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

September 15, 1970

Report No. 3 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 completes the presentation of the results of our evaluation of all review areas with the exception of the applicant's multi-node ECCS analysis, the final Technical Specifications, and the provision of a diverse reactor trip for use with the ECCS. There are no new major areas of review covered in this report. The need for a post-LOCA hydrogen control system is still being evaluated as a potential backfit problem, but we have concluded that resolution is not required prior to licensing of Oconee Unit 1.

Pending satisfactory completion of the ECCS analysis, the provision of an acceptable diverse reactor trip for use with the ECCS and completion of the Technical Specifications, we have concluded that Oconee Nuclear Station Units 1, 2, and 3 can be operated without undue risk to the health and safety of the public.

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1.0 INTRODUCTION AND SUMMARY

This is our third report to the Committee on the Duke Power Company application for operating licenses for their Oconee Nuclear Station. Since issuance of the last report, the applicant has submitted Amendment No. 19, dated September 8, 1970, and Amendment No. 20, dated September 14, 1970. The information provided in these amendments is responsive to most of the outstanding concerns that were discussed at the ACRS Subcommittee meeting September 9, 1970. In addition to completing our review of the Technical Specifications, the review areas that remain outstanding and affect the licensing of Oconee Unit 1 are the results of the applicant's multi-node ECCS analysis and the provision of an acceptable diverse reactor trip for use with the ECCS. These areas are discussed in Sections 9.1 and 7.2 of this report. We are continuing to review the need for a post-LOCA hydrogen control system as a potential backfit item. This is discussed in Section 6.2 of this report.

For continuity, we have retained the same major section numbers in this report as were used in the prior two reports. For brevity, however, this report contains only those major sections for which our review was not completed in an earlier report. The subsection headings were chosen to reflect the nature of specific concerns identified in prior reports.

Pending satisfactory completion of our review of the Technical Specifications, the applicant's multi-node ECCS analysis and the provision of a suitable diverse reactor trip for use with the ECCS, we have concluded that the Oconee Nuclear Station, Units 1, 2, and 3 can be operated without undue risk to the health and safety of the public.

3.0 REACTOR DESIGN

3.1 Fuel Design

As noted in Section 3.3 of our September 2, 1970 report to the ACRS, we requested additional information from the applicant with respect to the use of prepressurized fuel in the Oconee reactors and the development of a suitable fuel rod surveillance program.

Additional information was submitted in Amendment Nos. 18 and 20 in response to our request. The applicant now proposes to select the following seven assemblies for inclusion in the program: (1) the center fuel assembly which will be removed during the first refueling outage, (2) four assemblies which will be removed during the second refueling outage (two with pressurized pins, two with unpressurized pins), and (3) two pressurized assemblies which will be removed during the third refueling outage. Prior to operation, base-line pin length and diameter measurements will be made on all seven of these assemblies. Prior to operation, eight of the unpressurized pins on the periphery of the center fuel assembly (two in each outer row) will be pressurized. After these assemblies are removed from the core, at the times indicated above, the applicant will again take measurements to determine the extent of pin dimensional changes (diameter and length) due to irradiation.

On the basis of our review, we have concluded that the use of prepressurized fuel is acceptable and that the applicant will institute a suitable fuel surveillance program encompassing both pressurized and unpressurized fuel pins. The surveillance program will be included in the Technical Specifications.

3.2 Reactor Internals Vibration Monitoring Program

We noted in our September 2, 1970 report to the ACRS that the applicant's reactor internals vibration monitoring program, as proposed in Amendment No. 17, did not provide for obtaining information on several key internal components. As a result of additional discussions wherein we

further explained our concerns, the applicant stated his intent, in Amendment No. 19, to provide three additional accelerometers to be attached to the plenum cylinder above the core for the purpose of measuring the bell-mode vibrations of the plenum cylinder. The applicant has further agreed, in Amendment No. 20, to add an additional accelerometer to the upper core support barrel approximately 2 feet below the check valve assembly at the location of one of the reactor inlet nozzles, to measure the response of the upper core support barrel, and to monitor the effects of the combined forces of fluid flow and inlet-outlet pressure variation. The applicant has agreed to establish in the startup test procedures a "go, no-go" limit for this monitor on the upper core support barrel together with a supporting technical basis that one accelerometer will determine all response characteristics. In the event that this limit is exceeded, the applicant has agreed to perform additional analyses or vibration tests of this critical internal component.

As an aid in interpreting the results obtained from the vibration monitors installed on the Unit 1 reactor internals, B&W expects to utilize data obtained by shaker-table testing of a later production set of reactor internals which are identical to the Oconee Unit 1 internals. The data derived from this test will be available in October or November 1970 and will establish the base line for interpreting the Oconee hot functional test data.

We have concluded that the applicant's preoperational vibration monitoring program, expanded as discussed above, is acceptable.

4.0 REACTOR COOLANT SYSTEM

4.1 Stress Analysis of Unit 1 Reactor Coolant Loop

The applicant has presented, in Amendment 20, the results of his review of the stress analysis of the Oconee Unit 1 reactor coolant system. These results confirm that the modified system configuration, with the replacement pumps, is in compliance with the USAS B31.7 Code for Nuclear Power Piping with the new configuration using the replacement pumps. Based on this

additional information we have now concluded that the pump replacement modification is acceptable. We will, of course, continue our review of the stainless steel clad cracking that was discovered in sections of the reactor coolant piping during the replacement operations and assure ourselves that adequate corrective action is taken.

4.2 Reactor Pressure Vessel Material Surveillance Program

As noted in Section 4.2.1 of our July 24, 1970 report to the ACRS we found the reactor pressure vessel material surveillance program acceptable except that we would require separate scheduled withdrawal programs (four capsules) for each of the three reactor vessels at Oconee. The applicant has agreed to this requirement and the actual time schedule for Oconee Unit 1 will be provided in the Technical Specifications.

On this basis, we have concluded that the applicant's reactor pressure vessel material surveillance program is acceptable.

4.3 Steam Generator Pipe Whip Protection

As noted in Section 4.2.2 of our September 2, 1970 report to the ACRS, we were not satisfied that the applicant had clearly demonstrated that a reactor coolant pipe break would not rupture the shell of the steam generator. As a result of our concern, the applicant performed additional analyses of breaks in both the hot-leg and cold-leg reactor coolant piping within the steam generator compartment and concluded that, for certain types of breaks, additional pipe restraint capability should be provided to limit the energy in the broken pipe and to distribute the resulting forces over a larger surface area on the steam generator shell. These restraints will be installed prior to operation in the locations shown on Figures 12-1 and 12-2 of FSAR Supplement 8.

We have reviewed the applicant's analyses and the proposed design revisions and we have concluded that, with installation of the additional restraints, there is reasonable assurance that a reactor coolant pipe break will not rupture the shell of the steam generator.

4.4 Steam Generator Tube Vibration Verification

As discussed in Section 4.2.4 of our September 2, 1970 report to the ACRS, we requested the applicant to establish a program to verify that the tubes in the fully assembled steam generator have no significant vibratory motion.

Based on discussions with the applicant, this has proven to be difficult to accomplish. In an effort to satisfy this concern, he has agreed to perform a tube "pluck" test on a steam generator now being assembled at the B&W factory to determine natural frequency characteristics of tubes in a production unit. He is also seeking to find a feasible means of monitoring a steam generator under actual full-load operating conditions. Although we will continue to pursue this matter on the B&W steam generator, we have concluded that the applicant's commitment on verification of steam generator tube vibration is acceptable for the licensing of Oconee Unit 1.

5.0 CONTAINMENT AND CLASS I STRUCTURES

5.1 Effect of Secondary System Blowdown on Containment Pressure During LOCA

The Oconee containment design pressure is 59 psig. The applicant calculates that a peak pressure of 53.9 psig would result from the blowdown of the reactor primary coolant system alone. Our independent analyses of the peak containment pressure following a break in the reactor coolant system give results that are in close agreement with the applicant's calculation. Based upon the above numbers, there is a 9.5% pressure margin between design pressure and the calculated peak pressure.

Since January 1969, we have required at the construction permit stage, that all PWR plants with "dry" containments have a 10% pressure margin on the containment design. Although Oconee's construction permit predates January 1969, the pressure margin provided in the containment design is just under the currently required 10% margin.

The review model established in January 1969, included guidance on acceptable methods for calculating containment pressure transients. Containment pressure transients were to be calculated assuming either the reactor coolant system or the secondary system ruptured. In support of this analytical model, the reactor system would have to be designed to prevent pipe whips and/or jet forces that could cause one coolant system to damage another.

The Oconee applicant has analyzed the effects of the combined break in the primary coolant system and a simultaneous break in the feedwater ring. Based upon minimum containment heat removal systems turning on at 25 seconds and the use of the Tagami condensing heat transfer correlation, the applicant calculates a peak containment pressure of 59.3 psig, occurring at 51 seconds. Although this peak pressure due to the combined primary and secondary blowdown slightly exceeds the design pressure, we have concluded that it is acceptable. In arriving at this conclusion we have considered the magnitude of the calculated overpressure (0.3 psi) and the fact that the reactor coolant piping layout and restraints act to minimize the kinds and locations of reactor coolant pipe breaks that could cause loss of the feedwater ring.

In addition to the effect of steam generator blowdown on containment pressure, we also evaluated its potential for diluting the borated water used to cool the core during the recirculation phase. We have concluded that the addition of the secondary inventory from one steam generator to the reactor coolant inventory, as supplemented by the inventory in the ECCS tanks (core flooding tanks and borated water storage tank), would increase the inventory of recirculating water by about 6%. This would result in dilution of the boron concentration from 2200 ppm to 2070 ppm. A conservative calculation indicates that the diluted water would have at least 500 ppm more boron than would be required to maintain the cold core subcritical (even with all control rods removed).

On the basis of the above evaluation we conclude that the dilution effects of a steam generator blowdown coincident with a loss-of-coolant accident are acceptable.

6.0 ENGINEERED SAFETY FEATURE

6.1 Availability of Core Flooding Tank Inventory

Following a loss-of-coolant accident, until emergency power is available to provide pumped injection from the borated water storage tank, the Oconee Units have two 600 psig core flooding tanks. Both tanks are required for injection of borated water directly into the core through 14-inch core flooding lines attached to nozzles on the pressure vessel. The Oconee core flooding tank system differs from systems used in some PWRs where a single tank out of service may be tolerated.

In Section 6.1 of our September 2, 1970 report to the ACRS we noted that the motor-operated valve in the line connecting each core flooding tank to the reactor vessel must remain in the fully open position during the initial stages of a loss-of-coolant accident. We also noted that the applicant had been requested to provide additional means to assure that this valve will be open whenever the reactor is in operation. The applicant has now agreed to provide these means. As noted in Amendment 20, the applicant will modify the design to provide (1) two independent means of determining valve position, (2) an alarm in the control room if the valve is not fully open, and (3) lock out of the power to the valve during normal operation without causing loss of the two means of indicating valve position or the alarm. With these additions, we have concluded that acceptable provisions have been made to assure that these valves will be open when required.

6.2 Post-LOCA Hydrogen Control

We discussed our proposed position on post-LOCA hydrogen control with the Commission on September 4, 1970. In accord with the Commission's

comments, we intend to begin to implement the position we have previously discussed with the ACRS. This position will be taken on all future construction permit applications and on all plants with construction permits which were issued after mid-1969. Since this safety issue was identified as an unresolved matter on all construction permit applications we have processed since mid-1969, implementation of this position will not be considered a backfit item on these plants. We will review the need for backfitting all other plants on a case-by-case basis. The provision of a post-LOCA hydrogen control system in the Oconee Unit 1 facility is a backfit consideration.

The applicant presently intends to purge the containment to control the accumulation of hydrogen. The assessment of the hydrogen problem for the Oconee plant is unique because of the fact that: (1) although it has a large "dry" containment, the plant has no special provisions for reducing the inventory of iodine in the containment atmosphere after a LOCA, and (2) the boric acid solution to be used in the containment sprays will attack the approximately 5800 pounds of zinc within the containment with the evolution of significant amounts of hydrogen (the amount evolved within a 26-day period is estimated to be equivalent to the amount evolved by a metal-water reaction involving about 10% of the fuel cladding).

The fact that no iodine reduction features were provided was overlooked when we initially estimated the consequences of purging the Oconee containment. We informed the Committee, in our July 24, 1970 report, that a thyroid dose of 30 Rem could result at the site boundary (1 mile) if the containment were purged to control hydrogen following a LOCA. This estimate was in error. We now estimate a thyroid dose approaching the guideline value of 10 CFR 100, at the site boundary. Both our former and current estimates were obtained by the use of an approximate method of analysis. The results could be affected in a significant manner by a more exact evaluation.

We intend to study the hydrogen evolution and control problems for Oconee in greater depth. We will recommend to the Commission that the Oconee plant backfit if we find "that such action will provide substantial, additional protection which is required for the public health and safety or the common defense and security" in accordance with Section 50.109 of 10 CFR 50.

We intend to complete our review of this problem as expeditiously as is practical, but we have concluded that licensing of Oconee Unit 1 for operation should proceed without further resolution at this time.

7.0 INSTRUMENTATION, CONTROL, AND POWER SYSTEMS

7.1 Axial Power Imbalance Protection

In Amendments 18, 19, and 20 the applicant has provided information describing a modification to the reactor protection system to prevent possible core damage due to axial power imbalance between the upper and lower core regions. We have reviewed this information and discussed in some detail the manner in which the reactor protection system instrumentation has been modified to accomplish this protection. Basically, the applicant has taken the signals from the bottom and top sections of the power range flux detectors through a difference amplifier to get an axial imbalance signal which is then sent to a function generator programmed to activate a trip signal if the imbalance exceeds a predetermined amount in either direction (a higher flux at top or bottom). To accomplish this, two existing instrumentation modules (Power Range Test Module and Function Generator) have been modified and a new module (Sum/Difference Amplifier) has been added.

On the basis of our review, we have concluded that the modules involved have been designed, tested, and installed consistent with the requirements of IEEE-279, and are acceptable.

7.2 Diverse Reactor Trip with ECCS

By reference in Amendment No. 19 the applicant has incorporated B&W Topical Report 10019 - "Systematic Failure Study of Reactor Protection Systems" dated September 4, 1970. In Section 4.6.3 of that report B&W describes the capabilities of the power-to-flow reactor trip as a diverse backup for the low reactor pressure trip for loss-of-coolant accidents. Based on an examination of that description we have concluded that for some conditions, substantial time is required to achieve a

power-to-flow trip limit; for other conditions it is implied that no power-to-flow trip may occur. Since the flux sensors are not designed to continue to operate during LOCA temperature conditions we cannot conclude that the power-to-flow reactor trip is available for all LOCA conditions.

We have therefore informed the applicant that we plan to require another means of obtaining the required diverse reactor trip. There should be no problem in obtaining such a trip since there are at least two trip conditions available; low reactor flow and high containment pressure. We will require provision for a diverse reactor trip for use with the ECCS prior to licensing Oconee Unit 1.

7.3 Three-Pump Operation

In response to our concern that there be backup protection for three-pump operation, the applicant has presented information in Amendment No. 20 which shows that there is adequate backup protection to prevent core damage in the event of a failure of the power-to-flow protection. Further, as discussed in Amendment No. 18 (on page 7-5 of Supplement 7), loss of one of four pumps while running at full rated power without reactor trip or power runback would result in an acceptable minimum DNBR of 1.34. On the basis of our review of this additional information, we have concluded that the protection instrumentation for three-pump operation is acceptable.

9.0 ACCIDENT ANALYSES

9.1 Emergency Core Cooling System

It is our intent to license Oconee Unit 1 for operation only after we have been provided with calculational results that demonstrate the adequacy of the installed emergency core cooling system to limit peak clad temperatures to acceptable levels in the event of a loss-of-coolant accident. The applicant was formally requested to submit additional information on this matter by a letter dated July 15, 1970, although our concerns were made known to the applicant orally several weeks prior to our letter.

At the time we requested the additional information from the applicant we were aware that the FLASH 2.5 Code would be used by B&W to perform the necessary analyses. We were also aware, however, that while the FLASH 2.5 program had been under development by B&W for several years, it was still not completely operational. We recognized that the information that would be submitted by the applicant might be limited and not sufficient as a basis for concluding that the Oconee Unit 1 emergency core cooling system was acceptable. We concluded, if the limited information we expected to be submitted by the applicant was supported by acceptable results obtained by another multi-node analysis, independent of the B&W analysis, then a suitable basis for licensing would have been provided. We arranged for the Idaho Nuclear Corporation (INC) to perform the independent analyses. These have been completed and the results would, in our opinion, confirm acceptable results obtained by the applicant with the FLASH 2.5 program, even though his results were minimal.

The applicant informed both us and the ACRS, at the Committee's August meeting, of preliminary results obtained with the FLASH 2.5 program. These results indicated that the peak clad temperatures would be within acceptable limits. Further information was provided to us verbally by B&W, at a September 1, 1970 meeting arranged for the purpose of informing the applicant of the results obtained by INC. This information also indicated that the results being obtained by B&W were satisfactory. The applicant informed us at that time that the information we had requested in our July 15, 1970 letter would be submitted to us formally on September 9, 1970, the day of the last ACRS Subcommittee meeting on the Oconee application. We were informed by the applicant on September 8, 1970 that B&W had performed some additional analyses and had become concerned about their ability to explain the reasons for the results, which appeared to be in conflict with results obtained previously.

The applicant informed us that B&W was not prepared to submit the requested information to us, as had been expected. We have not been able to obtain a clear understanding of the problems encountered by B&W either from the discussions held during the September 9, 1970 ACRS Subcommittee meeting or during a separate topical meeting held that same day with B&W and attended by Dr. Isbin and Mr. Allemann (ACRS consultant). The applicant stated that the information requested by the staff could not be provided by September 17, 1970, the date of the scheduled meeting of the full Committee for the review of the Oconee application.

The applicant submitted Amendment No. 20 on September 14, 1970. Unexpectedly the ECCS problem was addressed in the submittal. The applicant stated that B&W had completed initial calculations with a modified FLASH 2 computer code and that the results indicated that a maximum clad temperature of 2260° F would be reached in a loss-of-coolant accident. The applicant stated further that a proprietary topical report, describing the analyses, the assumptions, and the results, would be filed with the Commission prior to fuel loading for Oconee Unit 1.

We recognize the difficulty faced by the applicant and B&W in responding in a suitable manner to our late requests for multi-node analyses to demonstrate the adequacy of the ECCS. However, in our opinion the information provided to us is not sufficient for our present needs. The specific requests for additional information made of the applicant in our July 15, 1970 letter were developed on the premise that the information provided in response to the requests would be the minimum acceptable, when supported by acceptable results from an independent source, for the licensing of Oconee Unit 1. It is still our opinion that the requested information must be provided in order to reach a favorable conclusion with respect to the adequacy of the ECCS.

We have concluded that we will not publish a notice of intent to issue an operating license for Oconee Unit 1 until sufficient information is documented to resolve the question of ECCS acceptability to our satisfaction and to the satisfaction of the Committee.

10.0 CONDUCT OF OPERATIONS

10.1 Emergency Plan Recovery and Reentry Measures

On the basis of the additional information provided in Amendment No. 20, we have concluded that the applicant has described acceptable general plans for recovery and reentry measures following a major accident.