also, it is reasonable to assume that, as for other designs, the reactor cavity will be capable of withstanding higher peak pressure differentials without gross failures; while it will remain structurally intact, it may not be floodable.

Because of the advanced stage of construction, we have not investigated the effects of the 14.1 ft^2 double-ended break of a hot leg on the Oconee reactor cavity structures.

On the basis of our review, we have concluded that the reactor cavity design for the Oconee units is acceptable.

5.8 Penetration Room

We have reviewed the design of the penetration room provided for each unit and have concluded that the negative pressure produced in the sealed room by the two independent fans will assure that essentially all post-accident containment leakage into this room will pass through one of the two particulate-absolute-charcoal filter trains which are located in the equipment ventilation room directly above the penetration room. As noted in Section 5.9 below, we have assumed that the radioactivity content of containment leakage processed through this room will be substantially reduced by these filters prior to discharge from the unit vent. There are deficiencies in this system at present:

(1) There is no control room indication of filter flow or filter assembly pressure drop.

(2) The design requires an operator to go to the ventilation room, in the vicinity of the filter assemblies, under accident conditions to either correct for reduced flow due to filter crud buildup or to reestablish flow by valve realignment in the event that flow is lost in one of the filter assemblies.

We will require the applicant to provide IEEE-279 grade instrumentation in the control room to monitor the filter flow and to provide assurance (1) that the operator can manually adjust or align the necessary valves in the ventilation room without undue radiation exposure to himself and (2) that this can be done soon enough to prevent excessive temperatures in the charcoal filters. We will discuss the resolution of these matters in a subsequent report.

5.9 Containment Leakage

The applicant has agreed to an integrated leak rate under accident conditions of 0.25 w/o per 24 hours or less for Unit 1. The technical specifications will require that no more than 50% of this leakage leave the containment without passing through the penetration room charcoal filter system. Verification of the permitted leakage distribution will be based on local leakage testing of individual penetrations.

Based on our review of the containment design and the provisions available for testing of individual penetrations, we have concluded that the applicant can assume that one-half of the containment leakage will be processed through the penetration room charcoal filters.

5.10 Isolation Capability

In Section 5.2.2 of the FSAR the applicant lists four classifications of isolation for all fluid penetrations. We find these classifications acceptable. We have reviewed the individual system flow diagrams and have concluded that the isolation methods used are in conformance with the applicant's classification requirements and that no single failure of an active component (including valve actuation instrumentation channels where applicable) which must function in an accident situation could result in loss of isolation or intolerable leakage.

We have also reviewed the analyses, provided in Sections 6.5 and 14.2.2.4.4 of the FSAR which account for auxiliary building radioactive leakage from the low pressure injection system valves, flanges, and pump seals during post-accident recirculation of the emergency core coolant. Section 6.4.2.1 of the FSAR lists only three types of fluid penetrations which are not vented to the penetration room or do not pass through it. These are the main steam lines, the refueling tubes and reactor building normal sump drain and emergency sump recirculation lines.

Steam line releases are separately discussed in Section 9.6 of this report. The refueling tubes exit into the bottom of the spent fuel pit under several feet of water. Leakage that might escape by this route would be reduced by a water partition factor and should not contribute significantly to potential accident doses. Under accident conditions the normal sump drain will have a water seal to minimize leakage. The emergency sump recirculation line leakage was included in the analyses given in Section 6.5 of the FSAR.

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

5.11 Missile Protection

The turbine missile analysis presented in the FSAR was modified slightly to take into account a smaller last stage wheel missile than analyzed in the PSAR (5944 lbs vs 6600 lbs), having smaller impact areas ("side on" -8.368 vs 8.89 ft^2; "end on" -3.657 vs 3.91 ft^2), and higher impact kinetic energy (cylinder -22.5 x 10^6 vs 20.1 x 10^6 ft-lbs; dome-18.0 x 10^6 vs 16.48 x 10^6 ft-lbs). The net results did not change the three conclusions given in the PSAR for the "side on," Case (I) "end on," Case (II), and no loss of initial kinetic energy, Case (III) events. The information given in Table 5.11-1 compares the capability of the reactor building structures to withstand the loss of tendons with the number of tendons that could be struck by the assumed turbine missile.

TABLE 5.11-1

No. of Adjacent Tendons That Can Be Lost	Tendon Location Dome	Vertical	Horizontal
Building design capability	5	3	3
Case I "side on" (tendon strikes)	0	0	0
Case II "end on" (tendon strikes)	2	1	3
Case III no turbine casing attenuation (tendon strikes)	5	1	3

We were informed during the ACRS site visit on June 23, 197) that two horizontal tendons were left out of the Unit 1 structure because construction personnel failed to install two tendon sheaths at the construction opening level. This matter is being investigated by the applicant for possible consequences on the missile withstanding capabilities of the Unit 1 structure. We will review the results of this investigation and provide our conclusions in a subsequent report.

The applicant's tornado missile analysis remains unchanged from that given in the PSAR.

The reinforced concrete spent fuel pit is designed to prevent entry of both turbine and tornado missiles as noted in Table 5-5 of the FSAR. This information also confirms that given in Supplement 4 of the PSAR to Request 11.1 which indicates that vital components will be protected by concrete walls and roofs designed to prevent missile penetrations or will be separated to prevent failures in redundant systems from potential missiles.

Based on the above, we have concluded that the reactor building and other Class I (seismic) structures, components and systems have been adequately protected against potential missiles.

6.0 ENGINEERED SAFETY FEATURES

6.1 Emergency Core Cooling System

During the post-construction permit period, the applicant revised the Oconee ECCS design to comply with our General Design Criterion 44 (July 1967). This involved providing a second recirculation line from the emergency reactor building sump to the low pressure injection pumps, providing an extension of the reactor building containment beyond the sump valves to accommodate a valve body failure, and providing valved tie lines at appropriate places in both the high pressure and low pressure injection systems. We conclude that these revisions to prevent loss of function due to single failures are satisfactory.

Recent computer program developments in modeling the reactor emergency core cooling systems performance for PWR plants have raised uncertainties about past predictions of ECCS functional performance for large cold leg breaks. We have requested the applicant to provide additional information obtained with the use of the more advanced computer program codes now available. In addition we are using the Commission-developed RELAP-3 code to obtain independent data on the Oconee ECCS performance. We have also asked the Idaho Nuclear Corporation (INC) to perform independent analyses for the Oconee plant using the Commission developed RELAP-3B code. We will report the results of our evaluation in a subsequent report.

6.2 Post-LOCA Gas Evolution and Control

Our evaluation of the hydrogen control requirements for Oconee is not complete. The applicant has proposed to control the post-LOCA reactor building hydrogen concentration by purging through charcoal filters to maintain a 3.5 v/o hydrogen concentration. The applicant estimates that the post-LOCA dose at the LPZ (6 miles) will be increased by 4 Rem to the thyroid due to the purging operation.

The containment volume-to-reactor power ratio for the Oconee plants is 0.74. For a plant of this type, and using our assumptions for evolution of hydrogen, we have estimated that the dose due to purging would be on the order of 30 Rem to the thyroid at the site boundary (1 mile). This is a gross estimate, but is indicative of the order of magnitude to be expected for the dose due to purging from the Oconee plant.

Since the Commission has not yet considered the joint position reached by the staff and the Committee on the matter of hydrogen control, we are not taking a final position on this matter with any applicant. We expect that the Commission will consider this matter in the near future. If the Commission approves the proposed position, we will require the applicant to install the necessary engineered safety features to comply with our joint position.

In the interim, we have directed the applicant's attention to the applicable sections of recent Committee letters in which the hydrogen matter is discussed. We have also informed the applicant of the details of our current thinking that are not evident from the Committee's letters, and of the status of the remaining steps we are presently taking to reach a final position on this matter.

We will present any further developments on this matter in a subsequent report.

6.3 Other Engineered Safety Features

Our review of the containment cooling and spray systems is incomplete at this time. We have requested a limited amount of additional information on these systems which we expect will enable us to conclude that these systems are satisfactory. We will report the results of our evaluation in a subsequent report.

7.0 INSTRUMENTATION, , AND POWER SYSTEMS

7.1 General

The design crotection system, which consists of the reactor trip system and tneered safety feature actuation system, is identical for all three Occts. Each Unit's protection system is completely independent excephe shared 125 Vdc instrument power system which is discussed later ireport. Our review included a detailed study of the schematic diaf the reactor trip system and the actuation circuitry of the eng safety feature systems.

Conformance protection system to the Commission's proposed General Design Cr(GDC), as published in the Federal Register on July 11, 1967, anroposed IEEE Criteria for Nuclear Power Plant Protection System-279) dated August, 1968, served, where applicable, as the principal or our conclusion that the protection system is, except as discussr in this report, acceptable.

7.2 Reactor Protectiom

The reactor ion system consists of four identical channels, each of which utieneral logic and de-energizes (trips) upon detection of any one oonditions listed in Table 7-1 of the FSAR. Each channel terminatereactor trip module which controls one or more breakers in the crod drive power system. The system logic is two-out-of-four; i.e.y two protection channels trip, all reactor trip modules trip, com all control rod breakers to trip. The entire system, from the sensors to the control rod breakers, is testable during reactor op.

(1) Bypassing

Section 7.1.f the FSAR discusses the three means by which various reacp signals can be bypassed. Based on our review, we conclude ministrative controls provide the only significant protectinst improper use of these bypasses. Our evaluation of each three bypass provisions is discussed below:

(a) Channıs Switches: Section 4.11 of IEEE-279 permits one channı bypassed during reactor operation but positive meansıring that the remaining portion of the protection systeıues to meet the single failure criterion are not speci:required. Although it is possible to completely bypasstomatic portion of the reactor trip system, we conclı administrative control of the number of channel bypass keys (one per reactor unit) and of the number of channessed concurrently (one per reactor unit), togetł indication of the channel which is bypassed, meets ent of IEEE-279 and is acceptable. The technical specil will require that no more than one trip channel be byponcurrently.

(b) Shutdcss Switches: Although a pressure interlock preverof these switches during power operation, the applic stated that, in order to provide adequate protecring physics testing and control rod drive testinhigh power level trip setpoints must be lowered. The ap proposes to change the setpoints manually. As noted FSAR, the reactor will first be shut down before these s are used to permit control rod drive and core physicng. We have not yet decided whether to concur with ticant's intent to manually change the setpoints, or to that system modifications be made so that the necesspoint changes are obtained automatically. The questised on this matter are similar in many respects to the qu raised (in context with our reviews of several other tions, but not the Oconee application) as to whetheıanges in setpoints required for operation with lı all four primary coolant pumps, can be made manually or should be made automatically. This issue is to be the subject of an ACRS Subcommittee meeting that is presently scheduled prior to the August meeting of the full Committee.

We intend to inform the Committee of our conclusions on this matter in a subsequent report. Any guidance obtained from the Subcommittee meeting to be held on this general subject, or from our discussions with the Committee at the August meeting will be considered in the development of our final position.

(c) Dummy Bistables: Dummy bistables, which bypass the individual input signals, can be installed in each reactor trip channel. As presently proposed, there is no provision to indicate either the number of dummy bistables installed or the instrument channel in which they are installed. Although we are unable to report our final position on the use of the dummy bistables, there are only three alternatives presently under consideration: (1) if the design is not changed, we would not permit the use of dummy bistables; (2) the applicant has stated that the design could be easily changed to provide indication of the trip channel, but not the instrument channel, in which dummy bistables are installed. If this design change is made, the use of dummy bistables in one trip channel at a time would meet IEEE-279. Concurrent use of a channel bypass switch and dummy bistables would not meet IEEE-279; (3) if the design is changed to meet our interpretation of the IEEE-279, i.e., the status of protection system is continuously, and in a non-ambiguous manner, indicated to the operator, we could permit the use of dummy bistables within the Technical Specification requirements for a minimum of two operable instrument channels per trip parameter with the trip channels arranged in a one-out-of-two trip logic.

(2) Instrumentation for Operation With Less Than Four Reactor Coolant Pumps (Partial-Loop Operation)

The design of the reactor protection system includes provisions for operation with less than four reactor coolant pumps in service. The applicant contends that operation with three pumps running requires no adjustment of protection system setpoints because the power/flow trip can provide a means of protection. An automatic setpoint change is made when only one pump in each loop is in operation; this setpoint change limits reactor operation to less than 55% of rated power. Loss of two pumps in the same loop will cause a reactor trip regardless of power level. In order to resume operation with only the other two pumps in service, the applicant proposes to manually change several protection system set points. Operation with only one pump in service is not proposed.

As noted in Section 3.3 of this report, we have not completed our review of the thermal-hydraulic concerns associated with partial-loop operation (less than four pumps). We are not yet convinced that operation with less than four pumps is acceptable from a thermal-hydraulic viewpoint or that the thermal margins are known well enough to reliably establish limits for such operation. Until these questions are resolved we cannot take a final position on the instrumentation aspects of partial-loop operation. This issue is expected to be resolved in the near future and we will provide our conclusions to the Committee in a subsequent report.

(3) Reactor Coolant Flow Instruments

We have not completed our review of the reactor coolant flow instruments. A total of eight differential pressure transmitters are used to provide inputs to the reactor protection system. The four transmitters associated with each loop derive their input from a common hot leg flow nozzle which has one pair of pressure taps serving all four transmitters. We have informed the applicant that, based on the information provided to date, we cannot conclude that these instruments meet the single failure requirement of IEEE-279. One basic problem is that failure of either common pressure tap or associated instrument tubing can cause loss of protection against low flow. We have asked for additional information describing the pressure tap and instrument tubing sizes, layout and valving arrangements. We also asked that we be provided with a complete failure analysis of these common elements as well as specific information on how all unsafe failures are identified, the time required to make the identification and specific correction actions planned. We expect to be able to satisfactorily resolve this matter and provide our conclusions in a subsequent report.

We have concluded that, except for the item discussed in (1)(b) and (2) above, the reactor protection system meets the proposed GDC and IEEE-279 and is acceptable.

7.3 Engineered Safety Feature Actuation Systems

The engineered safety feature actuation system consists of eight channels. Two independent channels are provided for each engineered safety feature system by using a "split-bus" concept (e.g., from FSAR Table 7-3, Channel 7 activates only spray system A and Channel 8 activates only spray system B).

The emergency core cooling systems, i.e., high pressure injection and low pressure injection, are actuated from the sensing of either low reactor coolant pressure or high containment pressure. The applicant has stated that, for some break sizes, a reactor trip is required for the emergency core cooling systems to be effective. To date, the applicant has not clearly established the availability of a diverse reactor trip which was required by our letter of February 13, 1970. That letter stated that all functions required for effective emergency core cooling of the Oconee reactors should be actuated from the sensing of diverse variables. We are awaiting the necessary additional information which will demonstrate that the required diverse reactor trip signal is available to be used in conjunction with high containment pressure ECCS actuating signal.

We have reviewed the schematic diagrams and the test procedures for the engineered safety feature actuation circuits with the applicant. In view of our concerns with the test capability provided by the Westinghouse design, we wish to point out some features of the B&W design. The entire system, from the sensors to the actuated components (e.g., pumps, valves) and including the bypass provisions, can be tested during reactor operation. During the periodic tests, the channel under test is not incapacitated and a valid trip signal will actuate both channels associated with each engineered safety feature system. Each actuated component has its own unit control module. The unit control modules are the equivalent of the Westinghouse slave relays except that each slave relay actuates several components. Although the B&W design, like the Westinghouse design, does not permit an integrated system test during reactor operation, the individual components can be actuated one at a time using the associated unit control module in a manner which adequately duplicates the action required under accident conditions. We conclude that an acceptable means of completely testing the ESF actuation circuits during reactor operation is provided.

We have concluded that, except for the lack of evidence of the required diverse reactor trip signal, the engineered safety feature actuation system meets the proposed GDC and IEEE-279 and is acceptable.

7.4 Offsite Power

Offsite power is available to each unit from the 230-kV switchyard via the three 230/4.16-kV startup transformers. Eight 230-kV transmission lines (four installed with Unit 1; two added with Unit 2; two added with Unit 3) converge at the site via several rights-of-way. The 230-kV switchyard is arranged into a breaker-and-a-half configuration and each circuit breaker is provided with dual trip coils supplied from the two independent 125 Vdc station switching power systems. Circuit protection is provided by redundant relaying. Commencing with the operation of Unit 3, the 500-kV switchyard will be connected to the 230-kV switchyard via an autotransformer. The applicant has stated that the Duke system is designed to withstand the loss of any generating unit within its network. They will demonstrate 100 percent load rejection prior to commercial operation with Oconee Unit 1.

Our review indicates that the only portion of the offsite power system vulnerable to a single random failure is the single startup transformer for each unit. Prior to the operation of Units 2 and 3, the only source of offsite power for Unit 1 is via its startup transformer. We have accepted single startup transformers for three previous applications: Ginna, H. B. Robinson, and Palisades. This arrangement was accepted for those plants because of the reliability of such transformers. An additonal consideration in the case of Oconee Unit 1 is the fact that the single startup transformer circuit will exist for only about 1 year. With the operation of Units 2 and 3, additonal sources of power can be made available through manual breaker operations which connect another unit's startup transformer to the emergency buses of the affected unit.

Although the offsite power system does not fully meet the proposed GDC 39, it will meet draft Criterion 17 after Unit 2 begins operation.

Based on our review, as discussed above, we have concluded that the offsite power system is acceptable.

7.5 Onsite Power

Onsite power for Units 1, 2, and 3 is provided by two 87.5-MVa hydroelectric generators. This power is available either through the 230-kV switchyard and the Unit 1, 2, or 3 60-MVa startup transformers or through the 13.8-kV underground feeder which utilizes its own 20-MVa 13.2/4.16-kV transformer. Thus either hydro unit has ample power, via either circuit, for operation of the engineered safety feature loads of one unit plus the safe shutdown loads of the other two units.

Figure 8-2 of the FSAR shows the arrangement of the station's main buses. Three engineered safety features 4.16 kV buses are provided for each Unit and these buses are connected to both of the unit's 4.16 kV main feeder buses. The sources of power which are automatically connected to the main feeder buses, in the order that they are connected, are:

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

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

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

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

(1) Another Oconee unit's startup transformer via the station's emergency startup bus (as they become available);

(2) Another Oconee unit via the standby buses (as Units 2 and 3 are completed); and

(3) One of the three gas turbines located 30 miles away at Lee Steam Station via an independent overhead 100 kV transmission line which can be separated from the rest of the Duke transmission system and the standby buses (available for Unit 1).

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

While the Oconee system obviously has many sources of power available, an aspect of the design which would not be acceptable in a current construction permit application is the lack of independent load groups. Regardless of the source of power, the three redundant engineered safety feature buses are connected in parallel through the two main feeder buses. All other recently approved facilities have provided two or more electrically indpendent load groups, each with its own source of emergency power (split-bus). We have asked the applicant to submit an analysis of the Oconee design to show that the independence and reliability of the redundant engineered safety feature loads are comparable to the independence and reliability provided by a split-bus design. At present we believe that the applicant will be able to show that the Oconee design is acceptable, based on the large number of power sources, the relatively large capacity of these sources, and the high reliability of the hydro units.

One feature of the onsite distribution system on which we and the applicant have been unable to reach agreement involves the automatic transfer of power to redundant motor control centers. As presently proposed, the three ESF 600 volt motor control centers receive power via an automatic transfer device from two of the three 4160 volt engineered safety feature buses. We asked the applicant to identify those loads which require this automatic feature in order to meet the design bases. The only load so identified is one of the three reactor building fan coolers. However, it appears that if one fan cooler were connected to each of the three ESF buses, the design bases would be met without automatic transfer. It is our opinion that the use of the automatic transfer feature unnecessarily reduces the already limited independence of redundant engineered safety feature equipment. We will require that the design be changed to eliminate the automatic transfer of loads between redundant engineered safety feature buses.

The arrangement of the 125 Vdc Instrumentation and Control Power System for Unit 1 is shown in Figure 8-5 of the FSAR. Each of the four distribution panels associated with a particular unit receives power via diode assemblies from either of two 125 V battery buses, one in the associated unit and one in another unit. (The 125 V batteries for both Unit 1 and Unit 2 will be available prior to operation of Unit 1). Therefore, the source of power to each panel is automatically transferred, albeit in a unique manner, between redundant buses. Our concerns with the use of automatic transfer devices connected between redundant dc buses were most recently discussed in our report to the Committee on the Point Beach facility. Our conclusion that the Oconee design is acceptable does not conflict with our position that a split-bus design should be used. Our conclusion that the use of isolating transfer diodes is acceptable for the Oconee units is based on the following:

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

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

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

In summary, the Oconee 125 Vdc system design is unique in the respect that the large number of batteries, together with the capability of immediately detecting failures, provides a system which can withstand not only a loss of power to any single load group supplied via an automatic transfer device, but also the loss of both sources of power to the transfer device.

Based on our review, we conclude that, if the automatic transfer of power to the 600 V motor control centers is eliminated, the onsite power system meets the proposed GDC 39 and is acceptable.

7.6 Cable and Equipment Independence

We have reviewed the applicant's installation criteria relating to the preservation of the independence of redundant safety equipment by means of separation as well as that relating to the prevention of fires through derating of power cables and proper tray loading. We have found these criteria to be acceptable. We intend to visit the site for the purpose of reviewing the implementation of these criteria after a majority of the protection system equipment has been installed.

7.7 Environmental Testing of Equipment

In Section 6.1.2.12 of the FSAR, the applicant has listed the equipment which must be operable during and subsequent to an accident and has described some of the environmental tests performed on this equipment. We have reviewed this information and conclude that the test program is acceptable. However, we have requested the applicant to provide a brief description of the tests used to qualify the sensors which provide input signals to the protection system. We expect to receive this information prior to the ACRS meeting and will assure ourselves that these tests adequately simulated the post-accident environment.

7.8 Seismic Testing

The applicant's seismic design bases are that the protection systems shall function normally during and after the design basis earthquake.

The protection system equipment is being dynamically tested to show normal operation during excitation in excess of the maximum predicted DBE seismic accelerations and frequencies.

We have evaluated the applicant's seismic design bases and conclude that they are acceptable.

8.0 AUXILIARY SYSTEMS

8.1 General

The auxiliary systems are supporting systems required for normal plant operation. Those systems and features discussed below were selected for review on the basis of their importance to safety.

8.2 Radioactive Waste Management

As stated in the FSAR, liquid radioactive wastes are segregated in receiving tanks according to their source, and then sampled, analyzed, and treated. Based upon the analyses, the liquid wastes will be treated by one of the following methods:

(1) discharge to the Keowee Hydro Plant tailrace;

(2) discharge to the Keowee Hydro Plant tailrace after holdup for decay;

(3) concentration by evaporation with ultimate disposal as a solid waste.

Also,according to the applicant, liquid wastes will be diluted, as necessary, by the hydro plant discharge (from 30 cfs due to minimum leakage flow with the hydro plant shutdown to 19,800 cfs with the plant in full operation) to meet the concentration limits of 10 CFR 20. However, in order to retain operational flexibility, the applicant will assume the minimum dilution (30 cfs) in estimating the annual release limits. According to the applicant, the Keowee Hydro Plant, which is controlled from the Oconee nuclear plant control rooms, is expected to be operated at least weekly, if not on a daily basis. Thus, flow substantially greater than the minimum leakage flow will usually be available.

In the applicant's design evaluation of the Oconee liquid radwaste treatment system, a minimum holdup time of 30 days was assumed. The amount of waste storage tankage that is available in the plant includes 8,100 ft^3 for the miscellaneous liquid wastes and 66,000 ft^3 for the primary coolant (coolant storage system).

The applicant proposes to treat all liquid radioactive wastes, for which the radioactivity concentration without processing other than dilution with the Hydro Plant leakage at the point of release would be greater than 1/10 MPC, so as to reduce the quantities released to as low a level as practicable. We will require that all radioactive liquid wastes be treated with the waste evaporator. In addition, in view of the available holdup capacity in the Oconee plant and the expected operating routine of the Keowee Hydro Plant, we have concluded that the liquid radwaste should normally be discharged only if the hydro plant is being operated. Thus, the waste would receive a much greater instantaneous dilution than if the hydro plant was not operating.

The entire radwaste system is located below grade in Class I (seismic) structures and in the event of an accidental spill, the liquid waste will be retained within the structures. In order for accidental discharges to the environment to occur, wastes would have to be pumped from the below grade storage tanks to the Keowee Hydro Plant tailrace through a discharge valve, which is closed by a high radiation signal from a radiation monitor. We concur with the applicant that this is a highly improbable event and need not be considered as a design basis for this facility.

The reactor coolant treatment system, shown on Figure 9-15 in the FSAR, is provided to recover boron and to purify the coolant. It can also provide waste treatment of the primary coolant sent to the liquid waste disposal system. The coolant treatment system, which operates on a batch basis, receives liquid from the primary coolant bleed holdup tanks through bleed evaporator demineralizers (deborating demineralizers). The coolant is fed to the coolant bleed evaporator, and then the condensate is sent to the condensate test tanks. At the operator's option, the contents of the test tanks can be (1) routed to the waste evaporator feed tank, (2) returned to the reactor coolant evaporator feed tank, (3) returned to the coolant bleed holdup tanks, or (4) released to the liquid waste effluent header. In conjunction with options (3) and (4), the test tank effluent can be passed through the condensate demineralizers. For option (4) we will require use of the demineralizers.

The solid waste disposal system includes equipment to collect and store spent demineralizer resins accumulated during 2 years of operation, and a hydraulic press for use in handling compressible solid wastes. All solid wastes are ultimately drummed and shipped offsite for final disposal.

The gaseous waste disposal systems are connected to vent headers which collect potentially radioactive off-gases from all components which may contain radiogases. Before release to the environment, these off-gases are processed either by passage through a waste gas exhauster and a filter bank composed of a prefilter, an absolute filter and a charcoal filter, or by a period of retention in the waste gas decay tanks prior to passage through the filter train. The gas decay tank contents are sampled and analyzed prior to release to the plant vent. Units 1 and 2 share a gaseous waste disposal system and Unit 3 has an independent system. However, these systems can be interconnected through double isolation valves between the respective vent headers; thus, operational flexibility is provided in the event one waste gas system is temporarily out of service.

According to the FSAR, the waste gas exhauster will be used when large volumes of gas containing little or no radioactivity are to be released. The exhauster (fan) and the isolation valves on the waste gas exhauster and decay tank discharge lines are interlocked with a radiation monitor so that in the event of high level radioactivity, discharge through these paths will be terminated.

Normal purging of the containment will pass the discharged air through particulate (HEPA) and charcoal filters to the plant vents. However, under conditions of low level radioactivity in the containment air, the applicant proposes to bypass the filter system.

We will require that gaseous radioactive wastes passing through the off-gas system be held for decay (the holdup times will be established in the technical specifications) and that all radioactive gaseous wastes, including the containment normal purge flows, be filtered prior to discharge.

The site areas of nuclear facilities are often used for public recreational purposes, visitor centers, and other purposes that result in members of the public being present more or less routinely within the confines of the exclusion radius. For Oconee, for example, there will be a visitor center about 300 meters from Unit 1, construction personnel will continue to work on Units 2 and 3 after Unit 1 is in operation and Unit 3 when the other two are in operation. Within the exclusion radius temporary quarters are provided for some construction workers, and a portion of Lake Keowee within the exclusion radius will be used for recreational purposes. With the increase in the number and size of nuclear plants, site areas are being used by greater numbers of the public for more varied purposes.

We have not as yet reached a position on the problem as it relates to the Oconee application. We will inform the Committee of our conclusions on this matter and the related gaseous waste release limits and other potential restrictions in a subsequent report.

Based on our evaluation we have concluded that the Oconee radwaste systems are capable of providing waste effluents which can be considered to be "as low as practicable," and that appropriate technical specifications will be developed to ensure acceptable performance.

8.3 Spent Fuel Storage

The fuel handling system shown in Figure 9-11 of the FSAR is designed to transfer spent fuel to the fuel storage pools. One pool will be shared by Units 1 and 2. A separate independent, pool will be provided for Unit 3. Each pool is an integral part of a separate Class I (seismic) structure whose walls and roof as well as the pool itself are constructed of reinforced concrete. The pool is lined with 1/4-inch stainless steel. The only crane in the pool area is a 100 ton cask handling crane that will not travel directly over the spent fuel storage rack but will be limited by design to movement over an area located at one end of the pool reserved for spent fuel shipping casks. A smaller fuel handling bridge will be used to maneuver the individual fuel assemblies during fuel handling operations.

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

The cooling systems for the pools, shown in Figure 9-5 of the FSAR, are identical. Each system is provided with two circulation pumps and two heat exchangers. The failure of one pump and one heat exchanger with 1-2/3 cores in storage could result in an increase in the temperature of pool water to about 205° F over a long period of time. The loss of all pool cooling with 1-2/3 cores in storage could result in the attainment of this temperature in about 9 hours. Since each pool is well provided with diverse alarms (high temperature, low coolant flow, and low pool level) we conclude that the cooling system is adequate and that even in the event of a complete loss of cooling, adequate time will be available to restore cooling and replenish any water lost through evaporation prior to any significant fuel damage.

Except for the lack of adequate provisions to reduce radioactive releases to the environment in the event of a fuel handling accident, as discussed in Section 9.2 of this report, we conclude that the spent fuel storage provisions are acceptable.

8.4 Other Systems

8.4.1 Boric Acid Addition Systems

Boric acid can be injected into the primary system from either the letdown storage tank in the high pressure injection system or from the borated water storage tank and core flooding tanks in the emergency core cooling system. The boron concentrations do not require heating of these tanks and associated piping to prevent precipitation. All borated water is provided from the boric acid mix tank through associated pumps and piping. These components are provided with heater systems to prevent precipitation. All storage tanks containing boric acid are provided with sample lines to periodically monitor the concentration of boric acid. Other than those systems and components directly associated with the emergency core cooling system, which is still under evaluation, we have concluded that the boric acid addition systems are acceptable.

8.4.2 HP and LP Service Water Systems

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

There are two LPSW systems. One will be shared by Units 1 and 2 and the other, of identical capacity, will service Unit 3. Each LPSW system takes its water supply from the condenser circulating water system through three 15,000 gpm pumps. Two pumps are on one suction line, and the third pump is on another suction line. The HPSW system is also available as a backup source at the LPSW system pump discharge. All pumps and piping serving engineering safety features components are designed to Class I (seismic)standards. In addition, low pressure service water is provided to the redundant low pressure injection coolers and the reactor building coolers through separate supply lines. Two pumps are sufficient to provide all LPSW system performance requirements following a loss-of-coolant accident. The third pump provides protection against loss of a pump due to a single failure under accident conditions. All pumps are powered from the emergency power system.

We have concluded that the HPSW and LPSW systems will provide all needed normal and emergency services and are acceptable.

9.0 ACCIDENT ANALYSIS

9.1 General

We have evaluated the potential consequences of several design basis accidents. The calculated offsite doses for six of these accidents are given in Table 9.1-1.

TABLE 9.1-1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

	2-Hour Site Boundary Doses at 1 Mile (Rem)		Course of Accident LPZ Doses at 6 Miles (Rem)	
Accident	Thyroid	Whole Body	Thyroid	Whole Body
LOCA	190	2	200	1
Refueling	300*	<1	60	<1
Waste Gas Decay Tank Rupture	NA	2	NA	<1
SG Tube Rupture	<1	<1	<1	<1
Steam Line Break	21	<1	3.5	<1
Rod Ejection	≤ 5.4**	<2	12	<1

*We do not regard the 300 Rem as an acceptable value.

**The rod ejection dose will be equal to or less than 5.4 Rem at the site boundary, dependent on the assumed iodine reduction factor in the secondary system -- See Section 9.7.

Discussions of several of the design basis accidents and the assumptions that we have used in our evaluation are provided in the following sections of this report.

9.2 Loss-of-Coolant Accident

Our general review of this subject is still in process (see Section 6.1) and we will report our final conclusions to the Committee in a subsequent report. We have, however, analyzed the potential radiological dose consequences of the accident, for a given set of assumptions. The results of our analysis are listed in Table 9.1-1. Our analysis was based on the following assumptions.

(1) Reactor power level of 2568 MWt.

(2) 100% of the core noble gas inventory is released to the containment and is available for leakage to the environment.

(3) 50% of the core iodine inventory is released to the containment.

(4) 50% of the iodine plates out in the containment.

(5) 10% of the remaining iodine is in the organic form.

(6) Containment design leak rate is 0.25%/day.

(7) After the first day, containment leakage is reduced to 45% of design value.

(8) 50% of containment leakage is to the penetration room and treated prior to release by passage through HEPA and charcoal filters.

(9) 50% of containment leakage is released to the atmosphere unfiltered.

(10) Charcoal filter efficiency for iodine is 90%.

(11) Breathing rate is 0-8 hrs: 3.47×10^{-4} m^3/sec
8-24 hrs: 1.75×10^{-4} m^3/sec
1-30 days: 2.32×10^{-4} m^3/sec.

(12) Meteorology (2 hour period)
Pasquill Type "F" conditions
Windspeed - 1.0 m/sec
Terrain correction factor = 2.2
Building wake effect: C = 0.5
A = 2540 m^2
Ground level release

(13) Exclusion radius of 1 mile

(14) Low Population Zone of 6 miles

9.3 Refueling Accident

Our analysis of the refueling accident was based on the following assumptions:

(1) Reactor power level of 2568 MWt, for prior infinite operation

(2) Accident occurs after 72-hour decay time.

(3) 208 pins (1 fuel bundle) are damaged.

(4) Radial peaking factor is 1.68.

(5) 20% of noble gases released from the damaged fuel pins to the environment.

(6) 10% of the iodines released to the fuel pool water.

(7) Decontamination factor of 10 for the iodines in the water.

(8) Charcoal filter iodine removal efficiency is 0% (no filters now provided).

(9) Breathing rate of 3.47×10^{-4} m^3/sec.

(10) Duration of accident is 2 hours.

(11) Meteorology same as for LOCA.

The results of our analysis are listed in Table 9.1-1. The predicted 2-hour thyroid dose at the site boundary is 300 Rem. This prediction is based on the assumption that charcoal filters are not available in the fuel handling area. The applicant does not propose to install such filters. We have informed the applicant that the potential dose consequences for the refueling accident are unacceptable and that changes to the facility or additional information in support of less conservative assumptions are required. This item has not yet been resolved with the applicant and remains as an open issue.

Westinghouse has submitted a proprietary Topical Report, WCAP-7518-L "Radiological Consequences of a Fuel Handling Accident," dated June 1970, in support of less conservative assumptions for analysis of the refueling accident. The report includes discussion of test information developed by Westinghouse at their expense. We have just initiated our evaluation of the report and while no final conclusions can be made, it appears likely that we will accept the Westinghouse evidence to the extent that we will decrease the resultant consequences by a factor of 10 or more.

The use of this factor would reduce the resultant dose for Oconee to about the 30 Rem level without filters. The two together could reduce the dose to on the order of 3 Rem. While the reduction due to the phenomena addressed in the Westinghouse report will exist in the Oconee case, the applicant has not and apparently cannot, because of the proprietary nature of the evidence, provide information to support its use in our evaluation of the Oconee application. We have concluded that the applicant should be required to install filters or provide information on the Oconee record to warrant less restrictive assumptions.

9.4 Failure of Gaseous Radwaste Tank

Our analysis of the gaseous waste tank rupture accident was based on the following assumptions:

(1) Entire noble gas content of one primary coolant volume (11,830 ft^3) is in the decay tank prior to rupture.

(2) Primary coolant radioactivity concentration is as given by applicant in the FSAR.

(3) Entire contents of decay tank are released to the atmosphere in 2 hours.

(4) Average decay energy of fission products released is 0.7 MeV/disintegration.

(5) Meteorology same as for LOCA.

The results of our analysis are listed in Table 9.1-1. The doses are insignificant.

9.5 Steam Generator Tube Rupture Accident

Our analysis of the steam generator tube rupture accident was based on the following assumptions:

(1) Reactor operating with fission product inventory as given in Table 11-3 of the FSAR.

(2) Accident duration of 30 minutes.

(3) Volume of primary coolant lost is 1,980 ft^3.

(4) Primary system volume is 11,830 ft^3.

(5) All noble gases released to steam generator are exhausted to the environment.

(6) Iodine partition and plateout factor in condenser of 100.

(7) Breathing rate of 3.47×10^{-4} m^3/sec.

(8) Meteorology same as for LOCA.

The results of our analysis are listed in Table 9.1-1. The doses are insignificant.

9.6 Steam Line Break Accident

Our analysis of the steam line break accident considers two sources of iodine: the equilibrium iodine in the secondary system as limited by the loss-of-load event; and secondly, that activity carried over by the 10 gpm primary-to-secondary leakage during the course of the accident (3 hours). The assumptions used in our analysis included:

(1) Duration of accident is 3 hours.

(2) Total steam generator tube leak is 10 gpm.

(3) Secondary activity is released as for the loss-of-load event (see Section 9.8.4).

(4) Primary coolant source is as given by applicant in the FSAR.

(5) Steam generator tube leak rate is constant for the term of the accident.

(6) All iodines and noble gases that carry over to secondary side are released to the environment without plateout or partitioning.

(7) Breathing rate of 3.47×10^{-4} m^3/sec.

(8) Meteorology same as for LOCA.

(9) See Section 9.8.4 for assumptions used in deriving secondary system contribution.

To obtain 2-hour exclusion distance doses we assumed that 1200 gallons (10 gpm for 2 hours) of primary water was carried over. We also assumed that all of the iodine in the 1200 gallons was transported to the site boundary without plateout or partitioning. Further, we assumed that all of the secondary coolant system iodine (as determined by the loss-of-load event) was released in the same manner. We used the applicant's value of 14.1 Ci/m^3 of iodine activity as the specific concentration in the primary coolant and calculated 64 Ci of iodine as the primary source carried over in the 1200 gallons of primary coolant.

The dose due to the steam-line-break accident was estimated to be 6 Rem from the primary coolant and 15 Rem from the secondary coolant, for a total of 21 Rem. It should be noted that the Oconee steam lines, up to and including the main steam stop valves, are designed to Class I (seismic) standards. This is not true for most other PWR plants reviewed to date.

9.7 Rod Ejection Accident

We and the applicant have evaluated the rod ejection accident with markedly different but still acceptable results. Our analysis was based on the following assumptions:

(1) Reactor power level of 2568 MWt.

(2) 4.1% of fuel undergoes cladding damage.

(3) Prior reactor coolant source is as given by applicant in the FSAR.

(4) 20% of noble gases released from damaged fuel.

(5) 10% of iodines released from damaged fuel.

(6) Release path is from primary system to steam generator then to environment through air ejector or safety valves and assuming a prior 10 gpm steam generator tube leak.

(7) Partition factor of 10 used for release through steam-line safety valves; factor of 100 used for condenser.

(8) Primary coolant volume is 11,830 ft^3.

(9) Duration of release to environment is 2 hours.

(10) Breathing rate is 3.47×10^{-4} m^3/sec.

(11) Meteorology same as for LOCA.

(12) See Section 9.8.4 for assumptions used in deriving secondary system contribution.

The results of our analysis are listed in Table 9.1-1. The applicant's assumptions were much different and resulted in a 2-hour thyroid dose of 0.25 Rem at the site boundary.

9.8 Other Accidents

9.8 1 Sustained DNB

The applicant has referenced a B&W Topical Report, BAW-10014 "Analysis of Sustained Departure from Nucleate Boiling Operation," dated August 1969. We reviewed the report and requested additional information from the applicant. This was provided in Supplement 1 filed April 20, 1970. On the basis of the information now available, we concur that B&W can predict post-DNB surface temperatures with their correlation, although, in our opinion, the predictions may be low by 100° - 150° F. We do not share the applicant's confidence that the probability for fuel rod or fuel assembly loading errors is extremely improbable. We have concluded that initially at least, in-core flux maps should be made after every loading adjustment as an added precaution.

We agree with the applicant that sustained DNB should not be "contagious." We base our conclusion on the results of many rod bundle DNB tests, and on the fact that if cladding failure did occur, the system pressure would be higher than the rod plenum pressure, and the clad would probably collapse on the rod.

We will establish suitable in-core flux mapping requirements in the technical specifications.

9.8.2 Anticipated Transients Without Scram (ATWS)

We have recently transmitted to B&W a detailed list of conditions and assumptions on ATWS. We have concluded that Oconee Unit 1 should be licensed for operation without the need for the submittal of further information on this subject by the applicant at this time.

9.8.3 Loss-of-Flow and Cold-Water Events

The applicant filed Amendment 15 on July 9, 1970 containing additional information concerning the loss-of-flow and cold-water events. We will evaluate this material and report to the Committee in a subsequent report.

9.8.4 Loss-of-Load

The loss-of-load event will be our basis for establishing technical specification limits for primary and secondary radioactivity concentration limits for Oconee (as we did for Palisades and H. B. Robinson). We assumed that the secondary coolant at Oconee contained 777 Ci of iodine activity, and that this activity was released during a loss-of-load event. About 150,000 lbs of secondary coolant (about 92 m^3) was assumed to be released. The corresponding specific activity is about 8.4 uCi/cc for all the iodine isotopes, or about 1.9 uCi/cc for I-131. Assuming a factor of 10 reduction due to plateout or partitioning, as we did for the Palisades and Robinson analyses, results in a 2-hour site boundary dose to the thyroid of 1.5 Rem.

9.8.5 Startup Accidents

The applicant has submitted an analysis of startup accidents. The acceptance criteria were: (1) reactor thermal power should not exceed 114% of rated full power and (2) reactor coolant pressure should not exceed code allowable limits. Rod withdrawal from rated power was calculated to result in a power level of rated full power and a pressure of 2320 psia (less than the 2500 psia code allowable value). Parametric analyses were performed. No fuel damage was predicted in any of the cases analyzed.

We agree with the applicant's conclusion that no fuel damage will occur, if the reactor protection system performs as designed.

10.0 CONDUCT OF OPERATIONS

10.1 Technical Qualifications and Operating Organization

10.1.1 Staffing

The applicant has proposed a fairly conventional onsite staff organization, described in Section 12 of the FSAR and in Amendment 11. The minimum qualifications for key personnel in the operating organization are in general agreement with those proposed in the ANS proposed Standard for Selection and Training of Personnel for Nuclear Power Plants with the exception of experience requirements for the Technical Support and Performance Engineers, the Chemist, and the Maintenance Supervisor. We shall require that technical specifications delineating minimum qualifications for these positions be consistent with the ANS proposed standard. The applicant has agreed to include such qualifications.

10.1.2 Training

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

10.1.3 Shift Crew Size

The applicant proposes to operate the integrated facility with smaller shift crew sizes than we consider acceptable. We have informed the applicant that minimum crew sizes of five men will be required for Unit 1 operation and, based on our present guidelines, eight men per crew will be required for the combined operation of Units 1 and 2, and twelve men per crew will be required for the combined operation of all three units. We have also informed the applicant of the required composition of the crew with respect to the numbers and types of AEC operator licenses needed for each crew. We have indicated to the applicant that we are willing to consider smaller crew sizes after significant operating experience has been obtained. The applicant has agreed to our requirements for a five-man crew for operation of Unit 1. We will make this a technical specification requirement subject to reconsideration for possible reduction to a four-man crew after significant operation.

10.1.4 Review and Audit

The applicant has stated in Amendment 11, that provisions have been made for review and audit of plant operations by a group independent of the line organization. We will require that the essential requirements for review and audit by this group be included in the technical specifications, in a manner similar to that used for other recently licensed plants.

10.1.5 Operating Procedures

Written procedures covering overall plant operations, individual system operations, and abnormalities in operation will be prepared by the station operating staff prior to operation of any system. All procedures will be reviewed by the Station Review Committee and approved by the superintendent.

10.1.6 Summary on Organization

With the exception of the subject of shift crew size, discussed above, we have concluded that the proposed operating and technical support organization for operation of the Oconee facility is satisfactory.

10.2 Emergency Planning

The emergency plan is described in Section 12.3.2 of the FSAR. It provides for a broad spectrum of accidents that could affect both onsite personnel and the public in unrestricted areas. Offsite groups that may be required to participate have been identified and the applicant states that they are familar with the plan and have given assurance of cooperation as needed. Reliable means of communication have been identified. Provision has been made for medical support including treatment of radiation injuries and decontamination. Personnel will be trained in emergency procedures and periodic drills will be held to maintain competence. Provision is also made to include offsite agencies in simulated drills to the extent practical.

Available plant process, area, and vent radiation monitoring instruments will be used as the bases for determining the need for protective action. Additional emergency instruments and equipment will be available.

Detailed procedures for the implementation of the emergency plan will be prepared by the staff of the Oconee Nuclear Station. The plan will be updated as required and periodically reviewed by the General Office Staff of the Steam Production Department.

The need for recovery plans has been recognized but the general measures for recovery and re-entry have not been described. We believe the applicant can establish and we will require him to describe acceptable general recovery and re-entry measures prior to issuance of the operating license for Unit 1.

10.3 Industrial Security

Provisions for industrial security described by the applicant in Amendment 11 include perimeter fencing, gate and door access control, and a closed-circuit television and remote control lock system for off-hour identification and admission of personnel to the facility. Appropriate controls over access to Units 1 and 2 by construction personnel working on the units still under construction have been developed. We have concluded that the provisions for industrial security are adequate.

10.4 Preoperational Testing and Startup Organization

The applicant has proposed an overall testing program consisting of the following phases: (1) preheatup tests, (2) hot functional tests, (3) initial fuel loading, (4) initial criticality, (5) zero power tests, and (6) power escalation tests. Summaries of the 28 test procedures directly affecting plant safety have been provided for our review.

We are currently reviewing these summaries in conjunction with our review of the technical specifications. Also, as is usual, the Division of Compliance will verify the adequacy of the detailed procedures and test results.

We have requested additional information concerning the startup organization which is required to complete our evaluation of this phase of operation. This information will be reviewed for adequacy prior to issuance of an operating license for Unit 1.

10.5 Conclusions

We have concluded that the area of conduct of operations is generally acceptable and that the additional detailed information we require will be provided in a manner to permit issuance of an operating license for Unit 1.

11.0 QUALITY ASSURANCE

Although the Oconee Nuclear Station was approved for construction prior to formulation of Quality Assurance Criteria, as published in the Federal Register on April 17, 1969, we have used these criteria as a guide in reviewing the applicant's quality assurance measures.

As a result of our review, the applicant filed Appendix 1B of the FSAR to provide a summary description of the QA program. As noted therein, the applicant delegated design and fabrication QA responsibilities to B&W for the nuclear steam supply system, and to Bechtel for the design of the reactor containment building.

We have conducted QA audits of both B&W and Bechtel in the past in connection with quality assurance efforts on other applications and have found both organizations to be generally adequate.

The applicant's QA organization does not include a group that has quality assurance program responsibility without project schedule and cost-related responsibilities. However, the applicant has provided for independent checks to be made on design, on construction work performed at the site, and on the manufacture and testing of major components in vendor shops.

With regard to quality assurance matters at the site, the Principal Field Engineer and his staff are fully responsible for quality control. He performs no engineering work on the safety-related structures, equipment, and systems which are identified in Table 1B-1 in Appendix 1B of the FSAR. Further, neither he nor his staff perform any work for the job superintendent, but report directly to the job superintendent's superior, the Oconee Project Engineer.

With respect to maintaining quality assurance following construction of this plant, we will require the applicant to have any design changes reviewed and approved by the original design organization unless there is good cause for specifically designating a different organization. Design and design changes will also be addressed in the technical specifications.

In addition to the above, the AEC's Division of Compliance has performed detailed inspections of work in progress both at the reactor site and at vendor shops. As would be expected, those inspections have revealed a number of deficiencies. However, proper action has been taken in each instance to correct these deficiencies as they arose.

We have concluded that the applicant's quality assurance program, as described in the FSAR will assure an acceptable and adequate level of quality of the safety-related systems, equipment, and structures incorporated in Oconee Station Units 1, 2, and 3.

12.0 TECHNICAL SPECIFICATIONS

We are currently discussing technical specifications with the applicant. The technical content when revised will be quite similar to that developed for the Palisades and H. B. Robinson Unit 2 plants. Several of the matters under discussion have been indicated in previous sections of this report. Another matter under discussion and that is not presently included in the proposed technical specifications is the initial power level. We will require that operation be limited to 2452 MWt, the level approved at the construction permit stage, until submittal and approval by us of an operational performance evaluation report verifying performance in full conformance with design expectations. Only then will we permit operation at the presently requested level of 2568 MWt.

We expect to complete our effort on technical specifications within the next several weeks and will summarize the results of this effort in a subsequent report.

13.0 ACRS MATTERS

The Committee has recommended that our review at the operating license stage include 16 items which have come to be known as the "a-p" items, plus five additional PWR plant items. All 21 items have been considered during our review of the Oconee Nuclear Station and have been discussed in various sections of this report or will be covered in the next report. Table 13.1 provides a cross reference to indicate the sections of this report where discussions of specific items are presented.

In addition, in our review we have referred to the various recommendations and comments made by the Committee in its previous letter on the Oconee construction permit review and conclude that all these matters have been covered in this review except for matters relating to the Emergency Core Cooling System which will be covered in the next report.

TABLE 13.0-1

CROSS REFERENCE - ACRS "A-P" AND "1-5" ITEMS

a. Periodic inspection of primary system and containment - 4.6, 5.6.

b. Reactivity and power distribution anomalies - 3.2.

c. Leak detection criteria and response to identified leaks - 4.7.

d. Safety system adherence to current criteria - 7.2, 7.3, 7.4, 7.5.

e. Acceptability of shared safety and control system components - no sharing.

f. Safety system testing frequency - will be in Technical Specifications.

g. Adequacy of preoperational tests of all vital systems and functions - 10.4.

h. Pressure vessel surveillance program - 3.4.1.

i. Adequacy of onsite and offsite emergency radiation instrumentation - 10.2.

j. Adequacy and independence of operator safety review - 10.1.4.

k. Adequacy of procedures and equipment for fire protection - 8.4.2.

l. Routine release of radioactivity to the environment - 8.2.

m. Loss of all offsite power - not significant due to unique condenser cooling capability (Section 1.3).

n. Adequacy of operational procedures that affect public safety - 10.1.5.

o. Adequacy of operational staff - 10.1.3.

p. Identification of items requiring individual review after licensing - incomplete.

1. Instrumentation to diagnose the course of a serious accident - incomplete.

2. Applicant's plans for malfunction analysis, alarms, and operator instructions with regard to symptoms of abnormal operation having potential safety significance - incomplete.

3. Provisions during maintenance and repair to avoid changes which could result in common failures in systems important to safety - 10.1.4, 11.0.

4. Provision for use of methods for inservice monitoring of vibration or mechanical damage in important components and implementation of such methods as they offer promise of successful operation - 3.4.2.

5. Criteria for judging the acceptability of replacement cores not fabricated by the original vendor - criteria not yet developed.