Docket File
50-269

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

December 12, 1969

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
AS for

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

SUMMARY

A meeting was held with Duke Power Company November 18, 19, 1969 to discuss the FSAR review areas of site, instrumentation and electrical, reactor physics, conduct of operations, initial tests, internal vent valves, and steam generators. An attendance list is attached.

Prior to the meeting, reviewers in these areas had identified areas that will require further information. Discussions were held with each reviewer to identify areas that required discussion as opposed to straight forward information requests requiring no discussion.

An agenda was prepared and informally communicated to Duke in sufficient time for them to prepare for the meeting and bring the necessary technical personnel.

A brief general meeting for introductions and scheduling discussion sessions included a presentation by Ray Maccary of an NDT "Table A" prepared for Oconee Unit 1. Except for brief summary sessions the remainder of the two days were devoted to concurrent discussions of agenda items scheduled to minimize manpower commitments. D. Ross chaired discussions on site, conduct of operations and initial tests, and reactor physics. A Schwencer chaired discussions on instrumentation, electrical, internal vent values, and steam generators.

In a concluding discussion with the project staff Duke expressed concern on completion of review of balance of FSAR. We stated we are still working on an accelerated schedule from that discussed at our initial meeting but could offer no guarantees on completion. We expressed concern that Duke has not yet resolved the meterology problem posed by the valley drainage model which requires further justification if it is to be used.

Details of discussions are given below.

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NDT TABLE A

Ray Maccary, DRS, explained "Table A," an NDT requirement document he had prepared for Oconee Unit 1. It is applicable to piping, pipe fittings, pumps and valves within the reactor coolant pressure boundary. He stated that Duke must be in a position to certify (to Compliance) to Table A. Duke indicated that both they and B&W had surveyed their systems and believe they either meet or exceed Table A requirements. They indicated that in some cases (such as certain B&W supplied valves) there would be certification sheets only in their possession or in B&W's possession. In another area of concern, they were told that Table A would be applicable to Unit 2 and Unit 3 items that have already been purchased.

SITE

1. Onsite Monitoring - There will be two monitoring points downstream of radwaste release. One, at the Highway 183 bridge, is inside the exclusion area. Another is several miles downstream at Highway 26 bridge. There will be a continuous water sampler at the Highway 183 location. Boiled down samples will be counted periodically. During this part of the discussion we asked if Duke had verified the minimum dilution flow from the Keowee tailrace with no hydro units operating (30 cfs was assumed in answer to Q.2.3, Supplement No. 1). Duke indicated this could be done.

2. Flexibility in Environmental Monitoring- We pointed out there is a need to specify criteria for contracting or enlarging this program before the fact. (Duke had already told the Compliance inspector during a September 9-12 site visit that the statement in paragraph 2.7.3 of the FSAR was intended to cover only additional monitoring stations not fewer stations).

3. Penetration Room Leakage - We noted that the FSAR contains very little information on this engineered safety feature. (Boundaries are not well defined, design data are not given, discussion is not provided on doors or other openings, etc.) We learned they intend to operate at approximately 0.5 in Hg vacuum. Vacuum relief will be provided at 2 in.Hg. Alarms will annunciate at that level as well as at 0.1 in.Hg. Air flow will be controlled by a pressure signal, not flow. (They implied that filter face velocity could be independently controlled.) We expressed concern on how they can and will assure themselves that they get a good vacuum distributed throughout the penetration room which has several doorways and is formed into "east" and "west" sections by a constriction at the fuel pool. They apparently had not considered this. Part of this concern is how they will assure integrity of room seals such as around doors and at wall joints, etc. American Air Filter is supplying the filter package.

We discussed how the air would be exhausted and filtered. From Figure 6-5 it appears that a potential may exist for loss of air flow (cooling)in one of he two filter trains causing filter heating and potential desorption. Apparently Duke had not examined this as a possible mode of failure. In pursuing charcoal filter heatup, we confirmed that they had calculated heat load on the basis of the filter passing a 0.5% leak. We then pointed out and they agreed that their proposed tech spec would permit a containment leak rate up to 2.75%. This is 1 1/2 times the containment design leak rate. We noted this would be an area requiring furthern consideration. If we were to accept a leak rate above 0.5% then their heat load calculations are no longer valid. Also, the contribution of noble gases, now based on a 0.5% maximum leak rate, would be invalid. I think we would have to have convincing information showing the 0.5% leak rate is unreasonable before relaxing it at this stage.

4. Radiation Exposure of Station Personnel - The applicant stated that their calculations indicated less than 1 rem in 90 days would be received by station personnel under worst accident conditions.

5. Interaction of Liquid & Gaseous Radwaste Systems - We were concerned that malfunction of bleedback valve WD-V66 could overpressure the liquid waste tanks. Duke said this would be prevented by a relief valve that discharges through the gaseous release filter train. We did not obtain details of maximum pressure that might be reached on the discharge capacity of the relief valve. We said we would need additional information on this provision.

6. Radmonitoring Instrument Range Inconsistencies - We asked how Duke justified radiation monitors whose ranges did not cover maximum accident or technical specification radiation levels. As an example we cited the monitor RIA-36 which has an upper range of 50 uCi/cc for reactor coolant activity whereas Duke calculated 269 uCi/cc for the 1% failed fuel assumption. Duke intends to substitute a less sensitive GM type instrument for the NaI crystal type identified in the FSAR. We asked what this might do to their ability to promptly detect failed fuel by means of increase in primary activity. From their response, it appeared that there is no other failed fuel detection means being considered at this time.

7. Reactor Coolant Storage Inter Connections (3 Units) - In our discussion over the potential to make up from tanks of varying boron concentrations, Duke pointed out that make up is actually from the lewdown storage tank (Figure 9-2) downstream of the interconnections between plants. Since Duke has analyzed the case for dilution water with zero boron concentration entering the letdown tank we agreed that these interconnections present no unreviewed safety item.

8. Isolaticn of Waste Gas Exhauster Line - We noted that Figure 11-3 shows an apparent single failure situation that would cause inadvertent discharge of gaseous waste through the waste gas exhauster line (a similar situation exists on discharge from the waste gas tanks). Since this could constitute an uncontrolled release we will need to see how it is prevented. From the discussion and information contained in Secion 11 of the FSAR it is apparent they will have the ability to hold up release by use of the waste gas tanks and need not use the waste gas exhauster except under very favorable meterological conditions.

9. Moderator Dilution Accident - The accident analysis in Section 14.1.2.4.1 ι tes an inflow of 500 gpm but does not give letdown flow to letdown storage tank. Duke said this would be a maximum of 140 gpm. They did not feel this flow rate would tax the available tankage since it would be terminated by a reactor trip.

10. Ability to Minimize Offsite Liquid Release - We discussed in general the ability of the present system designs to minimize offsite releases. From the discussion and the data contained in the FSAR it appears evident that they ahve ample capacity to holdup releases containing significant levels of activity for extended periods of time (over 60 days). Except for showers and laundry which can be expected to contain only trace activity and reactor coolant bleed (expansion dilution and partial drain) which is normally processed and held in the coolant bleed holdup tanks for return to the reactor coolant system, less than 4,000 ft^3 waste is generated in 60 days. This compares with a holdup capacity of 6,900 ft^3 not counting the reactor building sumps.

11. Refueling Accident Doses - We noted that we had some concern over the applicability of their accident model which assumes only 56 of the 208 pins of a fuel element fail. Since failure of all 208 pins could result in a site boundary dose of approximately 500 Rem to the thyroid, we asked what means might be possible to accomodate a 208 pin failure within part 100 guidelines. Duke would prefer not to consider the addition of another set of charcoal filters. They said it might be possible to utilize the penetration room filter system, but they had not considered this. B&W did not offer any experimental or calculational data to support their 56 pin failure assumption.

12. Containment Integrity During Shutdown - Duke would like to keep the personnel and equipment hatches open during refueling and maintenance shutdown. We said it was not clear to us that it was necessary to keep them open except for those brief periods when material or personnel were being moved in or out. Further, we could not see the need for them to be open at all when irradiated fuel was being handled. It should be a simple matter to keep one of the two personnel hatch doors sealed and the equipment hatch closed with at least four (equally spaced) bolts in place. Duke

could offer no basis except "convenience" for wanting to keep them open, and said they would have to evaluate what the effect on shutdown time would be if they were required to keep these hatches closed.

REACTOR

1. Core Enrichments - Oconee 1 has a three zone core. The first cycle enrichments have been set. The reload enrichment is not yet definite but will be very close to the values used in the physics analyses. Initial enrichments for Oconee 2 will be set in the Spring, 1970 and for Oconee 3 in the Fall, 1970. The reloading patterns are known and fuel cycle calculations have been made for Oconee 1 and work on Oconee 2 fuel cycles has been started

2. Reactivity Eigenvalue Calculations - Eigenvalue calculations as a function of life for several conditions (e.g. hot, 100% power, poisoned,) have been calculated for Oconee 1 and results are available if we ask for them. They are, however, still labeled as "preliminary." They are 2-dimensional PDQ-5 and -7 calculations.

The basic power scheme for Oconee 1 is 0.8 to 1.2% $\Delta k/k$ reactivity held down by a Xenon transient control group (always designated as group 7) and an average of 0.2% $\Delta k/k$ held down by partial insertion of group 6 for boron dilution control. The rest of the excess reactivity at hot, rated power, poisoned conditions is held down by boron shim. During the last month or so of life when boron has been reduced to its minimum of 17 ppm the transient Xenon group will be slowly withdrawn. This increases the shutdown margin by about 1% $\Delta k/k$ during the last month of operation. Therefore shutdown margins calculated at end of life (EOL) are about 1% greater than those one month earlier.

A pending internal B&W report discusses 2- and 3-dimensional (PDQ -5 & -7) flux shapes and eigenvalue calculations. We did not discuss our possible need to see this report.

3. Doppler Coefficients - Doppler coefficients come from internal subroutines of relatively standard physics codes and are not verified experimentally. As a substitute, B&W makes sensitivity analyses in accident and transient calculations to account for possible errors in knowledge of Doppler coefficients. Doppler coefficients have been calculated for beginning of life (BOL) and EOL, and Doppler feedback has been used in the thermal hydraulic analysis of the core.

4. Reactor Plant Dynamic Codes - We asked about B&W's computational ability to predict overall plant performance during abnormal transients. Jim Mallay answered that he had developed a digital-analog hybrid full-plant simulator code and it was used for the steam line break accident and had the capability to be used for other abnormal transients (e.g. rod ejection, loss of flow). The code is referred to in the B&W Steam Generator topical, but is not fully described or even named outside of B&W's shop. When we asked for the code's name, B&W responded that they would prefer not to answer.*

5. Moderator Reactivity Coefficients - Isothermal temperature and uniform void moderator coefficients of reactivity are calculated by 2-dimensional PDQ-5 codes. Power coefficients are not calculated because the constant T-average control scheme makes, in B&W's thinking, power coefficients equivalent to Doppler fuel coefficients. We said this assumption would be invalid for transients faster than the response time in the Integrated Control System.

6. Temperature Dependence of the Moderator Coefficient - B&W said they had calculated the positive moderator coefficients at BOL for several temperatures, but could not say how it behaved as a function of temperature except that it was more positive at room temperature than at operating temperature. They are studying this temperature coefficient because of its importance to understanding physics start-up tests. We mentioned that axial fuel expansion coefficients as well as Doppler coefficients may cause start-up test interpretations to be in error.

7. Operation With Positive Moderator Coefficients - We explained that operation at power with a positive moderator coefficient offered the possibility of sudden insertion of reactivity due to voiding. Voiding could be caused by any sudden decrease in primary system pressure (e.g. loss of flow, loss of coolant, rod ejection). We pointed out that Figure 3-5 shows a potential for the sudden insertion of 0.75% $\Delta k/k$ if the core goes from zero to 27% uniform voids and that the possibility of non-uniform insertion of voids may show even greater reactivity insertion potential. B&W had not considered such reactivity insertions nor had they made scoping analyses of the safety problems related to operation with positive moderator coefficients. We noted elaborate Tech Spec limitations were used on Connecticut Yankee. We indicated they should estimate the expected number of days of operation with boron concentrations large enough to give positive moderator coefficients.

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* From a private discussion with Mallay we learned that the lumped parameter models incorporated in the code also form the basis for B&W's Link training simulator being installed this month in Lynchburg. The physical models and plant equations may not be considered proprietary but the numerical schemes and code logic and programming details are considered proprietary.

Further, we indicated that they should consider a change in the coefficients due to buildup of equiblibrium Xenon and Samarium.

8. Control Rod Grouping - The control rod assemblies (CRA's) will be regrouped once during life and the function (shutdown, Xenon transient, dilution control) of the groups will be changed several times during core life. An expected history of CRA grouping has been established for Oconee 1 and is being worked on for Oconee 2.

9. Control Rod Worth - CRA worths, Xenon effects and power distributions are calculated by PDQ -5 and -7. Most calculations are 2-dimensional; for selected configurations they are 3-dimensional. A series of ejected CRA worth calculations as a function of lifetime for Oconee 1 have been made. Also, BOL and EOL calculations for the reactivity worth of a stuck CRA have been made.

We suggested that minimum shutdown worths for Oconee 2, 1st cycle might be inadequate. From FSAR information the total CRA worth is only 7.4% $\Delta k/k$. Of this 1.2% is in the core for transient control, 2.3% is the power defect, and 1.7% is the stuck CRA worth. This leaves only 2.2% $\Delta k/k$ as the hot shutdown margin. Considering errors in CRA worth and one stuck CRA, the shutdown margin may not give the required operating margin of 1% $\Delta k/k$. B&W replied with two important pieces of information, one good and one bad:

1. The total CRA worths appearing in the FSAR have already been reduced by 10% to account for calculational errors. This is not spelled out in the FSAR.
2. B&W has not considered the possibility that the stuck CRA criterion for shutdown must be in addition to a withdrawn, inoperable CRA and they indicated that there were times in life that the 1% $\Delta k/k$ margin would not be met if both of these CRA's did not insert.

We did not pursue this matter because of lack of detailed information on CRA worths as a function of life and because B&W said it would be easy to regroup rods to give increased shutdown margin. We will need more information to understand how this increased margin is to be obtained.

We asked for a physical explanation of why Oconee 3, 1st cycle has only 7.4% $\Delta k/k$ total CRA worth while Oconee 1, 1st cycle has 10.6% and the equilibrium cycle has 9.6%. Because Oconee ?, 1st cycle will have once-burned fuel elements in one zone from Oconee 1, we reasoned that CRA worths would be very similar to the 2nd cycle of Oconee 1 and therefore have a total CRA worth between 10.6% and 9.6%. B&W replied that this apparent discrepancy could be explained by the fact that many of the CRA's would be placed in the once-burned fuel elements.

10. Control Rod Malpositioning - We asked what instrumentation in addition to CRA position indicators would be effective in detecting a misplaced CRA. They replied that the in-core flux detectors would detect malpositions, but asserted that the detectors have no safety function. B&W said there were two independent rod position indicating schemes, and we said it was our understanding that only one of these detected CRA position, and the other detected position of the drive mechanism relative to its group. B&W said that if a Tech Spec on in-core detectors was contemplated by DRL they would prefer that it be in the form of a reduced power limit to account for in-core instrumentation being inoperable or deficient.

11. Xenon Instability Control - B&W is developing a Xenon instability control manual for reactor operators. It is based on detection of axial offset as measured by half length out-of-core flux detectors. The change to half length out-of-core detectors will be documented to us by a pending FSAR amendment. B&W said they expect Tech Spec limitations on axial offset.

B&W intends to use part length control rods routinely for power shaping, not just for axial Xenon oscillations. The predicted absence of azimuthal Xenon oscillation will be confirmed by in-core instrumentation measurements.

12. Summary - In summing up the preliminary discussions on reactor physics the applicant and B&W reacted to our informal expressions of need for additional information by stating they wanted as little of the design information on the public record as possible. We noted that we have not completed our review of this part of the FSAR and would decide at a later date what information need be added to the public record, what could be received in a proprietary submittal, and what could be reviewed informally (e.g. by a visit to B&W).

VENT VALVES (Proprietary Topical BAW-10005)

Note: Bill Smith of B&W said none of the vent valve information discussed was proprietary except the Foster Wheeler design report. On that basis, these notes are not considered to contain proprietary data.

1. Design Information - We noted that although the vent valves were an R&D development, no information was available giving the design basis and development effort which resulted in the prototype valves. B&W made one copy of a Foster Wheeler proprietary design report available at the meeting and indicated they would make this an appendix to their Topical Report BAW-10005. Only minor modifications were made during development, mostly relating to elimination of jack screw galling.

2. Venting Capacity - B&W had just submitted this information on the Midland plant and provided us with a handout copy (their reply to Midland Question 6.1.4).

3. Materials - B&W's reply to Midland Question 6.1.1 was cited for this information.

4. Hinge Clearances - These clearances are provided in answer to Midland Question 6.1.2. A design criterion was a Class I fit or looser. It was noted that the hinge pin is held captive by a concentrically "staked" disc at each end.

5. Design Basis for Opening and Withstand Pressure - The opening pressure 0.5 psi is based on keeping approximately 1.5 feet of water above the core. The withstand pressure (600 psi) was derived from calculating a differential closing pressure of 585 psi for a 36" pipe break at full power. BAW 10008 Part 1 gives 515 psi for this pressure indicating a margin of conservatism in the 600 psi design (which was tested at 750 psi).

6. Basis for Opening and Closing Forces During Inspection - B&W stated that while they had measured 30 pounds to move the valve off its seat and 120 pounds to hold it in the full open position, they could permit 120 pounds and 540 pounds, respectively, based on criteria of 1/2 pound psi to start movement and 1 1/2 pounds to hold full open. They plan, however to establish a lower set of limits, taking into account "base line" data on production valve performance.

7. Partial Flow Performance - B&W stated they had included this information in BAW 10012.

8. Plastic Deformation - The impact analysis showing plastic hinging and deformation of the disc is contained in the proprietary Foster Wheeler report. Based on the discussion it did not appear that this impact would prevent the valve from remaining open following deformation. B&W said that once the valve opened, it would remain open and that repeated impacts are not to be expected during the course of an accident.

9. Vibration - B&W stated that their calculations show less than 1.0 mil of vertical vibratory movement possible at the vent valve location. They assured us that the ATL tests encompassed all frequencies and amplitudes considered possibly present at the vent valve location. They monitored for resonance throughout the tests with stethoscope, strobe light, and examination of all recording traces. None was found. They expected none because they said natural resonant requency of the valve assembly is approximately 1500 cps, well above the excitation frequencies used in the test.

10. Demonstration of Remote Inspection and Removal Capability - We said we believed they should demonstrate these capabilities after the system had experienced hot flow conditions. Duke had been thinking of doing this cold. This can be resolved during our review of the pre-operation tests.

11. Jackscrews - We expressed a concern that this mechanism might have a potential for galling and seizing or loss of small parts inside the core barrel. B&W appears to have been sensitive to these possible problems. They had changed thread type, material type, and thickness of thread coating. The jackscrew automatic locking device is covered by a housing. Nuts holding the housing in place each have one tack weld (not in high flux field) to prevent backing off. The two jackscrews are synchronized, but B&W says they can be misaligned by two turns of a screw with no problems.

ONCE THROUGH STEAM GENERATOR - Proprietary Report BAW-10002

Note: Bill Smith of B&W advised the following areas of the discussion were proprietary: (a) boiling length, (b) excess surface and fouling factors, (c) stability discussion and analytical stability computer model development, and (d) feedwater nozzle design and performance. Since these notes do not contain specific design information concerning these areas, they are not considered to reveal proprietary information.

1. Technology Base - B&W indicated that success (economic) of their design depended upon ability to transfer heat within the nucleate boiling regime over a wide range of steam quality at relatively lower velocities than more conventional heat exchangers. (It was not clear how this is unique with the B&W design.) They presented a curve showing that ample data is available to ensure these conditions at high velocities. They also said that based on their Research Center tests of 1-inch and 3/4-inch heated tubes they obtained reasonable confidence that these conditions could be satisified at very low velocities. These tests also established a relationship that indicated they could expect to stay within the nucleate boiling regime for qualities above 45%.

From the limited discussion it appeared that heavy reliance is being placed on empirical results obtained from their 7-, 19-, and 37-tube boiler tests. B&W apparently has also developed several computer programs to assist in the design of the steam generator and to predict its behavior under dynamic conditions. These programs were not described to us in any detail.

2. Full-Scale Verification - B&W said thermocouples would be placed at several locations on the S-G, but did not describe a planned design verification test program. We indicated further information in this area would be needed.

3. Extrapolation of Model Results - We noted that the largest of their models had 37 tubes which represents less than 0.3 percent of the tubes in the production units. They have attempted to justify this extrapolation by use of a "figure-of-merit" ratio (inner tubes to total tubes). We said we needed the basis for use of this figure of merit. Also, it was not clear that it is applicable to all aspects of the design, including the transient performance and primary and secondary flow maldistributions that might exist in the 15,000-tube production units.

4. Direct Feedwater Heating (B-- Spray) - The key to this process was to adequately heat the feedwater as it falls through a steam atmosphere (extracted from the boiling section of the S-G) in such a way as to prevent cold shocking the outer shell of the steam generator. From the discussion it appears that they have evolved a satisfactory design provided fouling conditions do not flood the nozzles. B&W said the nozzles can be inspected during plant shutdown.

5. Tube Fouling - We expressed concern that Duke have a good handle on steam generator "state-of-fouling" since as fouling increases (1) pressure oscillations have been shown to increase and (2) pressure drop increases-decreasing the spray fall height for the feedwater, which if excessive could flood the nozzles. We also asked what cleaning method has been selected and what thickness of sacrificial metal they intend to use to account for possible metal loss due to cleaning and possible mechanical abrasion at tube support plates. From the discussion, it appeared that no allowance may have been made for tube metal loss. Also, Duke appeared reluctant to discuss the possibility of verifying integrity of tube walls (UT techniques are available) due to the expected high radiation field at the tube sheet access areas.

We pointed out that chemical cleaning holds the potential for causing multiple tube failures and, for that reason, we expect to have some assurances that the cleaning process selected (materials, temperatures, and cleaning times) are well proven and conservative assumptions are used regarding metal loss.

6. Transient Response - We noted that while transient response curves are shown for normal system transients none are shown for the abnormal transient tests performed and that we would need more information here.

7. Basis for Conclusions - We noted that in several areas of the report conclusions are stated without a basis being available to us. We cited a conclusion in paragraph 3.4.3 that stated there were no failures based on an "examination." However, since no elaboration was provided, we have no clue as to the nature of the examination. We said we expected the bases for such conclusions to be in the report.

8. Vibration Testing - In discussing this test (about 3 days duration) we asked on what basis they concluded it represented an adequate wear test when compared to a 40 year plant life. B&W felt that the amplitude of the test vibrations (± .05 inches) was far greater than the tube would see in normal service and withstanding this amplitude showed the tube could survive lower amplitudes. We said we would need additional information on this test including nature of tube restraints and supports and some understanding of what is implied by "no consequential wear." As part of this discussion we learned that in the actual steam generator the tube supports will be solid plates drilled and broached to leave lands for the tube support and passageways for water/steam flow. The tube support plates will be the same throughout the length of the tube bundle. All tubes in the test models are made of Inconel 600. With some minor variations, probably trace impurities, they are identical to the full scale tube material.

9. Thermal Fatigue - We noted they had not addressed the subject of thermal fatigue at the liquid-vapor interface which will fluctuate at a given load and shift nominal level with change in load. B&W said they had looked into this and calculated that interface stresses will not exceed 20,000 psi. They stated Inconel 600 can withstand an infinite number of cycles at 30,000 psi and therefore no problem exists.

10. 19-Tube Model Tests - When we noted the treatment in the report was very sketchy, B&W said these tests were just getting started when the topical was prepared. They said much new data has been obtained from these tests. Further, a second 19-tube model has been constructed and they expect to obtain additional information from both of these models plus a short-section model which is being used for materials tests.

11. Stability Studies - We expressed concern on whether they had objectively predicted instabilities and subsequently confirmed predictions by the tests. Apparently they gave an outside consultant the necessary information to simulate S-G performance without telling him the nature of the test results and thereby obtained a form of objective confirmation. We said we would need some assurance that they have a handle on the causes of these instabilities and a basis for confidence in their ability to predict its exclusion in the full scale production units for Oconee.

INSTRUMENTATION

1. Seismic Design Considerations - B&W "plucked" the reactor protection cabinets and determined that resonant frequency is above the seismic range as given to them by Duke. B&W modules will be shaker tested "live" over range 1-20 Hz@1g and over range 0.1-1-G@4Hz. ITE switchgear has been tested for others through 3g loading with contacts being monitored. The various

process instrument sensors will also be tested "live". A pressure transmitter has been tested over range 1-20 Hz@ 10g for 2 1/2 hours and exposed to 100 Hz @ 10g. It appeared from the discussions that demonstration of seismic capabilities are being required for instrumentation.

2. Quality Assurance - Mr. E. Patterson of B&W is a member of the standards group that developed IEEE-279 and assured us that all B&W supplied items would meet that document's QA requirements. As evidence of how QA is handled on a specific instrument they selected their bistable unit which is used universally throughout the B&W instrumentation. A Bailey Meter Company production unit was examined. B&W identified their design documents which included "General Design Specs for Instrumentation C-92-18." A Fault analysis is performed as well as the expected performance tests including environmental and transient exposure. The B&W QA representative for Mechanical Systems, C. Fletcher, stated Bailey Meter QA is personally audited by him quarterly. Jim Wells of Duke explained how the equipment is recieved and stored on site and Ollie Bradham explained that it would be installed and tested out in accordance with written procedures.

3. Wire Run Separation & Fire Protection - These items were discussed in some detail for the several types of cables (power, control, and instrumentation). Typical specimens were shown. All power cables will have interlocked armor which Duke considers equivalent to a conduit for fire protection purposes. In general, it appears that Duke is executing acceptable designs in this area and should have no trouble in documenting this in the scope we feel necessary.

4. Control Room & Equipment Room Environment - Duke believes it incredible that air-conditioning to these areas will be lost, except momentarily, due to standby capacity available. However, they have looked at this and conclude they could operate continuously at 110° F with an R.H. of 80% or for 24 hours at an R.H. of 90%. They have also looked at loss of fans in the cabinets and found only one critical component, a power supply which, at nominal voltage would not overheat at 110° F ambient for 1 1/2 hours. It also appeared that means are available, through a reduced power bill, to prevent the room temperature from exceeding 110° F. It was brought out that the printed circuits are not coated (probably a good thing) and that administrative procedures are available to preclude cold surface condensation (sweating) on circuit boards. Critical high impedance circuit elements have been encapsulated. B&W had not tested their equipment under dew point conditions. During this discussion we learned that air-conditioning ducting is to be installed by a contractor, not Duke personnel.

5. Safety Circuit Identification - Duke has an extensive color control scheme that will be applied to all safety equipment, switch gear, MCC's, etc. to signify one of several redundant "chains" or independent circuits. All cables between such items will have exteriors (sheathes or armor) of the appropriate color. Personnel will be trained to recognize these identifications and their significance. "Green" personnel will not be permitted to work on the systems.

6. Plant Shutdown Outside Control Room - Duke will have ability to shut down to hot standby outside the control room with no prior action before leaving the control room. We learned that an auxiliary shutdown panel will be provided for each unit (by B&W at Duke's specific request per B&W) somewhere outside the control room(s). Duke has not yet decided how this panel will be protected from unauthorized use. We did not learn what has been added to this auxiliary panel or how it meets "safety grade" requirements.

7. Emergency Lighting and Offsite Communications - We learned that there are three independent emergency lighting systems; one dc system and two ac systems. We explained outside communications were of concern from the standpoint of alerting the outside world of an emergency within a unit. Duke uses a PBX with a 1-hour battery backup which connects directly to their microwave system. There is also a non-dial microwave circuit from each control room tying the plant to offices in Charlotte and Spartansburg. Both systems are ac powered. The direct link microwave system has an 8 hour capacity battery backup further backed up by a propane gas generator with a 7 day fuel supply. Also each unit has a transmitter/receiver operating at 49.98 and 47.84 Mega Hertz with sufficient range to reach central, a transmission substation. Duke plans to purchase one or more cars and a boat with this same X/R equipment. The boat and car will also have a powered bull-horn to warn personnel within the exclusion radius of a need to evacuate.

8. Core Injection Bypass - B&W stated the three ESF reactor coolant sensors are independent of the four RPS reactor trip sensors shown on FSAR Figure 7-1. Figure 7-17 shows that the RPS and ESF pressure sensors are positioned in each of the hot legs to give further separation. B&W is preparing a revision to Figure 7-3. Using the block diagram prepared for this revision, B&W showed that there are 3 LP and 3 HP bypass switches. The purpose of these bypass switches is to permit normal cooldown and heatup of the reactor coolant system without activating the HP or LP injection systems. From a discussion of the HP system it was apparent that B&W did not intend for the operator to be able to readily remove the bypass below 1900 psi once initiated. They stated, however, they were sure the bypass could be removed below 1900 psig "some how" if it were necessary to do so.

Our concern was that the bypass would be removed at 1900 psig, well above the 1500 psig trip point for activating the HP injection system. B&W explained that the wide range pressure instruments used for this purpose require a substantial margin above the set point to ensure prevention of HP injection system actuation. While we could agree to some margin being necessary, it was not clear that a 400 psi margin is required or that the diverse containment pressure channels will be tested for proper operation prior to bypass. B&W made the point that rods could not be pulled until the reactor coolant system was at 530° F and nominal operating pressure (2200 psig).

9. Instrument Status Indicators - B&W noted that trip status of all protective channels are displayed to the operator. Bypasses are at least indicated by lighted switches. There remains some doubt in our minds as to what indications are available to the operator to warn him that one or more channels have administratively been put in a test or maintenance condition and then supplied with an intentional "cheater circuit" which defeats the protective function of the channel until removed. The applicant claims that only one channel at a time (of 4 or of 3 as applicable) would be so compromised for test or maintenance.

10. Emergency Safety Feature Control Circuits - B&W clarified for us that no ESF trip or RPS trip can be automatically cleared. Once tripped, reset must be manual. Our concern arose in connection with the control circuit for solenoid operated valves appearing on Figure 7-4.

11. Rod Drive Control - B&W stated the feature that limited control rod drive speed to 30 in./minute was the use of synchronous programmer motors locked to the plant's 60 Hz frequency. Our subsequent examination of the FSAR indicates there may be an intervening static inverter supplying this "plant frequency." We intend to pursue this further in our formal questions to Duke. We understand there is also a slow speed programmer motor intended to be used only for jogging single rods. Based on the limited discussion at the meeting, it is expected that we will ask to see additional circuit details on the rod drive control system.

12. Testing of Reactor Building Spray Actuation Pressure Switches - Duke indicated that these switches would be periodically tested by applying a calibrated test pressure to each switch. The switches are inside containment, but will be accessible during reactor operation. (Duke plans to go inside containment routinely about once per shift to take readings,etc.)

13. Integrated Control System (ICS) - During our review of the FSAR we noted that, for the emergency feedwater system to be lined up, the ICS had to sequentially open and close several valves. B&W said they had checked this out as well as completely reexamining the ICS and are satisfied that it is not required to operate (even for the emergency feedwater system)

in order for any RPS or ESF system to meet required performance. For the emergency feedwater system they said all valves could be lined up manually if ICS failed and that there is ample time to do this (23 minutes per FSAR paragraph 14.1.2.8.3).

14. Neutron Detectors - B&W plans to use all Westinghouse detectors for the RPS channels. Source range detectors will use a B^{10} lining (instead of BF_3 gas). The intermediate range detectors will be standard Westinghouse design CIC's. The power range detectors will be UCIC's. B&W has changed from three 4-foot sections in one can to two 6-foot sections in one can. They have apparently retained the ability to monitor flux in each section (for use in detecting abnormal flux patterns).

15. Design of Reactor Protection Circuits - Maximum use is made of integrated circuits and solid state components, except several mechanical relays are employed. Selected Type 709 IC operational amplifiers are standard. Bailey Meter is supplying all B&W RPS instruments.

16. HP and LP Injection (With and Without Reactor Trip) - We carefully pointed out we were not taking a position, but were seeking information on whether B&W's LOCA analyses must take credit for a reactor trip. B&W stated that they take credit for a reactor trip in their analyses for small size breaks. They said they have not analyzed LOCA consequences without taking credit for a reactor trip, since redundant reactor low pressure trip channels are available (separate from the HP and LP injection low pressure trip channels) to trip the reactor.

CONDUCT OF OPERATION AND INITIAL TESTS

1. Single-Unit - We stated that, unless there is clear evidence to the contrary, we believe Oconee Unit 1 should have five men per shift including three licensed operators. After reaching commercial operation Duke might be able to justify elimination of one unlicensed operator except for startups and scheduled shutdowns.

Duke indicated that this was near their estimate of requirements, however they suggested that shift size be reduced upon the completion of the appropriate power testing at a given power level rather than "commercial operation." This would allow the use of a four-man shift at a power level less than 100% provided the testing at that level had been completed and some restraint prevented further escalation to full rated output with a four man crew. We agreed this approach seemed reasonable provided the exact details could be finalized prior to approval of technical specifications.

Duke intends to have at least one man per shift qualified to perform the duties of a Health Physics Technician.

2. Dual-Unit - Duke made a short presentation stating that Oconee 1 & 2 controls had been laid out in such an arrangement that four men per shift could effectively control both units from a common control room. Such a practice was common in their newer dual-unit coal-fired stations now in operation.

Our position was that perhaps as many as eight men per shift could be required unless justification for less could be provided. Duke had not carefully analyzed their manpower requirements for the worst case situation and was not prepared to justify their four man proposal. They informed us that they intend to carefully examine shift manpower requirements and will want to discuss this with us at a later date. This must be resolved in a timely fashion to insure selection and training of the necessary personnel for operation of Unit 2.

During the general discussion regarding dual-unit operation, Duke stated their intent to have one operator monitor the operation of both units if the remainder of the crew was occupied with other duties outside of the control room. We indicated such operation would be unacceptable. (This subject must be addressed specifically in the technical specifications.)

3. Cross-Licensing of Operators - Duke intends that all licensed personnel will hold licenses for all three units at the Oconee Station. This arrangement will allow the maximum flexibility in shift staffing.

4. Emergency Plan - A general description of the overall emergency plan was given by the applicant. A preliminary copy of the plan was left with us. Duke was advised to expect a request for formal submittal of the plan including medical preparedness arrangements.

5. Industrial Security - The applicant outlined the security measures to be taken during operation of Unit 1 while the remaining units are still under construction. These measures will include a perimeter fence with gate guards, closed circuit TV for back-shift gate monitoring and administrative control for operating equipment. We said that Duke should evaluate the sensitive areas of their plant and provide suitable security measures to prevent simple acts of sabotage. We discussed the general scope of the evaluation required. Duke inquired and we put off for later resolution how this evaluation would be reviewed by the regulatory staff. We did say, however, that it would be reviewed. The point of concern, obviously, was minimizing potential compromise of sensitive data.

6. Startup Organization - Duke expects ten men to be "cold licensed" for initial plant startup. Of these ten men, two will be management personnel, leaving eight licensed operators to supervise plant operations until

additional operators are obtained. Duke is planning on these men working a seven day week 12 hours on, 12 hours off during this period. B&W will not supply startup operating personnel but will provide eight to twelve people to assist in test evaluation and technical assistant. Supervision and performance of all test work will be done by Duke.

7. Initial Testing and Power Escalation - We told Duke we believe they should provide the following information for each test which could have safety significance: (a) test objective, (b) plant prerequisites, (c) general description of method including major steps, and (d) acceptance criteria with allowable deviations. Duke said this information would be available prior to testing; however, they objected to a formal submittal because it would be voluminous and subject to change. Duke offered to prepare the above information for a representative test for discussion at a later meeting at which time they would also identify the safety areas believed significant.

A. Schwencer

A. Schwencer
Reactor Projects Branch No. 3
Division of Reactor Licensing

Attachment:
List of Attendees

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

ATTENDANCE LIST DUKE OCONEE MEETINGS
DOCKET NOS. 50-269, 50-270 and 50-287
November 18 - 19, 1969

AEC	DUKE POWER COMPANY	B & W
C. Long	A. Thies	W. Smith
A. Schwencer	P. Barton	D. Montgomery
D. Ross	C. Wylie	J. Mallay
*B. Cady (19)	W. Owen	R. Craig
*T. Novak (18, 19)	J. Smith	*C. Fletcher (18)
*O. Parr (18)	R. Wells	*E. Patterson (18)
*R. Pollard (18, 19)	W. Parker	*J. McCreary (18)
*K. Wichman (19)	C. Price	*R. Abbot (18)
*J. Knight (19)	B. Rice	*T. Schuler (18)
*M. Dunnenfeld (19)	W. Foley	*G. Snyder (19)
*H. Richings (19)	J. Elliott	*B. McDonald (19)
*R. Waterfield (18)	J. Hall	*R. Turner (19)
*J. McGough (19)	L. Lewis	*J. Taylor (19)
*J. Buzy (19)	K. Canady	**C. Russell (18)
*P. Collins (18, 19)	D. French	**W. Brunson (18)
	O. Bradham	**R. Bybee (18)
	J. Hampton	**N. Hennessy (19)
	S. Nabow	
	C. Sansbury	
	T. Wyke	
	H. Lark	

* Part time (date participated)
** Part time - observers only (date present)