Docket 50-269

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 3, 1970

Roger S. Boyd, Assistant Director for Reactor Projects, Division of Reactor Licensing

THRU: Charles G. Long, Chief, Reactor Project Branch No. 3, Division of Reactor Licensing

MEETING WITH DUKE POWER COMPANY ON OCONEE NUCLEAR STATION, DOCKET NOS 50-269, 50-270, and 50-287

INTRODUCTION

A meeting was held with Duke Power Company January 21, 22, 1970 to discuss referenced B&W Topical Reports 10006 (RV Surveillance), 10007 (Rod Drives), 10018 (RV Thermal Shock), 10008 (Combined Loading of Reactor Internals including fuel assemblies), and 10012 (RV Model Flow Tests), to discuss accident analyses, Tech Specs, startup tests, prompt failed fuel detection and electrical penetrations.

These discussion areas were informally communicated to Duke in sufficient time for them to prepare for the meeting and bring the necessary technical personnel. An attendance list is attached.

DISPOSITION OF PROPOSED MANAGEMENT POSITIONS

1. Reactor Vessel Model Flow Tests, BAW-10012

We notified Duke Power (and thus B&W) that core flow distributions were not accurately determined by the tests. Our conclusion was based on these facts:

(1) The model assembly was not representative with respect to cross flow.
(2) Distribution tests were not run with an open vent valve.
(3) Back-flow was not simulated through idle loops for tests with less than four pumps running.

B&W stated that it was their design procedure to assume that the "hot assembly" receives 5% less flow than the average, and that their all-pumps data show this assumption to be conservative. Based on an independent review of their data we can also agree that this is conservative (for four pump operation).

Since the less-than-four pumps data was not truly represented, we stated that more thermal margin may be required in the consideration of the tech spec LCO for these cases.

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The next course of action is to direct Duke Power to submit the predicted safety limit curves for less than four pump operation. We should notify them to incorporate more conservatism in their bases for such operation.

2. Reactor Surveillance Program Capsule Requirements

Based on prior discussions with RT personnel B&W understood that the staff would not accept less than 5 mid-plane capsules for the Duke application. Accordingly, B&W presented at the meeting a revised program which they felt would more than meet our desires. Since the applicant has "volunteered" to increase from the four capsules the staff accepted at the CP review stage, we no longer have the difficult burden of proving that more than four capsules provide "substantial additional protection which is required for health and safety ..." per 10 CFR 50.109 (published April 16, 1969 as a proposed rule).

3. Summaries of Startup Tests

We told Duke that a brief summary of safety related startup tests would be required on the record for our review prior to meeting with the ACRS. Duke felt that this will be a substantial burden, one which they had not planned for. They will examine the list of tests informally discussed with the staff to see which of those tests are significant to safety. They also stated approximately half of the tests procedures have not yet been prepared. We agreed to consider changes (deletions) to the informal list provided significant safety related procedures would not be deleted from the summaries.

SUMMARY OF ITEMS DISCUSSED

1. Reactor Surveillance Program Revisions

A substantial change was presented which now appears capable of accomodating our current thinking on desired number of capsules per reactor vessel. There will be six capsules at the centerline position plus two thermal capsules. Total specimens have been reduced. Originally there were a total of 304 specimens (in four capsules). Now there will be a total of 252 specimens (in six capsules). There will be two centerline capsules "piggy back" in three tube holders (originally there were four tube holders, one centerline capsule each). Two of the three tube holders will also contain one of the thermal capsules.

Duke plans to remove capsules on the follo[illegible] schedule:
Oconee 1 & 2 - After 4, 20, and 40 years of equivalent full power operations (3 capsules will be spares).
Oconee 3 - After 8 and 30 years of equivalent full power operation (four capsules will be spares).

We learned that the reactor vessel for Oconee 2 will use centerline ring forgings (Lukens or Laddish?), hence are considered "prototype" by B&W for specimen withdrawal purposes.

B&W will submit a revised BAW 10006 reflecting these changes. Although we did not comment adversely on any aspects of the proposed revisions as presented at the meeting, it is improbable that we will agree on the withdrawal schedules as presented.

2. BAW Topical 10008 Parts 1 & 2 - Loadings on Reactor Vessel Internals

B&W presented information developed prior to the meeting in an attempt to be responsive to our expressed areas of concern. This was followed by discussion to obtain further clarification. In regard to seismic amplification, Duke said they considered no soil amplification for their site due to foundations being on bedrock. They did look at slab stiffness and concluded that it did not contribute significantly to the loading.

B&W also briefly described the R&D work done to set parameters for analyzing combined loads on the fuel assemblies. They discussed how they obtained natural frequency, assembly dampening, and spacer grid assembly factors.

3. BAW Topical 10007 - Control Rod Drive Systems

More information was presented at the meeting than was available in the report, including overload testing, drive-in tests against immovable object (testing machine), drive in and out past failed limit switches, and routine "idiot" tests on each drive. We said we thought the information presented at the meeting was pertinent to our review, should be available on the record, and that questions may therefore be expected. In general, the development program seemed adequate. We noted that dissimilar welds (two per drive, near the rotor assembly) are not generally inspectable without drive removal. We also were told that no dry scrams were done (no liquid for snubber action is termed a dry scram). [Thus care must be taken in low-pressure low-power tests to maintain a liquid filled torque tube; this could be a matter for further investigation].

4. BAW Topical 10018 - Reactor Vessel Thermal Shock

Although data are still developing from the HSST program, and some was presented on slides at our meeting, B&W does not intend to modify or supplement the report in the near future. We said that we did not anticipate resolving this issue (i.e., the acceptability of their thermal shock analysis) during the course of the Oconee review. We also said that we had no further questions on this matter at this time.

5. Pre Operational Testing

We reviewed the abstract of a typical test and considered it an adequate summary for our purposes. We also examined a listing of "safety-related" tests prepared by Duke and noted that it appeared quite comprehensive.

We then notified Duke that we will require a summary, one page or so, on each start-up test (in a depth similar to the typical test shown to us). The summary should include prerequisites, objectives, methods, type data required, and acceptance criteria. We recognized that methods as described in such summaries may not be final but we would expect the acceptance criteria to be binding and in certain cases would expect them to be incorporated in the technical specifications.

On timing, we said these summaries should be available on the record prior to making our report to the ACRS.

Duke indicated they would perform a review to see which were significant safety related tests. They clearly were concerned that preparation of summaries on all tests on their informal list would be a substantial burden which had not been anticipated. Duke will discuss with us planned revisions to the list of tests to be summarized before preparing the requested summaries.

6. Electrical Penetrations

Duke showed us color-marked drawings of wire runs through penetrations to illustrate that redundant circuits are widely separated. In addition, all penetrations are filled with an inert gas and fitted with pressure gages (monitors). Inside containment, all circuits except high voltage lines are completed through moisture-sealed, insulated connectors (no splices or bolted joints) with no exposed live circuit metal. The high voltage connectors all have built-up insulated stress cones at the point where the cables are joined to the ceramic insulators of the penetration assemblies.

Outside containment, "pig tails" from the penetration assemblies are brought to terminal blocks inside metal junction boxes. The metal sheathing on the external cables is terminated and firmly clamped to these boxes. Inside the boxes, the individual wires of each cable are then run to appropriate positions on the terminal blocks.

Not all cable entrances to these junction boxes are from the bottom. We said we would like to know if these cable entrance holes are to be sealed and would like some assurance that penetration room piping could not damage "redundant circuit" boxes through either leakage or movement.

7. Prompt Failed Fuel Detection

Duke intends to rely on RIA-36 Reactor Coolant Letdown radiation monitors for detection of prompt fuel failures. These monitors have NaI crystal detectors placed normal to 3/8" OD type 304 instrument tubing. Duke asserts that the instrument selected will be sensitive to the failure of one fuel rod and that the instrument will not be "over ranged" for activities in the primary coolant equivalent to 3% defective fuel rods. Response times of less than two minutes were indicated. They say they have looked at all the present schemes proposed for promptly detecting failed fuel and conclude they have selected the best practical one. They consider no R&D is required for the Oconee plants. We said we would request documentation of this detection capability.

8. BAW-10012 - Reactor Vessel Model Flow Tests

B&W summarized the report in a slide presentation. We noted our concerns that:

1. the model fuel assembly would not accurately represent cross flow, and thus the outlet flow patterns were not necessarily representative.
2. the effect of an open internals vent valve were not modeled
3. back flow through idle loops during tests with less than all pumps running was not simulated.

B&W stated that they have not used (up to now) individual location flow factors as input to thermal analyses. They said that since the mixing studies showed that elements generally received their flow from the nearest inlet, it is not important to simulate backflow. This logic is less than overwhelming and we said as much. We concluded that the tests generally show that interior elements receive preferential (higher) flow). Since the thermal analysis assumes an arbitrary 5% flow decrease (from the average) we believe the all-pumps data to be conservative. We also said that the less-than-all pumps thermal analyses (yet to be completed) should produce more conservative limits due to greater uncertainties in flow distribution.

9. Steam Line Break Accident

B&W described the steam-line-break accident with FSAR acceptance criteria (intact core with minimum shutdown margin, no tube rupture, and doses with 10 CFR 100). He constrasted the water inventory-vs-load characteristics of the Oconee plant with a plant using U-tube steam generators.

	Oconee	U-Tube Plant
Reactor Coolant Vol. (ft^3)	11,478	9,088
SG Inventory at 100% power (lb)	62,600	84,586
SG Inventory at 0% power (lb)	20,000	160,000
EOL Mod. Coeff. $(\Delta k/k)/°F$	-3×10^{-4}	-3.5×10^{-4}

Making a heat balance for the two worst cases (100% power for Oconee; 0% power for the U-tube plant) B&W computes a potential cooldown reactivity increase of 2% for Oconee and 8% for the U-tube plant.

The computer program was briefly described as a hybrid digital-analog model of the primary and secondary systems in the same language appearing in B&W proprietary steam generator topical report, BAW-10002. It contains a five-region model of the SG conserving mass, energy, and momentum. The model simulates reactor and steam system trips at appropriate levels of neutron flux and reactor coolant pressure taking into account instrument response times. (The reactor protection system, in addition to releasing the control rods, also provides "turbine trip" signals to close the turbine stop valves and the feedwater valves.)

From the presentation we understood that it takes 25 to 30 seconds after turbine trip has been commanded for the feedwater valves to close. This is not in agreement with the 10-second closing time (0% FW flow 16 seconds after break) listed on FSAR page 14-22.

If there is no operator action, the integrated control system will override this protective system action by reopening the FW valves when the water level in the steam generator reaches a nominal 2-foot level. Since the steam generator with the broken steam line will blow dry (0 water level) in about 50 seconds, the operator must act in less time to prevent this unwanted ICS override action.

To maintain the other steam generator as an effective heat exchanger the applicant must either rely on the ICS to reopen the feedwater valves to start replenishing feedwater before boiloff (through the bypass valve) can reduce it below the 2-foot level or manually reopen these FW valves. (Although not clearly stated, from the discussion on FSAR page 14-17 it is likely that this heat exchanger could be lost for about 15 minutes before reactor coolant would boiloff and for about 98 minutes before the top of the core would be uncovered.)

We told B&W and Duke that, from their statement, we could only assume credit was taken for operation of the ICS to show ability to safely withstand the steam-line-break accident. They must either demonstrate this is not the case (permit ICS to fail) or show that the portion of the ICS so used meets our requirements for a protective system (complies with IEEE-279).

We further noted their inability at the meeting to assure us that the "turbine trip" system meets IEEE-27 in achieving and maintaining the required steam generator isolation (proper closing of stop valves and feedwater valves). We told them we would need to see details of the "turbine trip" system showing how these valves are activated and the relationships which exist between protective action and control (ICS) function circuits.

The applicant explained he had examined five cases each relying on the integrated control system to close the feedwater valves:

(1) minimum shutdown, stuck rod, operator action
(2) nominal shutdown, no stuck rod, no operator action
(3) nominal shutdown, stuck rod, no operator action
(4) minimum shutdown, no stuck rod, no operator action
(5) minimum shutdown, stuck rod, no operator action

In the last four cases the FW valves open automatically (ICS action) and flow returns to about 30%. B&W stated that only cas (5) led to a return to criticality and a resulting power of 35%. (The AR states that case (3) also leads to a return to criticality.) cases (2) to (5) the HP injection system must activate. (Reactor pressure drops below 1500 psi and reactivity credit is taken for the injected boron.)

Note: The remaining accidents had not yet been reviewed in depth by the staff and the applicant was not expected to be prepared with detailed answers at this meeting.

10. Loss of Coolant Accidents

We asked the applicant for his acceptance criteria for LOCA accidents and B&W cited FSAR page 14-37 which calls for maintenance of core geometry and prevention of clad melting. We said his quantitative acceptance criteria appeared to be the 2300° F design limit for cladding temperature needed to support an assumption of less than 1% metal-water reaction, and a resulting containment design pressure of less than 59 psig. We said the 2300° F would have to be defended and asked when the results of his R&D program on cladding failure mentioned on page 14-47 of the FSAR would be available. Submittal within a month or so is expected. We asked about their use of the three-node FLASH program for the two-loop plant, but those present did not have the details. Because of the lack of information in the FSAR on cold-leg breaks, we asked whether B&W thought flow reversal took place. B&W replied the FLASH program allowed flow reversal.

We asked if B&W thought failure of one cold leg could cause failure of the adjacent cold leg in the same loop (these legs share the lower head

of the steam generator). B&W indicated this has not been evaluated, that to the best of their knowledge, no applicant had ever been asked on the record to consider multiple-leg failures and that such a question if asked officially would cause a major uproar in the nuclear industry. B&W would prefer to pursue any concern we may have in this area informally at this time.

We told the applicant we needed greater detail on the use of codes, and a complete description of QUENCH, the core heatup code.

We questioned why the applicant proposes to activate the HP injection system at a reactor coolant pressure of 1500 psi. This is below the hot leg saturation pressure and is 300 psi below the pressure originally proposed as stated in the PSAR at the CP review stage. Our concern was the possibility of a small-size break causing a pressure hang-up at saturation pressure (1600 psi) that could uncover a portion of the core without activating the HP injection system. The applicant replied that since the CP review stage, more detailed analyses of certain normal transients indicated the 1800 psi level might be breached; therefore they reduced the HP injection system actuation setting to 1500 psi to prevent "nuisance" actuation. They were not sure they needed to go as low as 1500 psi to achieve this however.

11. Maximum Hypothetical Accident

We commented that no details of the whole body noble gas cloud dose were given in the FSAR and we would want to discuss the details of their calculations.

12. Flywheel Missile Accidents

A presentation was made in support of the applicant's position that a flywheel accident is not credible. The 9000 HP reactor coolant pump motors have a rated speed of 1188 rpm and, as part of the station auxiliaries, are tied to the frequency of the turbine generator. The T-G has redundant overspeed governors which limit frequency (and motor speed) to 106% of rated value during an offsite load rejection transient (turbine still carries a 5% plant auxiliary load). There is also a mechanical overspeed T-G trip at 110% rated speed, an overfrequency T-G trip at 111%, an electrical back-up overspeed T-G trip at 112%, and individual reactor coolant motor overspeed protection at 115%. At flywheel design speed of 125% the maximum tangential stress has been calculated by Westinghouse, the motor-flywheel supplier, to be 20,000 psi, well below the 50,000 psi minimum yield stress of the flywheel material. At 100% speed the stress is 12,500 psi giving a normal operation safety factor of 4 when compared

to yield stress. The flywheel is made of degassed ASTM A533 steel and has a 100% volumetric (ultrasonic) inspection before machining the bore. Charpy V-notch specimens are obtained parallel and perpendicular to direction of plate rolling to assure 30 ft-lb at 10° F, after machining a liquid penetrant surface test is made on the finished bore. The pump motors and flywheels are subjected to a 125% overspeed shop test and the applicant concludes the flywheels cannot rupture. The shafts will see a maximum applied shear stress of 5500 psi which is well below their 23,000 psi minimum yeild shear stress. The maximum applied shear stress would occur due to braking action of the motor under short circuit conditions.

Kingsbury double thrust bearings are used having a 40-year design life. Experience with these bearings says they do not exhibit sudden failure (seizure). In addition, the temperature of the motor bearings will be monitored throughout plant life to detect abnormal heating should it develop.

The maintenance program will call for two base mappings of the flywheel. The first one will be a full volumetric (UT) examination and complete surface test. The second base mapping will be a volumetric examination of the flywheel bore and keyway following the 125% speed run; this test will be repeated at approximately 10 year intervals.

13. Fuel Handling Accident

The applicant arbitrarily assumes 15 fuel pins (one side) as the credible number of pins ruptured in a fuel handling accident. He then, also arbitrarily, increases this to 56 pins (outer ring) for computing the consequences of this accident. We told the applicant that unless he has a basis for 56 pins (tests or calculations), we have no alternative but to assume all pins in an element (i.e., 208) rupture. We discussed this with him in November 1969. He has made no apparent effort to justify his number of 56 pins since that time.

14. Pressurizer Level Instrumentation Related to Accidents

Noting that several accidents are analyzed assuming an initial "nominal" pressurizer level (18.33 feet), we questioned the reliability of level indication. Also, we noted that no reactor trip is provided by B&W which constrasts with such a trip being provided by other PWR NSS suppliers.

Although B&W was of the opinion that it would be incredible not to know pressurizer level they could not assure us that a single disabling failure in the level instrumentation could not occur (they declined to say it meets IEEE-279 requirements). We told them that based on what

they had said and the information now on the record, we would need assurance that no single failure could cause loss of pressurizer level indication and that we would need sufficient information (description, schematics, logic diagrams) on the final design to show this.

Also, B&W maintained they had examined the need for a pressurizer level reactor trip and read us their answer to Q 13.1.1 submitted in Amendment No. 6 on the Midland Plant. We said that since other PWR NSS suppliers have found it necessary to have a pressurizer level reactor trip, we want to be quite sure that B&W has not overlooked some serious consequences of loss of pressurizer level. As an example, we said if they had evaluated the effects of thermal stresses on the pressurizer vessel that might be caused by operating with the pressurizer heaters uncovered. [This could happen if heater interlocks (activated by pressurizer level) did not work because of failure of the pressurizer level instrumentation.] B&W has not examined this possibility. We will need more information on the lack of pressurizer level reactor trip.

When questioned about the capacity of the two code safety relief valves atop the pressurizer, B&W stated that they were sized by the start-up accident which would require greater relief capacity than the rod-ejection accident. B&W treats the relief valves as passive componenets and assumes that both code values will open as required if reactor coolant pressure should exceed their set points.

15. Technical Specifications

On rated power (page 15-1 of FSAR) Duke plans to use secondary heat balance because of their ability to readily calibrate flow devices used. We did not object to this.

We noted that figure 15-1, related to preventing DNB, covers only four pump operation whereas Duke plans to operate, on occasion, with less than four pumps. This must be addressed also.

With regard to the LCO for reactor coolant activity, Duke intends to base this on a 10 CFR 20 release path through the steam generator tubes (3 gpm) taking credit for partition factors associated with release through the air ejector exhaust. We noted that this assumes availability of condenser cooling water (ccw) and no significant direct release through the steam relief valves. We also noted that their proposed value of 1200 uCi/cc is a factor of 15 higher than recently approved PWR's, is based on a leak rate three times higher than we have been approving, and assumes an atmospheric dilution factor which has not yet been substantiated. Since Oconee is unique with its ability to maintain emergency condenser cooling upon loss of all ac power, we will need to know how long it takes to establish this emergency (gravity) cooling system and either assure ourselves that it will always be available (single failure criterion) on a timely basis or determine the consequences of its unavailability

upon loss of offsite power. (The emergency CCW line to the Keowee tailrace is normally held closed by a power-to-close valve.) We expressed concern about the proposed weekly calibration of the reactor neutron flux instrumentation, and said that radial redistribution due to Xenon shifts during load changes would require more frequent calibration. Applicant personnel at the meeting (including those from B&W) seemed unaware of a radial power distribution problem.

We called attention to the following LCO's not yet covered in the Tech Specs:

a. Reactor coolant chemistry
b. Reactivity anomalies (see surveillance 15.4.8)
c. Penetration room
d. Radwaste release limits - liquid and gas
e. Minimum temperature for cirticality
f. Coefficients of reactivity

We called attention to the following surveillance items which are not yet covered:

a. Environmental survey
b. Coolant chemistry activity
c. Reactor coolant system hydro tests

A. Schwencer
Project Leader
Reactor Project Branch No. 3
Division of Reactor Licensing

Attachment:
Attendance List

Distribution:
P. A. Morris
F. Schroeder
T. R. Wilson
DRL Branch Chief
S. Levine
D. Skovholt
R. C. DeYoung
AEC Attendees
Docket file
DRL Reading
RPB-3 Reading

ATTENDANCE LIST

DUKE POWER COMPANY OCONEE MEETING JANUARY 21-22, 1970

Duke Power Company
Austin Cole Theis
Paul Hodges Barton
Warren Herbert Owen
Charles Joseph Wylie
William Oscar Parker
John Edwin Smith
Lionel (NMI) Lewis
Carl Amos Price
William Joseph Foley
Thomas Fulton Wyke

AEC Regulator
C. G. Long
A. Schwencer
D. Ross
B. Cady*
M. B. Fairtile*
K. R. Wichman*
F. P. Schauer*
J. P. Knight*
O. D. Parr*
J. M. McGough*
R. A. Birkel*
M. S. Dunnenfeld*

Babcock & Wilcox Co.
Donald Wheaton Montgomery
William Ruben Smith III
George Eugene Kulynych
Ronald Clyde Hulto
Glenn Jacob Snyder (21)**
Charles Joseph Bowch (21)
John (NMI) Ficor (21)
Douglas Murray Collings (21)
David Warren Gerger (21)
Elbert O'Neil Hooker (21)
Harvey Darius Ferris (21)
Richard Raymond Steinke (21)
Robert Jackson Walker (22)
Lawrence James Stonek (22)
James Francis Malloy (22)
William Stanley Little (22)
Jose Mario Sonis (22)

(B&W also brought along five non-participating observers)

*Part Time
**Present on day indicated