UNITED STATES
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Richard C. DeYoung, Assistant Director for PWRs, DRL
THRU: Charles G. Long, Chief, PWR-2, DRL

NOTES ON MEETING WITH DUKE POWER COMPANY ON TECHNICAL SPECIFICATIONS FOR OCONEE UNIT 1, DOCKET NO. 50-269

The meeting was held at Bethesda, January 28, 1971. An attendance list is enclosed.

Our objective of finalizing the Technical Specifications was not ealized. A discussion of outstanding areas is enclosed.

Major unresolved areas are (1) the staff bases for pressurization, heatup and cooldown limitations (fracture toughness concerns) (2) reactor coolant leakage specification format, (3) auxiliary power degredations and bases, (4) restriction on cranes and hoists during refueling, (5) radioactive waste disposal. Other areas requiring resolution are noted in the enclosed discussion.

Duke indicated that plant completion has slipped so that they will not be ready for fuel loading until late May at the earliest with a more likely date being sometime in June. Duke is not aware of any intervenors other than the North Carolina Municipalities on antitrust matters (which is not expected to require a prelicensing hearing).

A. Schwencer
PWR Project Branch No. 2
Division of Reactor Licensing

Enclosures:
1. Discussion
2. Attendance List

DISTRIBUTION:
Docket files
DRL Reading
PWR-2 Reading
MMMann
PAMorris
FSchroeder
TRWilson
RCDeYoung
EGCase, DRS
Compliance (2)
MSHildreth, CO, Hq.
CMUpright, CO Region II
RKlecker
NThomasson
RLBursey
DRRoth
DRL Branch Chiefs
DRS Branch Chiefs

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DISCUSSION OF OUTSTANDING AREAS OF THE OCONEE

UNIT 1 TECHNICAL SPECIFICATIONS

GENERAL

We used marked-up copies of the Oconee Tech Specs submitted as Amendment 24 dated December 14, 1970 as a basis for discussion. (Page references below refer to that document.)

Rated Power Definition (p 15-2) - Resolved. Duke agrees it is 2568 MWt core output, not "system" output. We agreed that there should be no need to state in the definition that it is contingent upon "all four coolant pumps operating."

Containment Integrity (p 15-3) - Resolved. Specification 15.1.2 A and B will be revised to make it clear that both doors of the personnel and emergency hatches remain closed except during refueling operations or personnel passage through these hatches.

Heat Balance - Resolved. Duke agreed that it is core thermal power that is to be compared to neutron power. We understand that certain computations are required to obtain this thermal power, particularly since a secondary heat balance will be employed.

Single Loop Operation - (p 15-21) - Resolved. Duke agreed to proposed revisions which included (1) adding a statement that single loop is authorized for testing only (2) timely notification of tests and a report to DRL evaluating tests (3) written approval for subsequent single loop operation and stating that trip points shall be set "to no higher than" rather than "at" 50% of rated power and 610° F reactor outlet temperature. Also, as pointed out by Dr. Mann, this Technical Specification (15.2.4.1 and two others (15.2.4.2 and 15.4.3) belong under limiting conditions of operation. Duke will relocate them accordingly.

Reactor Coolant System Activity (p 15-24) - Resolved. Requirements were relaxed to permit activity to be 293 μCi/ml or 312/$\overline{E}$ μCi/ml Mev whichever is more limiting. The basis were expanded to show how X/Q value is obtained.

Pressurization, Heatup and Cooldown Limitations (p 15-28) - Unresolved It was determined that we, not the applicant, would provide the basis for the increased conservatism which we are imposing. We will not "marry" the points of view but superimpose our requirement (DRS has been asked to provide the basis input needed here).

Duke has agreed for Specification 15.3.1.3.D2 to use an "integrated" period of an additional 1.7 x 10^6 thermal megawatt days of operation in updating Figure 15.3.1.1-1. We also requested that they give proper consideration to use of capsule specimen test results in establishing actual vessel exposure.

Reactor Leakage (p 10-36) - Unresolved - Ray Klecker and Dr. Mann requested Duke to use the Point Beach Technical Specifications to extent possible. We want to standardize. Duke will examine Point Beach Tech Spec for this purpose. We also said we wouldn't accept the 2.7 gpm as acceptable for offsite Part 20 doses.

Moderator Temperature Coefficient (p 15-42) - We requested Duke to simply state that the max. moderator coefficient at full power shall not exceed +0.9 x 10^{-4} $\Delta k/k/°F$. Any discussion on how this is calculated should be put in the bases. There should be no exceptions for which this value could be exceeded.

Engineered Safety Features (p 13-47) - Unresolved - We noted their use of the term "safeguards" which has special meaning applicable to other than nuclear power plant accidents. We requested use of the term "safety features" instead in the tech specs.

We also noted that only single instruments are associated with the BWST level, the core flooding tank pressure, and the reactor building emergency sump levels. Duke says they have added a second pressure channel to the CFT and are considering the feasibility of adding a second BWST level instrument (control room or local). We agreed that a second emergency sump level indicator is not required. Either Duke will add a second BWST level instrument or the single instrument must be operational in order for the reactor to be critical except for a brief maintenance period on the instrument.

Operational Safety Instrumentation (p 15-53) - Unresolved - Specification B - Duke agreed to only one channel bypass key in control room but now contrary to prior commitment during FSAR review, Duke wishes to relax the restrictions placed on the use of dummy bistable trip units. There are 4 reactor protection channels each of which has 8 bistable trip units, 1-power imbalance-flow comparator trip, 2-power-pump-comparator trip, 3-low-pressure trip, 4-pressure-temperature comparator trip, 5-high temperature trip, 6-high pressure trip, 7-high-power (neutron flux) trip, and 8-shutdown bypass high pressure trip.

During the FSAR review Duke was given the choice of (1) indicate by light on the control console the specific protective channel bistable trip unit that would be eliminated by use of the dummy unit (2) indicate by light on the control console just the protective channel that has a dummy bistable unit in place of one of its bistable trip units and treat the entire protective channel as bypassed in determining minimum instrumentation requirements, or (3) eliminate the dummy bistables. It was pointed out by Dr. Mann that the bases for justifying the dummy bistable trip units has not been given.

As a minimum, this basis must be added. In addition, further review will be required if we relax our position on the use of dummy bistables.

With regard to failure of a control rod drive trip device the point was made that it is important both to eliminate a condition where one more failure can prevent a reactor scram and to verify that the remaining trip devices are still able to trip upon demand. Duke does keep spare breakers available for this system but at the meeting was unwilling to commit ability to take all required corrective action within 30 minutes as we proposed. We await Duke reply on this.

Reactor Building (p 15-69) - Resolved. Duke agrees to administratively verify that manual containment isolation valves are closed prior to returning to critical after a refueling shutdown.

Auxiliary Power (p 15-70 - Unresolved - Duke still wishes to have prior approval to degrade to complete loss of the Hydro Station substituting the Lee gas turbine during such an interval. Our concern is that, should a distribution system outage cause loss of the power grid, there would be only one source of power for Unit 1 shutdown loads. This unique situation will exist until Unit 2 becomes available.

The November 13, 1970 basis for this specification (15.3.7) needs to be revised to account for the final version of this Technical Specification. The staff will write these bases since we do not intend to permit system degredation to the extent originally requested by Duke.

Fuel Loading and Refueling (p 15-74) - Unresolved - Duke wishes to ignore mandatory controls on the movement of the polar crane during fuel handling operations. Because of the demonstrated accident potential associated with cranes and hoists we cannot overlook this matter. As a minimum, there must be a clear requirement which prohibits any polar crane movement over the open core and refueling canal when a fuel assembly is attached to or being transported by either of the fuel handling bridges located inside the reactor building. If possible, this should be accomplished by physical interlocks, otherwise administrative control in the form of direct supervision should be employed.

Radioactive Waste Disposal (p 15-76) - Unresolved - The project leader was requested to seek full compliance in this Specification to the new release requirements covered by Amendments to 10 CFR 20 and 10 CFR 50 which were published December 3, 1970 in the Federal Register (Volume 35, No. 235 on pages 18385 through 18388) to be effective January 2, 1971.

The applicant felt that he was in "substantial" agreement with this new requirement and expressed the opinion that the present specification wording was largely that supplied earlier by the staff. The one apparent major Duke objection to full compliance with the new requirements published December 3, 1970 is the need to pass all reactor building purge through the installed HEPA and charcoal filter system even if there were no "measureable" activity being exhausted. They contend this will unnecessarily use up these expensive filters. They also objected to increasing gaseous waste holdup for 20 days rather than 10 days as proposed. We pointed out they have in excess of a 60 day holdup capacity and therefore could not understand the basis for this objection.

They also objected to placing on the record the estimated quantities of each of the principal radio-nuclides expected to be released annually in the liquid and gaseous effluents, contending that it could be derived from the material already in the FSAR (in our judgement, this is true only if we assume 1% failed fuel is expected). Also, through discussion, it was established that the low activity waste tank and condensate test tank activity in Specification 15.3.9.B.9 is meaningless in that the restricted area dose level specification 15.3.9.B.5 would be exceeded before reaching the tank activity levels permitted. Therefore we intend to delete 15.3.9.B.9.

Table 15.4.1-1 (p 15-88) - Unresolved - Duke was requested to conform with other PWR Tech Specs on frequency of instrument surveillance. We did not accept the unproved premise that B&W instrumentation is better than that provided by other vendors and thus warrants relaxed surveillance. This is a matter to consider after experience is obtained.

On reactor heat balance the procedures for obtaining this heat balance have not been written. We suggested that, as a minimum, there must be a daily comparison of neutron flux measurements and heat balance measurements. We do not intend that instruments be adjusted daily to maximize agreement between heat balance data. However, criteria for "drift" should be given as the basis of recalibration. Duke was in general agreement with this approach.

<u>Table 15.4.1-2 (p 15-89)</u> - Unresolved - We recommended addition of appropriate tests on (1) fire pumps and power supplies, (2) containment isolation trip, (3) service water system, and (4) spent fuel cooling systems.

Table 15.4.1-3 (15-90) - Unresolved - We recommended that $\bar{E}$ determination be started when gross activity in the reactor coolant exceeds 24 μCi/ml because 240 μCi/ml is much too close to the Tech Spec limit of 293 μCi/ml to be reasonable.

We also recommended that unit vent gaseous release be tested for particulate as well as iodine and that a 50% increase in gross release rate within 24 hours be reason for determining the release rate increase for iodines and particulates provided the gross release rate exceeds 1% of the maximum allowable release rate.

<u>ECCS and RB Cooling System Testing (p 15-93)</u> - Unresolved - We noted that the motor-operated valves in the core flooding tanks should be exercised for as short a time as practicable to determine mechanical integrity. Duke did not feel these valves are engineered safety feature valves and, therefore had not intended to verify their operability. Based on further discussion, Duke will cover these valves to indicate functional verification upon return to pressure after cold shutdown.

A basis will be added to address motor-operated core flooding tank valves.

<u>Containment Leakage Tests</u> (p 15-102) - Unresolved - We require all applicants to determine that containment leakage measured at less than accident pressure be the lesser of two fractions of allowable accident condition leakage as determined by (1) the ratio of leakages measured at these two pressures prior to initial unit operation, (L_{tm}/L_{pm}), or (2) by the square root of the ratios of the two pressures, $(P_t/P_p)^{1/2}$.

Duke resists use of the L_{tm}/L_{pm} method as being potentially misleading or even indeterminate where actual leakage measured is extremely low magnifying measurement errors perhaps even to the point of resulting in apparent negative leakage (in leakage).

Our position is that actual test data indicates that containment leakage paths are too complex to make the $(P_t/P_p)^{1/2}$ model alone an infallable extrapolation tool. Containments have been known to leak even less as pressure is increased. At this point in time we are willing to permit the reduced pressure test provided both methods of extrapolation are used and the most conservative results accepted. Otherwise, a full accident condition pressure will be acceptable for the periodic integrated leak test.

Reactor Building Hydrogen Purge System (p 15-108a) - Unresolved - We stated than an in-place system test with the portable unit hooked-up should be performed initially and during each refueling period to verify full inventory and functional performance of the complete system. We also stated that the hydrogen concentration instrumentation accuracy should be verified initially and periodically.

Further we note that the FSAR (p 14A-14) indicates that hydrogen concentration samples can be obtained both from the main reactor and from the purge line to obtain representative conditions.

The bases should include justification for the hydrogen concentration measurement techniques to be employed. If moisture content is significant (as it may well be) means for taking this into account should be included also.

Emergency Feedwater Pump Tests - p15-117) - Resolved - Duke has agreed to specify a minimum operating time to assure that the pump has reached operating conditions during the periodic test.

Table 15.4.11-1 - Resolved - Duke has agreed to minor changes and corrections to this table which (1) will require gross alpha and beta activity analysis on a scheduled basis regardless of activity level and (2) add K^{40} and I^{131} to the gamma analysis for fish and milk samples and (3) require Keowee River water samples to be analyzed monthly.

Reactor Design (p 15-127) - Resolved - We informed Duke that these Technical Specifications cannot apply to other "similar" design reload fuel because this is, at present, an unreviewed safety matter. Therefore, Specification A.4 was eliminated.

Fuel Storage (p 15-129) - Resolved - It was agreed for Specification 15.5.4.B.2 that the spent fuel pool should be filled with borated water with a minimum concentration of 1800 ppm boron when fuel is present.

General Office REview Committee (p 15-132) - Resolved - It was agreed that this Committee shall have only one member who is also a member of the Oconee Station staff. Duke prefers that person to be the Superintendent of Oconee.

Station Operating Procedures - Resolved - In Specification 15.6.2.A, "procedures" will be stated to include applicable check-off lists, and instructions. In Specification C, shift supervisor approval of temporary minor changes to written procedures shall be written also. In Specification D, selected drills will be conducted quarterly.

Radiological Controls - (p 15-141) - Unresolved - Duke has been told to respond to the January 10, 1971 letter (on the subject of an exemption to CFR Part 20 for the use of respiratory equipment) with the data requested therein. We can then procede by incorporating the exemption in this Technical Specification as initially issued if response is timely. If not, the exemption and credit for use of this equipment will have to be covered by an amendment.

ENCLOSURE 2

ATTENDANCE LIST 1-28-71

Duke Power Company

Paul Barton
William Parker
Edward Smith
L. Lewis
L. Snow
K. Canady
B. Rice

B&W

R. Straub

AEC

M. M. Mann, DR
R. C. DeYoung, DRL*
R. Klecker, DRL
C. G. Long, DRL*
A. Schwencer, DRL
N. Thomasson, DRL*
M. S. Hildreth, CO Hq.
C. M. Upright, CO Region II
R. L. Burrey, DRL
D. R. Roth, DRL

*Indicates Part Time