D06/30/78

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SUBJECT:
FORWARDING RESPONSE TO NRC QUESTIONS CONCERNING STARTUP TESTING WHICH WERE RAISED TO APPLICANT"S LTR DTD 05/30/78 RE PROPOSED LIC AMEND TO SUPPORT OPERATION OF UNIT 3 DURING CYCLE 4.

PLANT NAME: OCONEE - UNIT 1
OCONEE - UNIT 2
OCONEE - UNIT 3

REVIEWER INITIAL: XJM
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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

June 23, 1978

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. Reid, Chief
Operating Reactors Branch #4

Reference: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

My letter of May 30, 1978 submitted a proposed license amendment to support the operation of Oconee Unit 3 during Cycle 4. In discussions with the staff, several questions concerning startup testing were raised. Please find attached responses to these questions.

Also, please note that the responses that pertain to specific physics tests are applicable not only for Oconee Unit 3, but also for Units 1 and 2.

Very truly yours,

William O. Parker, Jr.

RLG:scs
Attachment

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RESPONSES TO QUESTIONS
ON OCONEE 3, CYCLE 4 RELOAD

1. Describe in detail the tests being done to check for a misloaded assembly. What assurances are there that the core is as expected before going to powers >5% rated power?

RESPONSE

Following completion of refueling and prior to installation of the plenum and reactor vessel head, a visual inspection of the core is performed to verify the core loading is correct. This is accomplished by using under water camera equipment to read individual fuel assembly ID numbers and determine control component types, if any, which are located in the fuel assemblies. During the visual examination, the individual performing the test fills in a blank core map with the fuel assembly ID's and component types which he observed. After completion of this visual exam, the map constructed by the individual performing this check is compared with the designed core loading map to assure that the shuffle has resulted in the core configuration intended by the designer. In addition to this visual verification of the core loading, numerous tests are performed during the zero power phase of startup physics testing which are intended to assure that the core is behaving as expected before going above 5% of rated power. The performance of regulating rods out boron concentration, rod worth, boron worth and ejected rod worth tests and comparison of the results of these tests with predicted values should indicate any anomalous core conditions.

2. Provide the details of the procedures for the critical boron concentration tests. Discuss how corrections are made to the measured data and how the measured data is compared to the predictions. What are the acceptance criteria and what are the procedures if the acceptance criteria are not met?

RESPONSE

Execution of the all regulating rods out boron concentration test is performed by borating regulating rod groups 5 and 6 to 100% withdrawn and regulating rod group 7 to between 93 and 100% withdrawn. After equilibrium boron conditions are achieved, group 7 is withdrawn to 100% withdrawn position and the resulting reactivity change as calculated by the plant computer reactivity calculation is recorded. The all rods out boron concentration is then calculated by summing the equilibrium boron concentration and the boron worth equivalent to the group 7 withdrawal. The acceptance criterion for this test is that the adjusted all rods out boron concentration be equal to the calculated all rods out boron concentration plus or minus 100 ppm boron. If this acceptance criterion was not met, data and boron samples would be rechecked for any errors and an evaluation would be performed by Duke Power personnel and by B&W personnel, if appropriate, to determine what further actions or testing might be necessary to resolve this discrepancy. Further testing or escalation to power will depend on the nature and extent of the discrepancy in the test results.

3. Describe the procedures and methods used for the temperature reactivity coefficient tests. Also provide the acceptance criterion and the procedures to be followed if the acceptance criterion is not met.

RESPONSE

The temperature coefficient of reactivity test begins with the reactor at equilibrium critical conditions. The test is then performed by executing a change in reactor coolant average temperature of either plus or minus 5 degrees and establishing the reactor at the upper or lower temperature plateau while data is taken. The change in reactivity associated with this maneuver is compensated for by control rod movement. After data is taken at the first temperature plateau, reactor coolant temperature is changed to the opposite plateau, either 5 degrees above or below the nominal average coolant temperature. Changes in reactivity associated with this temperature transient from the first or second temperature plateaus are recorded by the reactivity calculation. The overall temperature coefficient is then calculated by dividing the change in reactivity between the first and second temperature plateaus by the change in temperature between the first and second temperature plateaus. The acceptance criterion on the measured temperature coefficient is that it shall agree with the predicted value within a tolerance of $\pm$ 0.4 x 10^{-4} $(\Delta k/k)/^{o}F$. The moderator coefficient of reactivity is obtained by subtracting the predicted isothermal Doppler coefficient from the measured temperature coefficient. The acceptance criterion for this test is that the maximum positive moderator coefficient shall not be in excess of + 0.5 x 10^{-4} $(\Delta k/k)/^{o}F$. If the acceptance criterion was not met, the test results would be reviewed by Duke Power personnel and by B&W personnel, if this were deemed appropriate, to determine what further action should be taken and whether further testing might be required to resolve this discrepancy. Further testing or escalation to power will depend on the nature and extent of the discrepancy in the test results.

4. Provide the details of the regulating control rod group reactivity worth tests. Give the predicted worth of each group to be measured and the stuck rod worth and the predicted total worth for all rods. State which groups will be measured and provide the acceptance criteria and the actions to be taken if the acceptance criteria are not met.

RESPONSE

The measurements of regulating group rod worths begin from a critical steady-state condition with all regulating groups withdrawn as far as possible (i.e., Group 7 between 93% and 100% withdrawn) from this point a boron concentration necessary to deborate control rod groups 7 and 6 to 0% withdrawn and group 5 to approximately 10% withdrawn is calculated. The boration is commenced, and chemistry sampling is initiated on a thirty minute frequency. The resulting reactivity change during deboration is compensated for by discrete insertion of control rods at least every 300 $\mu\rho$ with these reactivity insertions being recorded by the reactimeter calculation. Differential rod worths for these insertions are then calculated by dividing the difference in reactivity for each insertion by the difference in control rod position, and integral worths are calculated by summing the differential worths for each group.

Acceptance criteria for the rod worth measurements require that the individual rod group worths deviate from the predicted group worths by no more than ± 15% and that the total worth of rod groups 5, 6 and 7 deviate from the predicted total worth of these three groups by no more than ± 10%. If these criteria were not met, an evaluation would be performed by Duke Power personnal and by B&W personnel, if this were deemed appropriate, to determine what actions should be taken and/or what additional testing might be required to resolve this discrepancy. Further testing or escalation to power will depend on the nature and extent of the discrepancy in the test results. The predicted worths for control rod groups to be measured, the stuck rod worth and the predicted total worth for all rods is listed below:

Control Rods	Predicted Worth @ Hot Zero Power %Δk/k
Group 5	1.36
Group 6	0.85
Group 7	0.81
Stuck Rod	1.80
Total 1-7	8.18
Total 1-8	8.50

5. Describe the procedures for the ejected control rod reactivity worth test. State which rods will be measured. State the methods used to compare the measurements with predictions and the acceptance criteria. Also, include procedures if the acceptance criteria are not met.

RESPONSE

Ejected rod worth measurements are performed for one or more control rods predicted to have the highest ejected worth. The measurement of ejected control rod reactivity worth begins with the plant in the critical condition with rod groups 7 and 6 fully inserted and group 5 approximately 10% withdrawn. From this point, the boron concentration change required to withdraw the worst case ejected rod to 100% withdrawn is calculated. Boration is commenced and boron samples are taken for chemical analysis once every hour. The resulting reactivity change due to boration is compensated for by discrete withdrawal of the ejected rod. The reactivity corresponding to each of these withdrawal steps is calculated by the reactimeter calculation and is recorded along with the position of the ejected rod. As with the group rod worth measurements, the worth of the ejected rod is calculated by summing the worths measured for each of the discrete withdrawal steps. Next, the worth of the predicted worst case ejected rod will be determined by swapping the rod back into the core against group 5. This means that the ejected rod would be inserted into the core and control rod group 5 would be withdrawn to compensate for the change in reactivity. The initial and final group 5 positions would be recorded, and using the integral rod worth curve, the reactivity corresponding to the change in group 5 position would be the worth of the ejected rod. The ejected rod worths determined by the two methods above are each corrected for the position of control rod group 5 during each of the measurements. These rod worth values are then error adjusted for measurement uncertainties and inserted worth of control rod group uncertainties and compared to the safety analysis limit. The acceptance criteria for this test states that the worst case ejected rod worth must not deviate from the predicted value by more than $\pm$ 20% of the measured value and that the final error adjusted worst case ejected rod worth cannot exceed 1.0% $\Delta k/k$. If these acceptance criteria were not met, an evaluation would be performed by Duke Power personnel and by B&W personnel, if appropriate, to determine what further action and/or testing might be required to resolve this discrepancy. Further testing or escalation to power will depend on the nature and extent of the discrepancy in the test results.

6. Oconee 3 had a quadrant tilt at the beginning of Cycle 3. How did this tilt change during the cycle? How was the presence of this tilt used in the predictions of the power distributions for Cycle 4?

RESPONSE

The beginning of Cycle 3 quadrant power tilt of Oconee 3 was a result of an unlatched group 8 control rod. Recoupling of the rod reduced the measured tilt to within the Technical Specification limit. The indicated tilt at full power during the beginning-of-Cycle 3 was approximately +1.60%, the end-of-cycle value at 154 EFPD was less than +1%.

The Cycle 3 tilts were not included in the prediction of Cycle 4 power distributions; however, the shuffle pattern for Cycle 4 was designed to minimize the effects of any power tilts present in Cycle 3.

7. Provide the details of the core power distribution tests. Describe in detail the methods used to predict the assembly-by-assembly power as well as the analyses of the data obtained during the measurements. What are the assembly-by-assembly acceptance criteria? How are tilts accounted for in the anslysis of the data? If a ¼ or 1/8 core map is the result of the measurement, what method is used to determine the assembly power for those assemblies having their symmetric assemblies instrumented? For example, are the measured assembly powers averaged, or is only one of the symmetric measurements used?

RESPONSE

Core power distribution data will be obtained at various power levels and compared with predicted data to assure compliance with operating limits and Technical Specifications. Power imbalance, quadrant power tilt, linear heat rate, DNBR, and power peaking factors will be analyzed. For this test, 40% will mean 40% $\pm$ 2%FP, 75% will mean 75% $\pm$ 2%FP and 100% will mean highest attainable power without exceeding 100%FP.

Equilibrium xenon will not be required for the 40% tests. Control rod index is established at a position corresponding to the rod positions where core power distribution predictions were calculated.

The acceptance criteria are as follows:

(i) The maximum linear heat rate in the core is less than the LOCA limit per Technical Specifications for the axial location of the peak. When testing at a power level below rated power, the maximum LHR when extrapolated to rated power must also meet this criteria.

(ii) The minimum DNBR must be greater than 1.30 at rated power conditions and when extrapolated to rated power conditions from a lower test plateau.

(iii) The quadrant power tilt must not exceed the value allowed in the Technical Specifications.

(iv) The highest measured radial and total peaking factors shall not exceed the highest predicted peaks by more than 5% and 7.5% of the measured peaking factors, respectively, at the 75% and 100% power plateaus.

(v) The highest measured radial and total peaking factors shall not exceed the highest predicted peaks by more than 8% and 12% of the measured peaking factor, respectively, at the 40% power plateau.

These acceptance criteria are established to verify that core nuclear thermal hydraulic calculational models are conservative with respect to measured conditions thereby verifying the acceptability of data from these models for input to safety analysis. The acceptance criteria also serve to verify safe operating conditions at each test plateau and eventually at rated power conditions.

Predictions for the radial and total peaks at 40, 75, and 100% FP are calculated using the FLAME-3 with thermal-hydraulic feedback code (BAW-10124). Radial peaks are calculated from the predicted power output for each assembly in a 1/8 core. Total peaks are calculated from the predicted power output of the maximum segment for each assembly in a 1/8 core.

Assembly and segment power representations are calculated by the on-line computer based on current signal outputs from the 52 incore detector strings. Any tilt which exists in the core is inherent in the measurement of neutron flux by the incore detector system. Only instrumented assemblies are utilized in the analysis of the data to calculate measured radial and total peaks for comparison to predicted radial and total peaks. Symmetric instrumented locations are averaged to provide a single value for the assembly or segment power in the 1/8 core location. Radial and total peaks are then calculated. As previously stated, the maximum measured radial and total peaks are compared to maximum predicted radial and total peaks. There are no criteria for comparisons on an assembly-by-assembly basis.

Tilt effects are accounted for in the calculation of DNBR and linear heat rate. If a significant tilt does exist, a routine in the on-line computer adjusts the segment power representations of an instrumented assembly in order to provide segment power representations of a symmetric, non-instrumented assembly. DNBR and linear heat rate are calculated by the on-line computer for the maximum assembly. These values are then compared to acceptance criteria previously discussed. In addition, a hand calculation of linear heat rate is performed in order to obtain values for comparison with LOCA acceptance criteria which are level dependent.

8. Provide details of the power Doppler reactivity coefficient and temperature reactivity coefficient measurements near full power. What methods are used to compare measured values with predictions? What are the acceptance criteria for these tests and what procedures are followed if acceptance criteria are not met?

RESPONSE

The temperature coefficient of reactivity and power Doppler coefficient reactivity measurements at power are performed at a plateau near full power with core xenon in equilibrium at the start of the test. In order to measure the temperature coefficient of reactivity, control rod group worth measurements are performed using a fast insert-withdraw method at each separate temperature plateau during the test. These worths are then used to calculate the changes in reactivity accompanying the changes in average reactor coolant system temperature. The test begins with the reactor coolant system average temperature at approximately 579^{o}F. As the first step in the test, steady-state data and control rod group measurements are taken using fast insert-withdrawal method. Next, RC average temperature is decreased by approximately 5^{o}F. Data is taken at this lower temperature plateau, including control rod worth measurements, then RC average temperature is increased by about 5^{o}F and steady-state data and more control rod group worth measurements are performed at this final plateau. Using the data from the rod group worth measurements, the reactivity change associated with the temperature changes which have been performed is calculated and the reactivity values are divided by the changes in temperature between adjacent plateaus to obtain values for the temperature coefficient of reactivity. The moderator temperature coefficient of reactivity is obtained by subtracting the predicted full power Doppler coefficient from the measured temperature coefficient. The acceptance criterion for this test is that the moderator temperature coefficient shall not be positive at power levels above 95% full power.

The power Doppler coefficient of reactivity measurement is performed in the same manner as the temperature coefficient measurement above, except that reactor power changes are used instead of reactor coolant changes. The test begins with core xenon in equilibrium and reactor conditions steady at the power testing plateau. Steady-state data and control rod worth measurements using the rapid insert-withdrawal method are taken at this initial plateau. Next, reactor power is decreased by approximately 5% and data is taken at the lower plateau. Then reactor power is increased by approximately 5% and further data is taken at this final plateau. As with the temperature coefficient of reactivity measurement, reactivity changes corresponding to the changes in state between adjacent plateaus are calculated using the data from the control rod worth measurements taken at the individual plateaus. The $\mu\rho$ associated with the changes between adjacent plateaus are then divided by the change in power between adjacent plateaus to yield the value of the power Doppler coefficient of reactivity. The reactivity changes used in these calculations are corrected for changes in core xenon worth and any changes in reactor coolant system temperature which may have occurred during the testing. The acceptance criterion for this test is that the power Doppler coefficient of reactivity shall be more negative than -0.55×10^{-4} $(\Delta k/k)/\%FP$.

If the acceptance criteria for either of these tests were not met, an evaluation of test results would be performed by Duke Power Company personnel and by B&W personnel, if this was deemed appropriate, to determine what action should be taken and/or what other testing should be performed to resolve this discrepancy.

9. Discuss the changes to be made to the plant's computer prior to Cycle 4 operation. This should describe:

   a. What elements change from Cycle 3 to Cycle 4 coefficients, constants, correlations, etc., and why they change.

   b. What codes and methods are used to establish the new values?

   c. What quality assurance procedures (testing) are used at the site to verify that changes have been correctly made?

RESPONSE

The following elements in the plant's computer software will be changed prior to Cycle 4 operation due to the loading of a new fuel enrichment and the shuffling of the remaining fuel assemblies:

1. Signal to power conversion factors for the fuel.

2. Boron, xenon, and fuel and Samarium worths for the core.

3. Core macroscopic fission and xenon absorption cross sections and xenon and iodine yields.

4. Isotopic concentrations versus burnup for the fresh fuel.

5. Constants for new incore detectors

6. Integrated quantities related to cycle history are initialized and shuffled to account for changes in the core loading.

Items 1 through 4 above were calculated using the same PDQ models used in designing and analyzing Cycle 4. Item 5 is based on as-built incore detector data and experimentally determined factors.

All Cycle 4 software changes are implemented while the reactor is in cold shutdown. The as-implemented software is tested with test case data which has been previously run on an independent off-line computer system which has the same software implemented. The outputs of the test cases run on the plant computer and on the independent off-line computer are compared for agreement.

10. A brief summary report of physics startup tests is to be submitted to the NRC within 45 days of completion of the startup tests. This report should include both measured and predicted values. If the difference between the measured and predicted value exceeded the acceptance criterion, the report should discuss the actions that were taken and justify the adequacy of these actions. Please state that such a report will be submitted.

RESPONSE

As in the past, a summary report of physics startup tests will be submitted upon completion of the tests. However, the data obtained during the startup testing is available on site for NRC review upon completion of the startup testing program.

Docket 50-269

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

June 21, 1978

All Power Reactor Licensees

Gentlemen:

SUBJECT: REVISIONS TO INTRUSION DETECTION SYSTEMS AND ENTRY CONTROL HANDBOOKS AND NUCLEAR SAFEGUARDS TECHNOLOGY HANDBOOK

Enclosed is a copy of the Nuclear Safeguards Technology Handbook which was prepared under contract for the Department of Energy (DOE). The purpose of this handbook is to convey an understanding of the current SS safeguards technology development program and its prospective relevance and use to U.S. industrial and utility organizations, as well as to other U.S. government agencies and international organizations.

Also enclosed are updates to the "Entry-Control Systems Handbook" and the "Intrusion Detection Systems Handbook" that were sent to you earlier.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosures:
As stated

cc w/o enclosures:
Service List