Duke Power Company
Oconee Nuclear Station

Attachment 1

Proposed Technical Specification Revision
Oconee 2, Cycle 9

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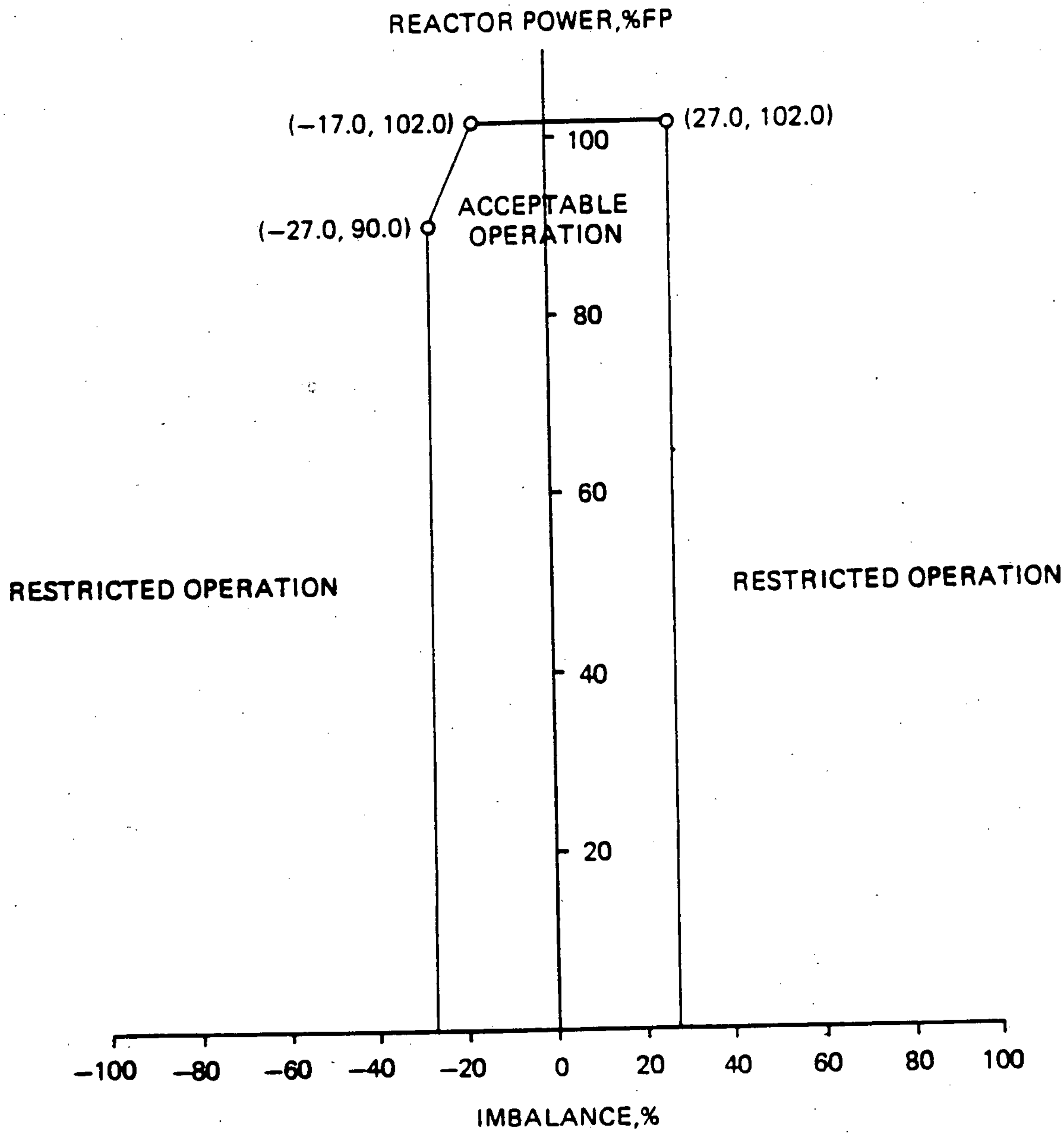

OPERATIONAL POWER
IMBALANCE ENVELOPE
FROM 0 EFPD TO EOC
UNIT 2

OCONEE NUCLEAR STATION

FIGURE 3.5.2-11
(1 of 1)

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Attachment 2

No Significant Hazards Consideration

No Significant Hazards Consideration Evaluation
for Oconee Unit 2, Cycle 9 Reload

Duke Power has determined that the present amendment request poses no significant hazards as defined by NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

(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 form any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

The commission has provided guidelines pertaining to the application of the three standards by listing specific examples in 48 FR 14870. Example (iii) of the types of amendments not likely to involve significant hazards considerations expressly applies inasmuch as the proposed amendment involves a nuclear power reactor core reload.

Example (iii) of amendments not likely to involve a significant hazards consideration concerns a core reload, assuming that:

(1) no fuel assemblies significantly different from those found previously acceptable to the NRC or a previous core at the facility in question are involved,

(2) no significant changes are made to the acceptance criteria for the technical specifications,

(3) the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed, and

(4) the NRC has previously found such methods acceptable.

The Mark BZ fuel assembly, 60 of which comprise Batch 11, is an improved version of the Mark B fuel assembly, in that the six Inconel Intermediate Spacer grids are replaced with Zircaloy grids. The Mark BZ spacer grids have the same functional design a the Mark B grids, but are slightly different dimensionally to accommodate Zircaloy's material properties. The other fuel assembly components such as the fuel rods, end grid, end fittings, and guide tubes are the same for both designs. The interfaces with the control rods and fuel handling equipment are unchanged, ensuring compatibility with the present reactor site operational procedures. The Mark BZ fuel assembly has been analyzed to ensure conformance with the standard review plan acceptance criteria. Therefore, on the whole, the Mark BZ fuel assembly is a minor design change from the Mark B -- with the Zircaloy spacer grids and related changes being the principal differences. The use of the Mark BZ fuel assembly in the Oconee Unit 1, Cycle 9 and Oconee 2 Cycle 8 core reload was accepted by the NRC via the Staff's approval of the Unit 1, Cycle 9 amendment request dated November 23, 1984 and the Unit 2, Cycle 8 amendment request dated April 18, 1985.

As was used in Cycle 8, gray (less-absorbing) axial power shaping rods (APSR's) are to be utilized. The staff approved the use of gray APSRs in the Oconee Unit 1, Cycle 9 and Oconee Unit 2, Cycle 8 reload amendment request.

The present reload involves no significant changes to the acceptance criteria for the Technical Specifications. Revisions of the Technical Specifications required for Cycle 9 operation were made in accordance with methods and procedures found acceptable in connection with previous reloads. The final acceptance criteria of the ECCS limits will not be exceeded, and thermal design criteria will be satisfied.

The Oconee Unit 2, Cycle 9 Reload Report (provided by Duke letter dated June 30, 1986) justifies the operation of the ninth 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 1985. The Reload Report employs analytical techniques and design bases established in reports submitted for previous reloads which were accepted by USNRC and its predecessor. These techniques are described in the Reload Report references.

With supporting reference to previously performed analyses, the following evaluation measures aspects of the Unit 2, Cycle 9 reload against the Part 50.92 (c) requirements to demonstrate that all three standards are satisfied.

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

> Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) has been examined with respect to changes in Cycle 8 parameters to determine the effect of the Cycle 9 reload and to ensure that thermal performance during hypothetical transients is not degraded. The transient evaluation of Cycle 9 is considered to be bound by previously accepted analyses. Section 7 of the Reload Report addresses "Accident and Transient Analysis" for this core reload. This analysis ensures that the proposed reload will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident from any accident previously evaluated.

> The analyses performed in support of this reload are in accordance with the USNRC document "Guidance for Proposed License Amendments Relating to Refueling", June 1975. The conclusion of the overall analysis is 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

(Amendment would not) involve a significant reduction in a margin of safety.

The issue of margin of safety for a reload modification involves the following areas:

1. Fuel System Design considerations,

2. Nuclear Design considerations, and

3. Thermal-Hydraulic Design considerations.

Sections 4, 5, and 6 of the Oconee Unit 2, Cycle 9 Reload Report addresses the above areas, respectively. The value limits and margins discussed in these areas are well within the allowable limits and requirements, and reflect no significant reductions to any margins of safety. One can conclude from the examination of these sections, and the Cycle 9 core thermal and kinetic properties (with respect to previous cycle values), that this core reload will not significantly reduce the ability of Oconee Unit 2 to operate safely during Cycle 9.

The above evaluation, with its accompanying references, shows that the three Part 50.92 (c) standards are satisfied. In summary, Duke has determined and submits that the proposed reload described herein does not represent any significant hazards.

Duke Power Company
Oconee Nuclear Station

Attachment 3

Startup Physics Test Program Revisions

STARTUP PHYSICS TEST PROGRAM

The Startup Physics Test Program for Oconee Nuclear Station, or OSPTP, is structured to provide assurance that the installed reactor core following each reload conforms to the designed 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. Control 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. Low Power Testing (5-30% FP)
   b. Intermediate Power Testing (40-75% FP)
   c. Full Power Testing (90-100% FP)

In addition to the above tests which comprise the basic Startup Physics Test Program, a separate test, the Reactivity Anomaly at Full Power, is performed approximately each 10 EFPD (Effective Full Power Days), 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" (ARO) hot full power (FP) 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 Company 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* will approve actions under the conditions stated for each test.

*This Committee will consist of representatives from Oconee Nuclear Station Performance and Compliance, and General Office Nuclear Engineering.

PRE-CRITICAL TEST PHASE

CONTROL ROD DROP TIME

PLANT CONDITIONS

Hot Shutdown, approximately 532^{o}F, approximately 2155 psig, full reactor coolant system (RCS) flow (4 pumps).

PROCEDURE

The control 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 operator aid computer (OAC) 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 CRAs 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 CRA group. In either case, the OAC records the drop time of each CRA individually.

The results are reviewed by the Test Coordinator and compared with the acceptance criterion, 1.40 seconds. The accuracy of the measurement of control rod drop time as performed by the OAC is approximately $\pm$ 0.005 seconds.

FOLLOW-UP ACTIONS

If any measured control rod drop time is greater than 1.40 seconds but less than 1.66 seconds, then 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 100% FP.

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

ZERO POWER PHYSICS TEST PHASE

CRITICAL BORON CONCENTRATION

PLANT CONDITIONS

Hot Zero Power, approximately 532^{o}F, approximately 2155 psig, steady RCS flow (3 or 4 pumps).

PROCEDURE

Critical boron concentration measurement is taken at the 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 through 7 are fully withdrawn. Control Rod Group 8 is maintained at the nominal designed position. 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 critical equilibrium conditions with Group 7 fully withdrawn, the small amount of inserted worth of Group 7 or worth of Group 8 (from its nominal designed position) is measured by a reactivity calculation or Reactimeter. This reactivity is then used to adjust the boron concentration to obtain the measured ARO boron concentration.

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

FOLLOW-UP ACTIONS

If the acceptance criterion ($\pm$ 50 ppmb) between measured and predicted ARO critical boron concentration 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 with the results and recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

If the difference between measured and predicted ARO critical boron concentration is greater than 100 ppm Boron, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended actions approved by the Technical Review Committee prior to exceeding 15% FP.

MODERATOR TEMPERATURE COEFFICIENT

PLANT CONDITIONS

Hot Zero Power, approximately 532^{o}F, approximately 2155 psig, steady RCS flow (3 or 4 pumps).

PROCEDURE

The moderator temperature coefficient (MTC) test begins with the reactor at critical equilibrium conditions. This test is performed by executing a change in RCS average temperature of approximately $\pm$ 5^{o}F while data are taken. Stability in RCS temperature is necessary at this first plateau. The hold time at each RCS temperature plateau during the test is approximately five minutes. After data are taken at the first RCS temperature plateau, the RCS average temperature is changed approximately 5^{o}F in the opposite direction and allowed to stabilize. Changes in reactivity associated with the induced RCS temperature transient are measured by a reactivity calculation or Reactimeter. This overall temperature coefficient is corrected for the contribution of the isothermal doppler coefficient of reactivity to give the moderator coefficient of reactivity. The measurement is also corrected to an average temperature of 532^{o}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 0.3×10^{-4} Delta $k/k/^{o}F$, then the results are acceptable.

FOLLOW-UP ACTIONS

If the measured maximum positive MTC exceeds 0.5×10^{-4} Delta $k/k/^{o}F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended actions approved by the Technical Review Committee prior to exceeding 15% FP.

If the 0.3×10^{-4} Delta $k/k/^{o}F$ acceptance criterion is exceeded and the maximum positive MTC is less than 0.5×10^{-4} Delta $k/k/^{o}F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended corrective actions approved by the Technical Review Committee prior to 100% FP.

CONTROL ROD WORTH

PLANT CONDITIONS

Hot Zero Power, approximately $532^{o}F$, approximately 2155 psig, steady RCS flow (3 or 4 pumps).

PROCEDURE

The measurements of regulating rod group worths begin from a critical steady state condition with all regulating rod groups withdrawn as far as possible (i.e., within 0.12% Delta k/k of ARO). From this point a boron concentration necessary to deborate control rod Groups 7, 6 and 5 to fully inserted is calculated. The resulting reactivity change during deboration is compensated for by discrete insertion of control rods with both signals being recorded by a reactivity calculation or Reactimeter. Differential rod worths for these insertions are calculated by dividing the difference in reactivity for the control rod insertion by the difference in control rod position, and integral rod worths are calculated by summing the differential rod worths for each control rod group.

The results are reviewed by the Test Coordinator and compared with the predicted control rod 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 control 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 control rod Groups 5, 6, and 7 exceeds 10%, then, following calculation of the minimum control

rod position for which the worth of the control rods withdrawn would equal 1% Delta k/k, additional control rod group worths will be measured. The worths of safety control 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 control rod groups measured does not exceed 10%, or the calculated minimum control rod position is reached. In the latter case, control rod worth testing will halt. 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 with the results and the recommended actions approved by the Technical Review Committee prior to exceeding 15% FP.

If the difference between the measured and predicted control rod worths of any of the individual control rod groups 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.

POWER ESCALATION TEST PHASE

LOW POWER TESTING

PLANT CONDITIONS

5 to 30% FP, approximately 579^{o}F, approximately 2155 psig, full RCS flow (4 pumps)

PROCEDURE

Once the unit is between 5 and 30% FP, the output of the plant OAC 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 for corrected assembly power in functioning instrumented symmetric core locations are compared.

The results are reviewed by the Test Coordinator. If the reactor calculations outputs appear normal, and the deviation between the highest and lowest corrected assembly power for symmetric core locations is less than $\pm$ 10%, then the results are acceptable.

FOLLOW-UP ACTIONS

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.

If the reactor calculations outputs appear normal and the deviation between corrected assembly powers for symmetric core locations is greater than $\pm$ 10%, the

cause of the indicated deviation is investigated. If the deviation is due to identifiable reactor calculations program problems, it is corrected per normal procedures and power escalation testing may continue. If the cause of the deviation cannot be identified, the Reactor Engineer is contacted to initiate a program of testing and evaluation.

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

INTERMEDIATE POWER TESTING

PLANT CONDITIONS

40 TO 75% FP, approximately 579^{o}F, approximately 2155 psig, full RCS flow (4 pumps)

PROCEDURE

Once the unit is between 40 and 75% FP, the output of the plant OAC 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 plant OAC are compared with the values calculated as part of the reload design process on an eighth-core basis.

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.

FOLLOW-UP ACTIONS

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

Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and the recommended corrective actions approved by the Technical Review Committee prior to any further escalation of power.

If any acceptance criteria are 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 with the results and recommended corrective actions approved by the Technical Review Committee prior to escalation to 100% FP.

FULL POWER TESTING

PLANT CONDITIONS

90 TO 100% FP, approximately 579^{O}F, approximately 2155 psig, full RCS flow (4 pumps)

PROCEDURE

Once the unit is between 90 and 100% FP with Xenon equilibrium, the output of the plant OAC 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 OAC are compared with the values calculated as part of the reload design process on an eighth-core basis.

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.

FOLLOW-UP ACTIONS

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

Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and the recommended corrective actions approved by the Technical Review Committee prior to any escalation of power.

If any acceptance criteria are 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 with the results and recommended corrective actions approved by the Technical Review Committee prior to any escalation of power.

REACTIVITY ANOMALY

PLANT CONDITIONS

Hot Full Power, approximately 579^{O}F, approximately 2155 psig, full RCS flow.

PROCEDURE

As a part of the periodic testing program and separate from the startup testing program, the ARO 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 ARO conditions, a sample of the RCS is taken and analyzed for boron concentration. This value of boron concentration is then adjusted to account for the reactivity worth of regulating control rod assemblies in the core at the time of the measurement, and any other minor variations from designed conditions.

The results are reviewed by the Reactor Engineer and are compared with the normalized predicted ARO boron concentration for the time in the cycle at which the measurement was taken. If the difference between the measured and predicted ARO boron concentration values does not exceed 50 ppm Boron, then the results are acceptable.

FOLLOW-UP ACTIONS

If the acceptance criterion (± 50 ppmb) is not met and the difference between measured and predicted ARO boron concentration is less than 100 ppm Boron, the results will be reviewed by cognizant engineers to determine the appropriate corrective action required to resolve the discrepancy. This review will be completed with the results and recommended corrective actions approved by the Technical Review Committee within 14 days.

If the acceptance criterion (± 50 ppmb) is not met and the difference between measured and predicted ARO boron concentration is greater than 100 ppm Boron, 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.

(PJN68)

STARTUP PHYSICS TEST PROGRAM

TECHNICAL JUSTIFICATION

The Oconee Nuclear Station Startup Physics Test Program (OSPTP) (Attachment 3) has numerous editorial changes and three minor changes based upon the recent approval of the American National Standard (ANS)-19.6.1, "Reload Startup Physics Tests for Pressurized Water Reactors."

Minor changes to the OSPTP are to the following sections:

1) "Followup actions" section for the Critical Boron Concentration, Moderator Temperature Coefficient, and Control Rod Worth, the "prior to exceeding 5% FP" has been changed to "prior to exceeding 15% FP". This change allows the plant to escalate to a point where the Reactor Calculations module on the Operator Aid Computer begins to operate producing additional Reactor data. This module which uses the fixed incore instrumentation system, and temperature and pressure sensors from around the plant, incorporates software which calculates thermal power, power imbalances, and power tilts for the Reactor. This module begins to calculate reliable data at greater than 5% FP.

2) Moderator Temperature Coefficient test, the testing band range has been modified from a "5 to 10 to 5^{o}F" temperature transient to "approximately $\pm$ 5^{o}F" temperature transient. The incorporation of this change enables the OSPTP to resemble the guidelines suggested and approved in ANS-19.6.1 which calls for a temperature transient between the ranges of $\pm$ 3^{o}F to $\pm$ 10^{o}F.

3) Power Escalation Test Phase, three testing plateaus have been established. This change replaces the previous document which contained four testing plateaus. The incorporation of this change enables the OSPTP to resemble the form suggested and approved in ANS-19.6.1.

These changes provide an enhanced OSPTP modeled after ANS 19.6.1.