ATTACHMENT 2

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

GENERIC STARTUP PHYSICS TEST PROGRAM

March 1983

(For clarity, the entire Test Program is provided. The changed pages are noted by a line on the right hand margin)

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

STARTUP PHYSICS TEST PROGRAM

The Startup Physics Test Program for Oconee Nuclear Station is structured to provide assurance that the installed reactor core conforms to the design core. This document provides the minimum test program which will be conducted on each Oconee unit. Additional tests may be performed during a specific startup test program as conditions warrant. However, in all cases, the following tests will be performed:

1. Pre-critical Test Phase
   a. Rod Drop Time

2. Zero-Power Physics Test Phase
   a. Critical Boron Concentration
   b. Moderator Temperature Coefficient
   c. Control Rod Worth

3. Power Escalation Test Phase
   a. Core Symmetry
   b. Low Power Core Mapping
   c. Full Power Core Mapping

In addition to the above tests which comprise the basic Startup Physics Test Program, a separate test, the reactivity anomaly check, is performed approximately each 10 EFPD, during steady-state operation, at equilibrium conditions pursuant to Technical Specification 4.10. This procedure is used to verify that the measured all-rods-out hot full power critical boron concentration is in agreement with the predicted value. The test conditions, procedure descriptions, acceptance criteria and review requirements for each of the above are provided in this document.

For all of these tests, specific acceptance criteria are provided. Upon completion of each test, the results are reviewed by a designated individual. If the results meet the specific acceptance criteria, then the test is considered to be satisfactorily completed. However, if the results exceed the specific acceptance criteria, an extensive review is performed by cognizant engineers from within Duke Power or from outside organizations, as appropriate, to identify and correct the cause of the discrepancy. Continuation of the test program, including any power escalations, will be dependent upon satisfactory resolution of any unacceptable test result. The Technical Review Committee[1] will approve actions under the conditions stated for each test.

[1]This Committee will consist of Representatives from Oconee Performance, Oconee Licensing, and General Office Nuclear Fuels.

ROD DROP TIME

CONDITIONS:

HSD, 532°F, 2155 psig, full reactor coolant flow.

PROCEDURE:

The rod drop time for each full-length control rod assembly (CRA) to fall from the fully withdrawn position to the 75% inserted position is measured. The plant process computer is used to record the time interval between initiation and termination of the event. The test may be performed either by dropping all full-length CRA's simultaneously from the fully withdrawn position, or by dropping one full length CRA group at a time and measuring the drop times for each individual group. In either case, the computer records the drop time of each CRA individually.

The accuracy of the measurement of rod drop time is performed by the computer and is approximately ± 0.005 sec.

The results are reviewed by the Test Coordinator and compared with the acceptance criteria, 1.40 sec.

FOLLOW-UP ACTIONS:

If any rod drop time exceeds 1.66 sec., then the results will be reviewed by cognizant design engineers to determine the appropriate corrective actions, and the actions specified by Technical Specifications 3.5 and 4.7 will be taken.

If any measured rod drop time is greater than the acceptance criteria, then the results will be reviewed by cognizant design engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed prior to 100% FP.

CRITICAL BORON CONCENTRATION

CONDITIONS:

HZP, 532°F, 2155 psig, full reactor coolant flow.

PROCEDURE:

Critical boron concentration measurement is taken at the "all-rods-out" (ARO) configuration.

The ARO critical boron concentration is measured by establishing an equilibrium RCS boron concentration at or slightly greater than the predicted ARO critical boron concentration. Control Rod Groups 1-7 are fully withdrawn to perform the Rod Drop Time Test. This rod withdrawal is also the initial approach to criticality. Control Rod Group 8 is maintained at the nominal design position but may be moved, if necessary, for reactivity control and to allow the Rod Drop Test to be performed. Based on the $\frac{1}{M}$ data or critical rod positions (if criticality is achieved), the boron concentration is adjusted to establish criticality at the ARO condition with subsequent rod withdrawal. A sample of the equilibrium boron concentration is then taken and analyzed to determine the critical boron concentration. Since it may not be practical to establish equilibrium critical conditions with Group 7 fully withdrawn, the small amount of inserted worth of Group 7 or worth of Group 8 (from its nominal design position) is measured by a reactivity calculation. This reactivity is then used to adjust the boron concentration to obtain the measured ARO boron concentration.

The uncertainty associated with these measurements is less than 20 ppm B.

The results are reviewed by the Test Coordinator and compared with the predicted boron concentrations. If the difference between the measured and predicted values does not exceed 50 ppm B, the results are acceptable.

FOLLOW-UP ACTIONS

If the difference between measured and predicted critical boron concentration is greater then 100 ppm B, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed, and the results and recommended actions approved by the Technical Review Committee prior to exceeding 5% FP.

If the acceptance criteria is not met, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the results and recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

MODERATOR TEMPERATURE COEFFICIENT

CONDITIONS:

HZP, 532°F, 2155 psig, full reactor coolant flow.

PROCEDURE:

The moderator temperature coefficient (MTC) test begins with the reactor at equilibrium critical conditions. The test is 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, by executing a 10°F temperature ramp from the first plateau to the second. 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. This overall temperature coefficient is corrected for the contribution of the isothermal doppler coefficient of reactivity to give the moderator coefficient of reactivity.

The Reactor Coolant System's average temperature values are obtained by taking the average of hot and cold leg RTD readings. The hold time at each temperature plateau during the test is approximately five minutes.

The measurement uncertainty associated with this measured value varies as a function of the magnitude of the temperature coefficient itself. In all cases within or near the acceptable range of temperature coefficient values, the error is less than $\pm 6.0 \times 10^{-6}$ $\Delta k/k/°F$.

The results are reviewed by the Test Coordinator and compared with the predicted MTC. If the difference between the measured and predicted values does not exceed $.30 \times 10^{-4}$ $\Delta k/k/°F$, then the results are acceptable.

FOLLOW-UP ACTIONS

If the measured maximum positive MTC exceeds 0.5×10^{-4} $\Delta k/k/°F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the results and recommended actions approved by the Technical Review Committee prior to exceeding 5% FP.

If the acceptance criteria is exceeded, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the results as well as recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

CONTROL ROD WORTH

CONDITIONS:

HZP, 532°F, 2155 psig, full reactor coolant flow.

PROCEDURE:

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 deboration 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 in steps of approximately 600 μρ 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.

The results are reviewed by the Test Coordinator and compared with the predicted group worths. If the difference between the measured and predicted individual rod group worths does not exceed 15%, and the difference between the measured and predicted total worth of rod Groups 5, 6 and 7 does not exceed 10%, then the results are acceptable.

FOLLOW-UP ACTIONS:

If the difference between the measured and predicted total worth of rod groups exceeds 10%, then, following calculation of the minimum rod position for which the worth of the control rods withdrawn would equal 1% $\Delta K/k$, additional rod group worths will be measured. The worths of safety rod groups will be measured in sequence from Group 4 to Group 1, until either the difference between the measured and predicted total worth of all rod groups measured does not exceed 10%, or the minimum rod position calculated above is reached in which case additional testing will be performed. The results will be reviewed by cognizant engineers to determine the appropriate additional corrective actions required to resolve the discrepancy. This review will be completed and the results as well as the recommended actions approved by the Technical Review Committee prior to exceeding 5% FP.

If the difference between the measured and predicted individual rod groups worths exceeds 15%, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed prior to reaching 100% FP.

CORE SYMMETRY

CONDITIONS:

5 to 20% FP, full reactor coolant flow.

PROCEDURE:

Once the unit is between 5 and 20% FP, the output of the plant process computer reactor calculations program is analyzed. This program processes the signals from fixed incore detectors and provides a relative core power distribution as output. The incore detector outputs are checked in order to identify malfunctioning detectors. After these have been eliminated, the results of the power tilt calculation are compared with the Error Adjusted Technical Specification limit for that cycle. The quadrant tilt is calculated by the following method:

$$\% \text{ Tilt} = 100 \left(\frac{\text{Power in Any Core Quadrant}}{\text{Average Power of All Quadrants}} - 1 \right)$$

The average quadrant power and average core power are calculated from the sixteen detectors which make up the inner and outer symmetric rings.

The results are reviewed by the Test Coordinator. If the reactor calculations outputs appear normal and there is no indicated incore tilt of more than the Technical Specification limit (positive or negative), then the results are acceptable.

FOLLOW-UP ACTIONS:

If the reactor calculations outputs appear normal and an indicated incore tilt of greater than the Technical Specification limit (positive or negative) exists, the cause of the indicated tilt is investigated. If the tilt is due to identifiable reactor conditions (such as biased cold leg temperatures), it is corrected per normal operating procedures. If the cause cannot be identified, the Reactor Engineer is contacted to initiate a program of testing and evaluation before further power increase.

If the reactor calculations outputs appear abnormal, the raw detector signals are evaluated to determine if a significant core asymmetry exists. If no significant asymmetry exists, power escalation is continued. If an asymmetry exists, the Reactor Engineer is contacted to initiate a program of testing and evaluation before further power increase. The problem with the reactor calculations program is investigated and corrected, but this is not a prerequisite for power increase if no significant asymmetry exists.

LOW POWER CORE MAPPING

CONDITIONS:

30 to 60% FP, ∿ 579°F, ∿ 2155 psig, full reactor coolant flow.

PROCEDURE:

Once the unit has attained 30 to 60% FP, the output of the plant process computer reactor calculations program is analyzed. This program processes the signals from fixed incore detectors and provides a relative core power distribution as output. The incore detector outputs are checked, in order to identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained from the process computer are compared with the values calculated as a part of the reload design process on an eighth-core basis. For those eighth-core locations which have more than one symmetric instrumented location, instrument signals are averaged for comparison with design peaking values.

Independent of this comparison with design values, the measured core power distribution values are used to calculate maximum linear heat rate values for comparison with Technical Specification limits. No averaging is performed, so that the limiting value will come from the highest individual detector output for the entire core. This output is adjusted for axial local peaking, radial local peaking, hot channel factor, nuclear uncertainty and power uncertainty. This adjusted value is then used for comparison with the specified limits.

The measurement uncertainty for radial peak is less than 5% and for a total peak is less than 7.5%. The uncertainties are taken care of in the case of the maximum linear heat rate comparison by the application of adjustment factors to conservatively account for the individual sources of error, so that the result which is compared against Technical Specification Power limits is a conservatively high value.

The results are reviewed by the Test Coordinator. If the highest measured radial peaking factor does not exceed the highest predicted radial peaking factor by more than 8.0% of the highest measured radial peaking factor, and if the highest measured total peaking factor does not exceed the highest predicted total peaking factor by more than 12% of the highest measured total peaking factor and if the RMS difference between predicted and measured radial peaking factors is less than 10%, then the results are acceptable.

If any observed parameter exceeds its specified value in the Technical Specifications, actions will be taken as required by the Technical Specifications.

FOLLOW-UP ACTIONS:

If the adjusted maximum linear heat rate value exceeds the LOCA limiting heat rate specified in the Technical Specifications, reactor power will be reduced to a power level such that the adjusted maximum linear heat rate value is less than the specified value. The results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the recommended action approved by the Technical Review Committee prior to any further escalation of power.

If the acceptance criteria is exceeded, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the results and recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

The low power core mapping results are compared to the Full Power Core Mapping Acceptance Criteria. If any of the results at low power exceed the criteria at full power, an intermediate core mapping between 70 and 85% FP will be performed. If the results at low power meet the criteria at full power, only a full power mapping will be required.

INTERMEDIATE POWER CORE MAPPING

CONDITIONS:

70 to 85% FP, ∿579°F, ∿2155 psig, full reactor coolant flow.

PROCEDURE:

Once the unit has stabilized at 70 to 85% FP, the output of the plant process computer reactor calculation program is analyzed. This program processes the signals from fixed incore detectors and provides a relative core power distribution as output. The incore detector outputs are checked, in order to identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained from the process computer are compared with the values calculated as part of the reload design process on an eighth-core basis. For those eighth-core locations which have more than one symmetric instrumented location, instrument signals are averaged for comparison with design peaking values.

Independent of this comparison with design values, the measured core power distribution values are used to calculate maximum linear heat rate values for comparison with Technical Specification limits. No averaging is performed, so that the limiting value will come from the highest individual detector output for the entire core. This output is adjusted for axial local peaking, radial local peaking, hot channel factor, nuclear uncertainty and power uncertainty. This adjusted value is then used for comparison with the specified limits.

The measurement uncertainty for radial peak is less than 5% and for total peak is less than 7.5%. The uncertainties are taken care of in the case of the maximum linear heat rate comparison by the application of adjustment factors to conservatively account for the individual sources of error, so that the result which is compared against Technical Specification Power limits is a conservatively high value.

The results are reviewed by the Test Coordinator. If the highest measured radial peaking factor does not exceed the highest predicted radial peaking factor by more than 5.0% of the highest measured radial peaking factor, and if the highest measured total peaking factor does not exceed the highest predicted total peaking factor by more than 7.5% of the highest measured total peaking factor, and if the RMS difference between predicted and measured radial peaking factors is less than 7.5%, then the results are acceptable.

If any observed parameter exceeds its specified values in the Technical Specificaions, actions will be taken as required by the Technical Specifications.

FOLLOW-UP ACTIONS:

If the adjusted maximum linear heat rate value exceeds the LOCA limiting heat rate specified in the Technical Specificaitons, reactor power will be reduced to a power level such that the adjusted maximum linear heat rate value is less

than the specified value. The results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the recommended actions approved by the Technical Review Committee prior to any further escalation of power.

If the acceptance criteria is exceeded, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the results and recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

FULL POWER CORE MAPPING

CONDITIONS:

90 to 100% FP, $\sim$579°F, $\sim$2155 psig, full reactor coolant flow.

PROCEDURE:

Once the unit has stabilized at 90 to 100% FP, the output of the plant process computer reactor calculation program is analyzed. This program processes the signals from fixed incore detectors and provides a relative core power distribution as output. The incore detector outputs are checked, in order to identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained from the process computer are compared with the values calculated as part of the reload design process on an eighth-core basis. For those eighth-core locations which have more than one symmetric instrumented location, instrument signals are averaged for comparison with design peaking values.

Independent of this comparison with design values, the measured core power distribution values are used to calculate maximum linear heat rate values for comparison with Technical Specification limits. No averaging is performed, so that the limiting value will come from the highest individual detector output for the entire core. This output is adjusted for axial local peaking, radial local peaking, hot channel factor, nuclear uncertainty and power uncertainty. This adjusted value is then used for comparison with the specified limits.

The measurement uncertainty for radial peak is less than 5% and for total peak is less than 7.5%. The uncertainties are taken care of in the case of the maximum linear heat rate comparison by the application of adjustment factors to conservatively account for the individual sources of error, so that the result which is compared against Technical Specification Power limits is a conservatively high value.

The results are reviewed by the Test Coordinator. If the highest measured radial peaking factor does not exceed the highest predicted radial peaking factor by more than 5.0% of the highest measured radial peaking factor, and if the highest measured total peaking factor does not exceed the highest predicted total peaking factor by more than 7.5% of the highest measured total peaking factor, and if the RMS difference between predicted and measured radial peaking factors is less than 7.5%, then the results are acceptable.

If any observed parameter exceeds its specified values in the Technical Specificaions, actions will be taken as required by the Technical Specifications.

FOLLOW-UP ACTIONS:

If the adjusted maximum linear heat rate value exceeds the LOCA limiting heat rate specified in the Technical Specificaitons, reactor power will be reduced to a power level such that the adjusted maximum linear heat rate value is less

than the specified value. The results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed and the recommended actions approved by the Technical Review Committee prior to any further escalation of power.

If the acceptance criteria is exceeded, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy.

REACTIVITY ANOMALY

CONDITIONS:

HFP, 580°F, 2155 psig, full reactor coolant flow.

PROCEDURE:

As a part of the periodic testing program and separate from the startup testing program, the all-rods-out critical boron concentration at power is checked against normalized predicted values approximately each 10 EFPD of steady-state operation. With the reactor at steady-state conditions, as near as practical to full-power rods-out conditions, a sample of reactor coolant is taken and analyzed for boron concentration. This value of boron concentration is then adjusted to account for the reactivity worth of regulating rods in the core at the time of the measurement, and any other minor variations from design conditions.

The results are reviewed on-site by the cognizant Performance Engineer and are compared with the normalized predicted all-rods-out boron concentration for the time in the cycle at which the measurement was taken. If the difference between the measured and predicted values does not exceed 50 ppm B, the results are acceptable.

FOLLOW-UP ACTIONS:

If the difference between measured and predicted all-rods-out boron concentration is greater than 100 ppm B, then the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy pursuant to Technical Specification 4.10.

If the acceptance criteria is not met, the results will be reviewed by cognizant engineers to determine the appropriate corrective action required to resolve the discrepancy. This review will be completed and the results and recommended corrective actions approved by the Technical Review Committee within 14 days.

TECHNICAL JUSTIFICATION FOR OGSP CHANGES

Elimination of the 75% FP Power Distribution Measurement

The attached table presents the peaking factor data applicable to the acceptance criteria for Oconee PET's. The following trends are evident:

- The results are generally very similar at the three power levels. This is shown most clearly by the more complete descriptor applied to the latter cycles, the root-mean-square (RMS) radial peaking factor deviation.

- Significant changes tend to be toward less positive deviations at higher power levels (i.e., toward smaller measured peaks). This is expected, since the deviations from prediction are often contributed to by minor flux asymmetries which decrease in magnitude with increased burnup and power level. When power distribution problems have existed, they have been evaluated during low power testing.

- In no case has a criterion not been met at 75 or 100% FP after having been met at 40% FP.

Low power testing has proved sufficient for conservative extrapolations of fuel thermal parameters to the trip and LOCA limits, thereby demonstrating that full power operation can safely be attained. Redundant extrapolations of this type have been eliminated from the 75 and 100% FP portions of the Oconee startup test procedures. Alarms are generated if the plant computer-measured maximum linear heat rate reaches the value assumed in the LOCA analysis; this is the most restrictive of the thermal limits.

It is concluded that no useful information is gained by power distribution measurements at 75% FP. Therefore, Oconee startup testing should include only one formal power distribution measurement below full power. Some PWR's have already adopted this practice.

Provision for Power Level Ranges for PET Measurements

The 5 to 20% FP range for the Core Symmetry Test takes advantage of a planned change to the plant computer software which will activate incore detector signal processing at 5% FP rather than at 15% FP. This will allow earlier identification of core power distribution problems, as well as computer or instrumentation malfunctions.

The 30 to 60% FP and 90 to 100% FP ranges for the low and full power tests provide operating flexibility for the largely arbitrary test power levels. The Core Symmetry Test, a reduced overpower trip setpoint, computer-generated alarms, and test engineer cognizance assure that an initial power level of up to 60% can be safely attained.

The power distribution tests are renamed "Core Mapping" to avoid confusing wording (e.g., "Low-Power Power Distribution").

OCONEE POWER ESCALATION TESTS
PEAKING FACTOR DATA APPLIED TO ACCEPTANCE CRITERIA
(obtained from startup test reports)

		Max Radial Peak %Dev[1]			Max Total Peak %Dev[1]			Radial Peak RMS Dev[2]		
Unit	Cycle	40%	75%	100%	40%	75%	100%	40%	75%	100%
1	1	+0.1	+4.4	+2.0	not reported			not used		
1	2	+4.9	+3.8	+2.6	+3.5	-0.5	+1.6	not used		
1	3	data not available			data not available			not used		
1	4[3]	+2.3	+6.0	+3.1	+13.3	+6.2	+1.3	not used		
1	5	+0.8	+1.4	+1.5	-5.2	-4.1	-7.3	not used		
1	6	+5.3	+0.6	+0.9	+6.7	+4.8	+4.7	not used		
1	7	+0.9	+1.1	+1.0	+1.0	+2.3	-0.4	0.042	0.041	0.041
2	1	-4.1	+0.7	-2.8	not reported			not used		
2	2	+4.3	+4.1	+4.1	+2.1	+2.2	+1.0	not used		
2	3	+7.7	+4.4	+4.3	+8.8	+6.0	+5.9	not used		
2	4	-2.9	-2.5	-2.1	+0.8	-1.4	+1.2	not used		
2	5	-3.8	-3.3	-2.8	+7.2	-9.0	-6.6	0.036	0.035	0.032
2	6	+2.3	+2.1	+4.1	0.0	-2.6	+5.1	0.042	0.030	0.033
3	1	-4.3	-10.2	-4.3	not reported			not used		
3	2	+0.4	+0.8	+0.7	+6.0	+1.8	-4.2	not used		
3	3[3]	+9.4	0.0	-0.5	+10.0	-9.8	-9.9	not used		
3	4	+4.9	+6.3	+6.5	+5.9	+0.4	-2.2	not used		
3	5	-2.1	-2.2	-1.0	+6.0	0.0	-3.4	not used		
3	6	-3.3	+2.0	-1.1	-3.0	-2.6	+1.2	0.037	0.022	0.028
3	7	-4.8	-5.4	-4.7	-6.0	-9.4	-6.9	0.063	0.061	0.061
acceptance criteria		≤+8.0	≤+5.0	≤+5.0	≤+12.0	≤+7.5	≤+7.5	≤0.100	≤0.075	≥0.075

[1]Maximum peak % deviation is defined:

$$\% \text{ dev} = \frac{(\text{max measured peak}) - (\text{max predicted peak})}{\text{max measured peak}} \times 100$$

[2]Radial peak root-mean-square deviation is defined:

$$\text{RMS dev} = \left[\sum_{1}^{n}(\text{predicted peak}_n - \text{measured peak}_n)^2/(n-1)\right]^{\frac{1}{2}}$$

Where n is the number of operable incore detector strings.

[3]Oconee 1 Cycle 4 and Oconee 3 Cycle 3 started up with large tilts which caused rapidly changing power distributions.

Duke Power Company

Oconee Nuclear Station

Attachment 3

No Significant Hazards Consideration
Evaluation

10 CFR §50.91 requires that requests for amendment must be accompanied by an evaluation of the hazards considerations involved. Such evaluation is to focus on the three standards set forth in 10 CFR §50.91(b) as quoted below:

> The Commission may make a final determination pursuant to the procedures in §50.91 that a proposed amendement to an operating license for a facility licensed under §50.21(b) or §50.22 or for a testing facility involves no significant hazards consideration, unless it finds that operation of the facility in accordance with the proposed amendent would:
>
> (1) Involve a signficiant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a signficiant reduction in a margin of safety.

For a nuclear power reactor, a change resulting from a reloading of the core has been given as an example of an amendement that is considered "not likely to involve a significant hazard consideration" as referenced in the procedures of the office Nuclear Reactor Regulation. This assumes that no fuel assemblies are significantly different from those found previously acceptable to the NRC for a previous core. In addition, that no significant changes are made to the acceptance criteria for the Technical Specifications, that the analytical method used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed and that the NRC has previously found such methods acceptable.

The Oconee Unit 1 reactor core and fuel design basis are described in detail in section 3 of the Oconee Nuclear Station Final Safety Analysis Report. The cycle 8 core contains 177-fuel assemblies, each of which is 15-by-15 array of 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. The fuel consists of dished-end, cylindrical pellets of uranium dioxide clad in cold-worked Zircaloy-4. The standard Mark B fuel assemblies in all batches have an average fuel loading of 463.6 kg of uranium. The undensified active fuel lengths, theoretical densities, fuel and fuel rod dimensions, and other related fuel parameters are given in the Oconee Unit 1, cycle 8 reload report (Attachement 4).

Five of the fresh batch 10 assemblies are gadolinia lead test assemblies (LTA). These assemblies are part of a joint Duke Power/Babcock and Wilcox (B&W)/ Department of Energy program to develop and demonstrate an advanced fuel assembly design incorporating UO_2-Gd_2O_3 for extended burnup in PWRs.

Design Report for Oconee 1 Lead Test Assemblies, BAW-1772, Babcock and Wilcox, Lynchburg, Virginia (to be published) describes the LTA's. In addition, four Mark BZ demonstration fuel assemblies containing Zircaloy-4 intermediate spacer grids will be reinserted for a second cycle of irradiation. The Mark BZ assemblies are described in the Mark BZ Demonstration Assemblies in Oconee 1, Cycles 7, 8, and 9, BAW-1661, Babcock and Wilcox, Lynchburg, Virginia, March 1981.

The LTA's and the Mark BZ assemblies are being loaded in the core in such a manner as to ensure that there will be no significant affect on cycle 8 operation and sufficient margin to offset any negative impact on the LOCA KW/ft limits.

The Oconee 1 Cycle 8 reload report (Attachment 4) justifies the operation of the eighth cycle at the rated core power of 2568 MWt. Included are the required analyses as outlined in the USNRC document "Guidance for Proposed License Amendments Relating to Refueling," June 1975. To support cycle 8 operation of Oconee 1, this report employs analytical techniques and design bases established in reports that have been submitted to and accepted by the USNRC and its predecessor.

The Technical Specifications have been reviewed, and the modifications required for cycle 8 operation are justified in the reload report. The revisions were made, in accordance with methods and procedures found acceptable through previously submitted reloads, in order to account for changes in power peaking and control rod worths. The final acceptance criteria of the ECCS limits will not be exceeded, and the thermal design criterion will not be violated.

Based on the analyses performed, which account for the postulated effects of fuel densification and the final acceptance criteria for emergency core cooling systems, it has been concluded that Oconee Unit 1 can be operated with no signficant reduction in the margin of safety for cycle 8 at the rated power level of 2568 MWt.

The following evaluation demonstrates by reference to previously performed analysis, that not one of the three significant safety hazards consideration standards are met. Each of the three standards are briefly discussed below:

First Standard

> Involve a signficiant increase in the probability or consequence of an accident previously evaluated.

The probabilities and consequences of an accident resulting from the changes included in the cycle 8 reload are addressed in Section 7 of the BAW-1774 Oconee Unit 1, cycle 8 reload report (Attachment 4).

Each accident analysis addressed in the Final Safety Analysis Report (FSAR) has been examined with respect to changes in cycle 8 parameters to determine the effect of the cycle 8 reload and to ensure that thermal performance during hypothetical transients is not degraded. The probability of each accident described in the FSAR does not increase as a result of the cycle 8 reload changes.

The radiological dose consequences of the accidents presented in chapter 15 of the FSAR were recalculated using the specific parameters applicable to cycle 8. The bases used in the dose calculations are identical to those in the FSAR except that updated dose conversion factors were used. The use of the updated dose conversion factors resulted in reduced whole body dose values. Comparisons of the revised FSAR dose values with those calculated specifically for cycle 8 indicate some cycle 8 doses vary slightly from the FSAR values.

However, all cycle 8 doses are either bounded by the values presented in the FSAR or are a small fraction of the 10 CFR 100 limits, i.e., below 30 REM to the thyroid or 2.5 REM to the whole body. Thus, the radiological impact of the accidents during cycle 8 are not significantly different than those described in chapter 15 of the FSAR.

Thus, it is shown that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

> Create the possibility of a new or different kind of accident from any accident previously evaluated.

The analysis performed in support of this reload are in accordance with the USNRC document "Guidance for proposed license amendments relating to refueling". The analysis found that the proposed reload does not in any way create the possibility of a new or different kind of accident from any accident previously evaluated.

Third Standard

> Involve a Significant reduction in a margin of safety.

The issue of margin of safety for a reload modification needs to address the following areas:

1. Fuel System Design considerations

2. Nuclear Design considerations

3. Thermal-Hydraulic Design considerations

Section 4, 5, 6 of the Oconee 1 cycle 8 reload report (Attachment 4) addresses the above areas, respectively. Evaluation of each margin of safety area listed above is summarized below:

Fuel System Design Consideration

The Fuel System Design consideration is described in section 4 of Attachment 4. The most limited fuel assemblies, from the collapse of the clad standpoint are the batch 8C assemblies because of their longer incore exposure time. The most limiting fuel assembly was compared to the generic MK-B creep collapse analysis and were found to be enveloped. The analysis performed is based on the methods and procedures described in the B&W report, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse" (BAW-10084A, Rev. 2). The generic analysis predicts a collapse time of more than 35,000 EFPH which exceeds the maximum projected residence time of 29,112 EFPH. A detailed creep analysis of the gadolinia bearing fuel rods in the LTA's indicated that the collapse time for these rods was greater than maximum projected residence time.

The stress parameters for the Oconee 1 fuel rods were analyzed. For design evaluation, the primary membrane stress must be less than two-thirds of the

minimum specified unirradiated yield strength, and all stresses (primary and secondary) must be less than the minimum specified unirradiated yield strength. In all cases, the margin is in excess of 30%. The fuel design criterion specified for the cladding average circumferential strain is not to exceed 1.0% inelastic strain. The pellet design is established for a plastic cladding strain of less than 1% at maximum design local pellet burnup and heat generation rate values that are higher than the values the Oconee 1 UO_2 fuel is expected to see. the strain analysis for the gadolinia fuel showed that these rods are also below design limits. Thus, fuel rod cladding strain will not effect the fuel performance of cycle 8.

The batch 10 fuel assemblies are not new in concept, nor do they utilize different component materials, execpt for the Zircaloy grids of the four Mark BZ assemblies and the UO_2 -Gd_2O_3 pellets in the LTAs. Therefore, the chemical compatibility of all possible fuel-cladding-coolant-assembly interactions for the batch 10 fuel assemblies is acceptable.

Nuclear Design Consideration

The Nuclear Design Analysis was performed using similar methods and procedures that were employed for previous reload reports that have been submitted and accepted by the NRC. Section 5 of Attachment 4 describes the core physics parameters of cycle 8. The required shutdown margin is 1.00% $\Delta k/K$, the shutdown margin at the beginning of the cycle and at the end of the cycle is 3.11% $\Delta k/K$ and 2.28% $\Delta k/K$ respectively. Thus, the analysis shows that the cycle 8 design meets all criteria including those applicable to radial power peaking, ejected rod worths and shutdown margin.

The gadolina fuel assemblies are being loaded in such a manner as to ensure that there will be no significant effect on the core physics parameters.

Thermal-Hydraulic Design Consideration

All fuel in the cycle 8 core is thermally similar except the five LTAs. The fresh batch 10C fuel inserted for cycle 8 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The fresh batch 10A and 10B fuel containing the gadolinia LTA demonstration assemblies have different fuel performance characteristics, but are not more limiting than the remainder of the core. The cycle 8 thermal analyses represent a change in an analytical method, in that the fresh batches of fuel have been analyzed with the TACO2 code using the methodology described in J. H. Taylor April 8, 1982 letter to J. S. Berggrenan; "B&W's" response to TACO2 questions. The analysis uses nominal undensified input parameters. The TACO2 code densification model accounts for densification effects. TACO2 analyses was also applied to the reinserted batch 8C and 9 fuel since this fuel is identical in design to batch 10C.

Results of the thermal design evaluation for the cycle 8 core are summarized in Table 4-2 of attachment 4. The TACO2 fuel performance code was used to determine linear heat rate to melt capabilities for batch 8C, 9, 10 fuel (95% TD nominal initial density). Maximum linear heat rate to centerline melt was determined as a function of fuel burnup. The maximum allowable linear heat rate was 20.5 kW/ft for 8C, 9, and 10C batches of fuel. The maximum allowable linear heat rate for the batch 10A and 10B LTA gadolinia fuel is 17.6 kW/ft.

The maximum fuel rod burnup at EOC 8 is predicted to be 40,238 MWd/mtU. Fuel rod internal pressure was evaluated with the TACO2 computer code for the highest burnup fuel rod and is predicted to be less than the nominal RC system pressure of 2200 psia.

The incoming batch 10 fuel is hydraulically and geometrically similar to the fuel reamining in the core from previous cycles. Thermal-hydraulic design evaluation supporting cycle 8 operation used the methods and models described in section 6 of Attachment 4, and which have been found to be acceptable by the NRC through previously submitted reloads. A rod bow penalty has been calculated according to the procedure approved in L. S. Rubenstein October 18, 1970 letter to J. H. Taylor on the evaluation of interim procedure for calculating DNBR reductions due to rod bow". The maximum fuel assembly burnup of the batch that contains the limiting (maximum radial-local peak) fuel assembly is used. For cycle 8, this burnup is 17,511 in a batch 10C assembly. The resultant net rod bow penalty, after inclusion of the 1% flow area reduction factor credit, is 0.2% reduction in DNBR. Thermal-hydraulic design for cycle 8 includes a margin greater than 0.2% above the minimum DNBR of 1.30.

It can be concluded from the examination of these sections, that the cycle 8 core thermal and kinetics properties, with respect to acceptable previous cycle values, that this core reload will not significanlty reduce the ability of the Oconee 1 plant to operate safely during cycle 8.

Thus, it has been shown that the proposed Oconee 1 cycle 8 reload does <u>not</u>:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

As such, Duke has determined and submits that the proposed reload described herein does <u>not</u> involve a significant safety hazard.