DUKE POWER COMPANY
OCONEE NUCLEAR STATION

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS
CYCLE-SPECIFIC CORE OPERATING LIMITS

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LIST OF FIGURES

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3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525°F except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply.

3.1.3.2 Reactor coolant temperature shall be above the criticality limit of
3.1.2-1A (Unit 1)
3.1.2-1B (Unit 2)
3.1.2-1C (Unit 3)

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.9 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1%$\Delta k/k$ until a steam bubble is formed and a water level between 80 and 396 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. The regulating rods shall then be positioned within their acceptable operating position limits which are determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

Bases

At the beginning of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525°F, the consequences are acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative more more positive than at operating temperature,(2) startup and operation of the reactor when reactor coolant temperature is less than 525°F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1% $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the limits of Specification 3.1.2.1 provides increased assurance that the proper rela-

tionship between primary coolant pressure and temperature will be maintained relative to the RT_{NDT} of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than 1% subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a startup accident.[3]

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated. The acceptable operating position limits for the regulating rods for the appropriate unit and cycle are determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

REFERENCES

(1) FSAR, Section 4.3.2

(2) FSAR, Section 4.3.2.4

(3) FSAR, Section 15.3

3.1.11 Shutdown Margin

Specification

The available shutdown margin during all system conditions except refueling shall be greater than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn.

Bases

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

During power operation and startup the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

During refueling conditions equivalent protection is provided in the requirements of Specification 3.8.4.

c. If a control rod is declared inoperable by being immovable due to excessive friction or mechanical interference or known to be untrippable then:

1. Within 1 hour verify that the shutdown margin requirement of Specification 3.5.2.1 is satisfied and,

2. Within 12 hours place the reactor in the hot standby condition.

d. If a control rod is declared inoperable due to causes other than addressed in 3.5.2.2.c above then:

1. Within 1 hour either restore the rod to operable status or,

2. Continue power operation with the control rod declared inoperable and

a. Within 1 hour verify the shutdown margin requirement of Specification 3.5.2.1 with an additional allowance for the withdrawn worth of the inoperable rod and,

b. Either reactor thermal power shall be reduced to less than 60% of the allowable power for the reactor coolant pump combination within 1 hour and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow/imbalance, shall be reduced within the next 4 hours to 65.5% of thermal power value allowable for the reactor coolant pump combination or,

c. Position the remaining rods in the affected group such that the inoperable rod is maintained within allowable group average limits of Specification 3.5.2.2.a and within acceptable operating rod position withdrawal/ insertion limits determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

e. If more than one control rod is inoperable or misaligned, the reactor shall be shut down to the hot standby condition within 12 hours.

3.5.2.3 The worths of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the control rod position limits determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

3.5.2.4 Quadrant Power Tilt

a. Except for physics tests, the maximum positive quadrant power tilt shall not exceed the Steady State Limit of Table 3.5-1 during power operation above 15% full power.

b. If the maximum positive quadrant power tilt exceeds the Steady State Limit but is less than or equal to the Transient Limit determined in accordance with the methodology specified in the Reload Design Methodology Report, then:

   1. Either the quadrant power tilt shall be reduced within 2 hours to within its Steady State Limit or,

   2. The reactor thermal power shall be reduced below 100% full power by 2% thermal power for each 1% of quadrant power tilt in excess of the Steady State Limit, and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within 4 hours by 2% thermal power for each 1% tilt in excess of the Steady State Limit. If less than four reactor coolant pumps are in operation, the allowable thermal power for the reactor coolant pump combination shall be reduced by 2% for each 1% excess tilt.

c. Quadrant power tilt shall be reduced within 24 hours to within its Steady State Limit or,

   1. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1 and if there is a simultaneous indication of a misaligned control rod then:

   1. Reactor thermal power shall be reduced within 30 minutes at least 2% for each 1% of the quadrant power tilt in excess of the Steady State Limit.

   2. Either quadrant power tilt shall be reduced within 2 hours to within its Transient Limit or,

   3. The reactor thermal power shall be reduced within the next 2 hours to less than 60% of the allowable power for the reactor coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 4 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

e. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit of Table 3.5-1, due to causes other than simultaneous indication of a misaligned control rod then:

   1. Reactor thermal power shall be reduced within 2 hours to less than 60% of the allowable power for the reactor

coolant pump combination and the Nuclear Overpower Trip Setpoints, based on flux and flux/flow imbalance, shall be reduced within the next 2 hours to 65.5% of the thermal power value allowable for the reactor coolant pump combination.

f. If the maximum positive quadrant power tilt exceeds the Maximum Limit determined in accordance with the approved methodology, the reactor shall be shut down within 4 hours. Subsequent reactor operation is permitted for the purpose of measurement, testing, and corrective action provided the thermal power and the Nuclear Overpower Trip Setpoints allowable for the reactor coolant pump combination are restricted by a reduction of 2% of thermal power for each 1% tilt for the maximum tilt observed prior to shutdown.

g. Quadrant power tilt shall be monitored on a minimum frequency of once every 2 hours during power operation above 15% full power.

3.5.2.5 Control Rod Positions

a. Technical Specification 3.1.3.5 does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.

b. Except for physics tests, operating rod group overlap shall be 25% ± 5% between two sequential groups. If this limit is exceeded, corrective measures shall be taken immediately to achieve an acceptable overlap. Acceptable overlap shall be attained within two hours or the reactor shall be placed in a hot shutdown condition within an additional 12 hours.

c. Position limits are specified for regulating and axial power shaping control rods. Except for physics tests or exercising control rods, the regulating control rod insertion/withdrawal limits shall be maintained within acceptable operating limits as determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9 for the particular number of operating reactor coolant pumps (4, 3, 2).

If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. An acceptable control rod position shall then be attained within two hours. The minimum shutdown margin required by Specification 3.5.2.1 shall be maintained at all times.

3.5.2.6 Reactor power imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope determined in accordance with the approved methodology and provided in a Core Operating Limits Report per Specification 6.9.

If the imbalance is not within the acceptable envelope, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within two hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the manager or his designated alternate.

Bases

Operation at power with an inoperable control rod is permitted within the limits provided. These limits assure that an acceptable power distribution is maintained and that the potential effects of rod misalignment on associated accident analyses are minimized. For a rod declared inoperable due to misalignment, the rod with the greatest misalignment shall be evaluated first. Additionally, the position of the rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments. When a control rod is declared inoperable, boration may be initiated to achieve the existence of 1% $\Delta k/k$ hot shutdown margin.

The power-imbalance envelope obtained in accordance with the approved methodology is based on LOCA analyses which have defined the maximum linear heat rate (see Figure 3.5.2-5) such that the maximum clad temperature will not exceed the Final Acceptance Criteria. Corrective measures will be taken immediately should the indicated quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.** Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification power spike factors
d. Hot rod manufacturing tolerance factors
e. Fuel rod bowing power spike factors

The 25% ± 5% overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

Group	Function
1	Safety
2	Safety
3	Safety
4	Safety
5	Regulating
6	Regulating
7	Regulating
8	APSR (axial power shaping rod)

** Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument calibration errors. The method used to define the operating limits is defined in plant operating procedures.

The rod position limits obtained in accordance with the approved methodology are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. Therefore, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full out position(1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2,3,4, 5) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Because of the lower worth of the gray APSRs, operational limits for the Group 8 control rods are not necessary. Administrative position and maneuvering limits for the APSRs, however, will be utilized to ensure adequate fuel performance during reactor operation.

Control rod groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 25 percent. The normal position at power is for Group 7 to be partially inserted.

The quadrant power tilt limits set forth in Specification 3.5.2.4 have been established to prevent the linear heat rate peaking increase associated with a positive quadrant power tilt during normal power operation from exceeding 7.50% for Unit 1, 7.50% for Unit 2, 7.50% for Unit 3. The limits in Specification 3.5.2.4 are measurement system independent. The actual operating limits, with the appropriate allowance for observability and instrumentation errors, for each measurement system are defined in the station operating procedures.

The quadrant tilt and axial imbalance monitoring in Specification 3.5.2.4 and 3.5.2.6, respectively, normally will be performed in the process computer. The two-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service.

Allowance is provided for withdrawal limits and reactor power imbalance limits to be exceeded for a period of two hours without specification violation. Acceptable rod positions and imbalance must be achieved within the two-hour time period or appropriate action such as a reduction of power taken.

Operating restrictions resulting from xenon transients and power maneuvers are inherently included in the limits determined in accordance with the approved methodology.

Figures 3.5.2-1 Thru 3.5.2-15

(deleted)

6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1 The core operating limits shall be established for each reload core and shall be maintained available in the Control Room. The limits shall be established and implemented on a time scale consistent with normal procedural changes.

6.9.2 The analytical methods used to generate the core operating limits shall be those previously reviewed and approved by the NRC*. Any change to these methods shall be evaluated in accordance with 10 CFR 50.59. If the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation, then the change shall be submitted to the NRC for review and approval prior to its use.

6.9.3 A report containing the core operating limits shall be provided to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector within 30 days after their implementation.

*DPC-NE-1002A, Reload Design Methodology II, October 1985.
NFS-1001A, Reload Design Methodology, April 1984.

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

ATTACHMENT 2

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

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 from any accident previously evaluated; or

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

The proposed technical specifications concern the deletion of cycle-dependent core operating limits from the Oconee Technical Specifications. Specifically, the present Oconee Technical Specifications are revised for every cycle reload to reflect changes in cycle specific variables which are obtained from application of acceptable methodologies. This is an administrative burden to both the NRC staff and Duke Power Company (Duke).

By letter dated November 14, 1986 the NRC staff provided guidance for removal of certain cycle-dependent core operating limits from the Oconee Technical Specifications. This would allow changes to the values of core operating limits without prior approval (i.e., license amendment) by the NRC so long as an approved methodology for core reload design is followed. Instead, the future Oconee core reloads would involve a safety review in accordance with the requirements of 10 CFR 50, Part 50.59.

Currently, Duke utilizes the "Oconee Nuclear Station Reload Design Methodology II", DPC-NE-1002, dated October 1985 for design of Oconee core reloads. This methodology was reviewed and approved by the NRC staff in their Safety Evaluation Report (SER) dated October 1, 1985. The NRC concluded that the proposed methodology in report DPC-NE-1002A is an acceptable alternative to the Oconee reload methodology presented in NFS-100A (reviewed and approved in the SER dated July 29, 1981) and may be referenced by Duke Power Company in licensing submittals for the Oconee units.

The cycle dependent core operating limits removed in the proposed Technical Specification include:

(a) Minimum Conditions for Criticality,

(b) Shutdown Margin, and

(c) Rod Position Limits and Power Imbalance Limits.

The removal of cycle dependent variables from the Technical Specifications has no impact upon plant operation or safety. The Technical Specifications will

continue to require operation within the core operational limits for each cycle reload calculated by the approved reload design methodologies. Appropriate actions to be taken if limits are violated will also remain in the Technical Specifications.

The proposed changes are consistent with the requirements of 10 CFR 50, Part 50.36 and the staff's proposed policy for improving Technical Specifications, delineated in SECY-86-10, "Recommendations for improving TS". The policy allows that process variables such as core operational limits to be controlled by specifying them numerically in the Technical Specifications or by specifying the method of calculating their numerical values if the staff finds that the correct limits will be followed in operating the plant. The proposed revision references the approved Duke's core reload design methodology for future Oconee reloads. The development of cycle specific core operating limits will continue to be performed by the referenced methodology which has been accepted by the NRC.

The following provides a discussion of how the proposed amendment satisfies each of the three standards of 10 CFR 50.92(c).

<u>First Standard</u>

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

The removal of the cycle specific core operating limits from the Oconee Technical Specifications has no influence or impact on the probability of a Design Basis Accident (DBA) occurrence. The cycle specific core operating limits, although not in Technical Specifications, will be followed in the operating of the Oconee plants. The proposed amendment still requires exactly the same actions to be taken when or if limits are exceeded as is required by current Technical Specifications.

Each accident analysis addressed in the Oconee Final Safety Analysis Report (FSAR) will be examined with respect to changes in Cycle dependent parameters, which are obtained from application of the approved reload design methodologies, to ensure that the transient evaluation of new reloads are bounded by previously accepted analyses. This examination which will be performed per requirements of 10 CFR 50.59 ensures that future reloads will not involve a significant increase in the probability or consequences of an accident previously evaluated.

<u>Second Standard</u>

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

As stated earlier, the removal of the cycle specific variables has no influence, impact nor does it contribute in any way to the probability or consequences of an accident. The cycle specific variables are calculated using the NRC approved methods. The Technical Specifications will continue to require operation within the required core operating limits and appropriate actions will be taken when or if limits are exceeded.

The proposed amendment 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 margin of safety is not affected by the removal of cycle specific core operating limits from the Technical Specifications. The margin of safety presently provided by current Technical Specifications remains unchanged. The proposed amendment still requires operation within the core limits as obtained from the NRC approved reload design methodologies and appropriate actions to be taken when or if limits are violated remain unchanged.

The development of the limits for future reloads will continue to conform to those methods described in NRC approved documentation. In addition, each future reload will involve a Part 50.59 safety review to assure that operation of the unit within the cycle specific limits will not involve a significant reduction in a margin of safety.

In view of the preceding, Duke Power Company has determined that the proposed amendment does not involve significant hazards considerations.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 3

OCONEE 1 CYCLE 10

CORE OPERATING LIMITS REPORT

REVISION 0

28 AUGUST 1987

1.0 Core Operating Limits

This Core Operating Limits Report for O1C10 has been prepared in accordance with the requirements of Technical Specification 6.9. The core operating limits have been developed using NRC-approved methodology (Reference 1) and are documented in Reference 2.

The following cycle-specific core operating limits are included in this report:

1) Operational power-imbalance limits and

2) Operational and shutdown margin-limited control rod position limits for 4, 3, and 2-pump operation.

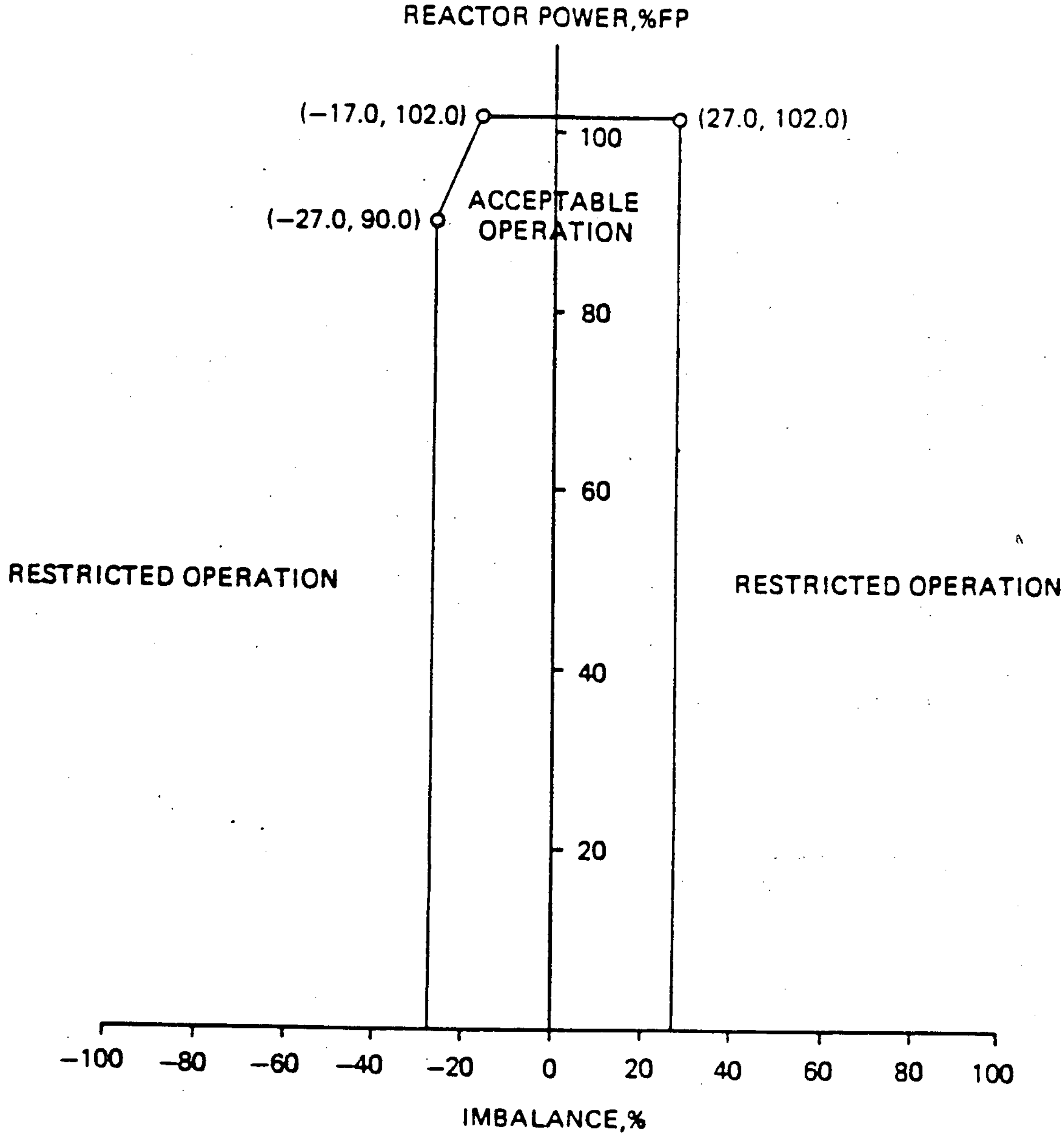
2.0 References

1. Duke Power Company, Oconee Nuclear Station, Reload Design Methodology II, DPC-NE-1002A, October 1985.

2. O1C10 Maneuvering Analysis, Duke Power Company calculational file, SRC-OS1-ND-85-007-0, August 1985.

Oconee 1 Cycle 10

Operational Power-Imbalance Limits, 0 EFPD to EOC



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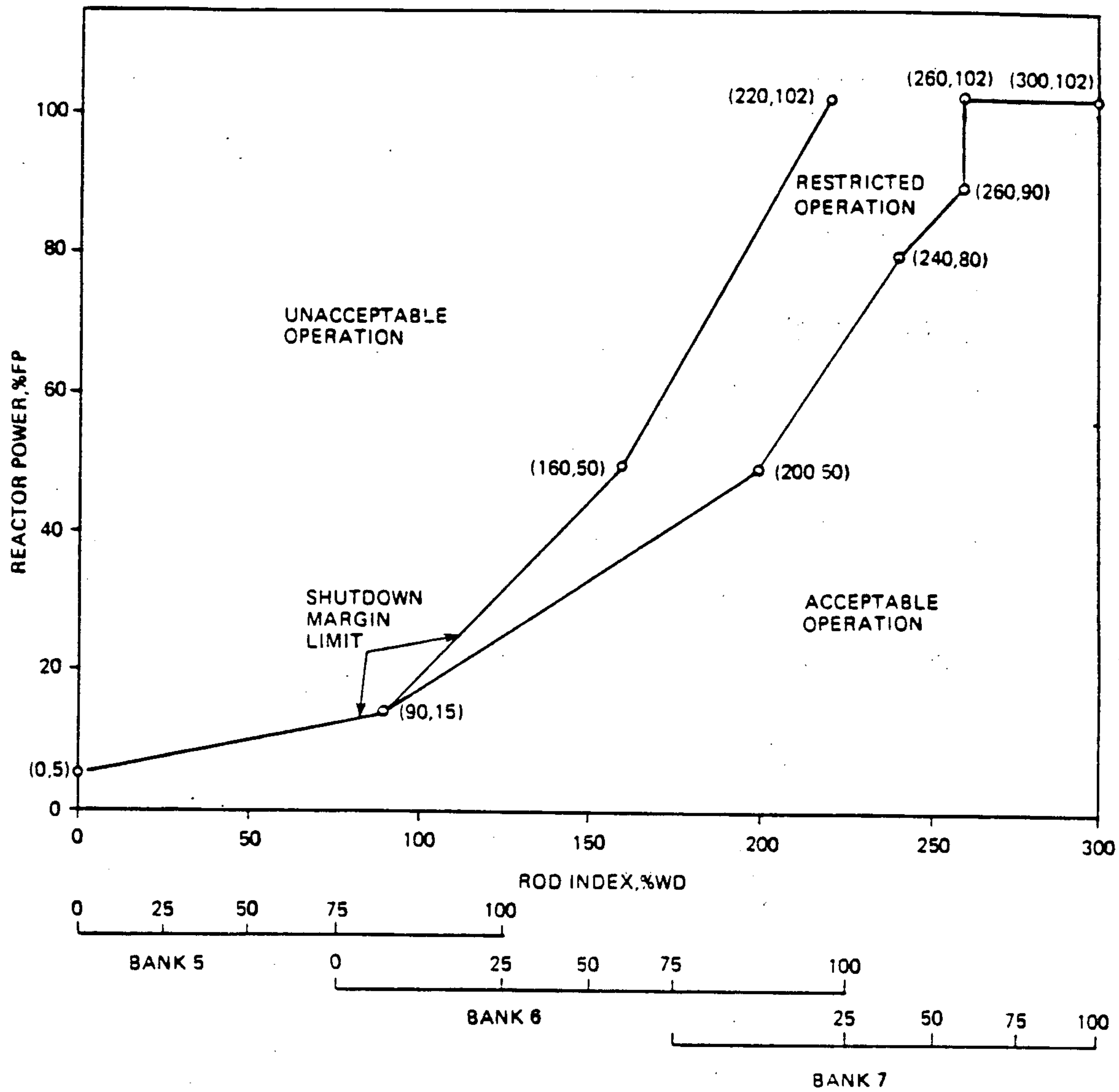
Control Rod Position Limits, 4 Pumps, 0 EFPD to EOC



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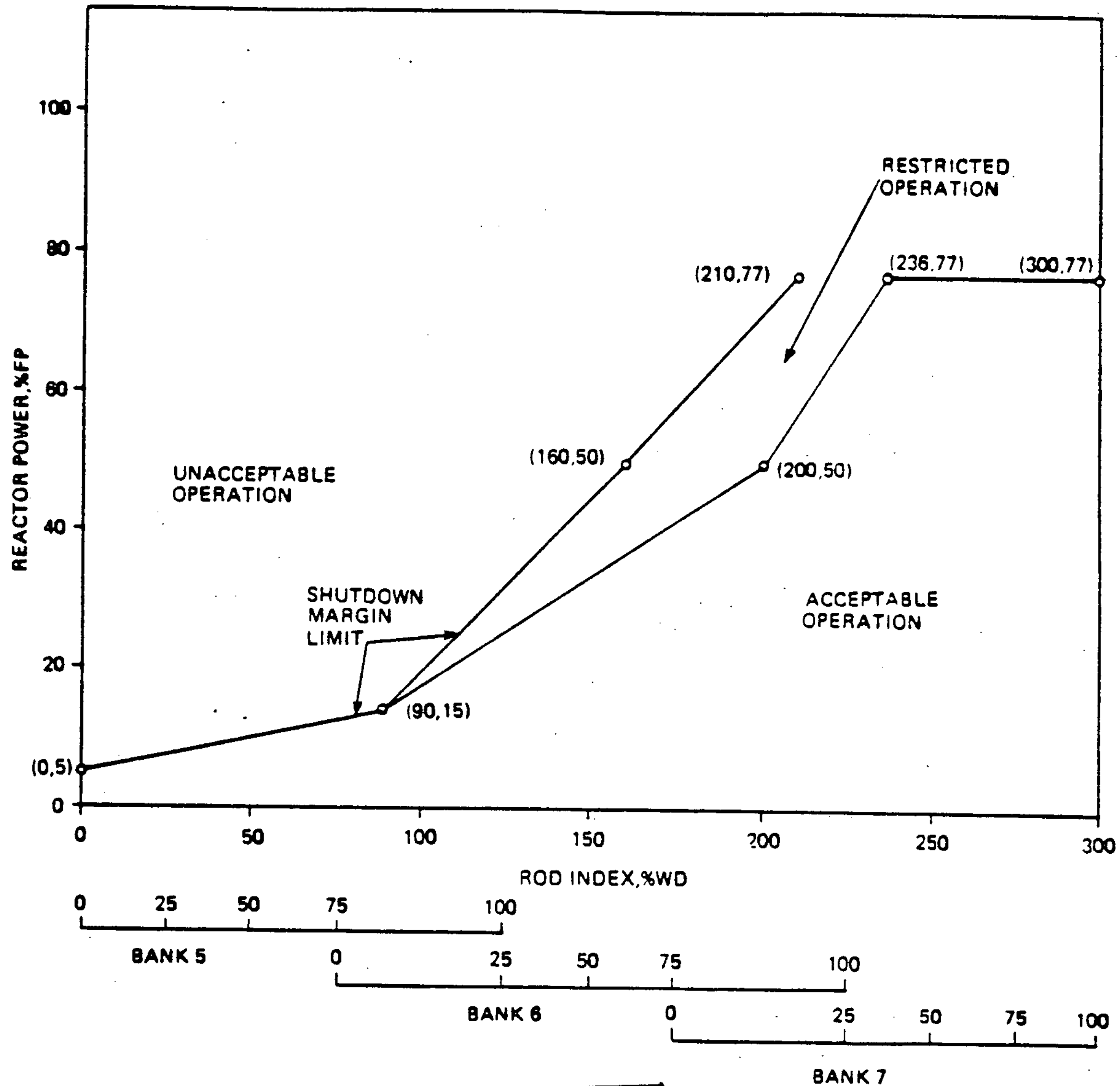
3.1.3.5
3.1.11
3.5.2.1.b
3.5.2.2.d.2.c
3.5.2.3
3.5.2.5.c

Control Rod Position Limits, 3 Pumps, 0 EFPD to EOC



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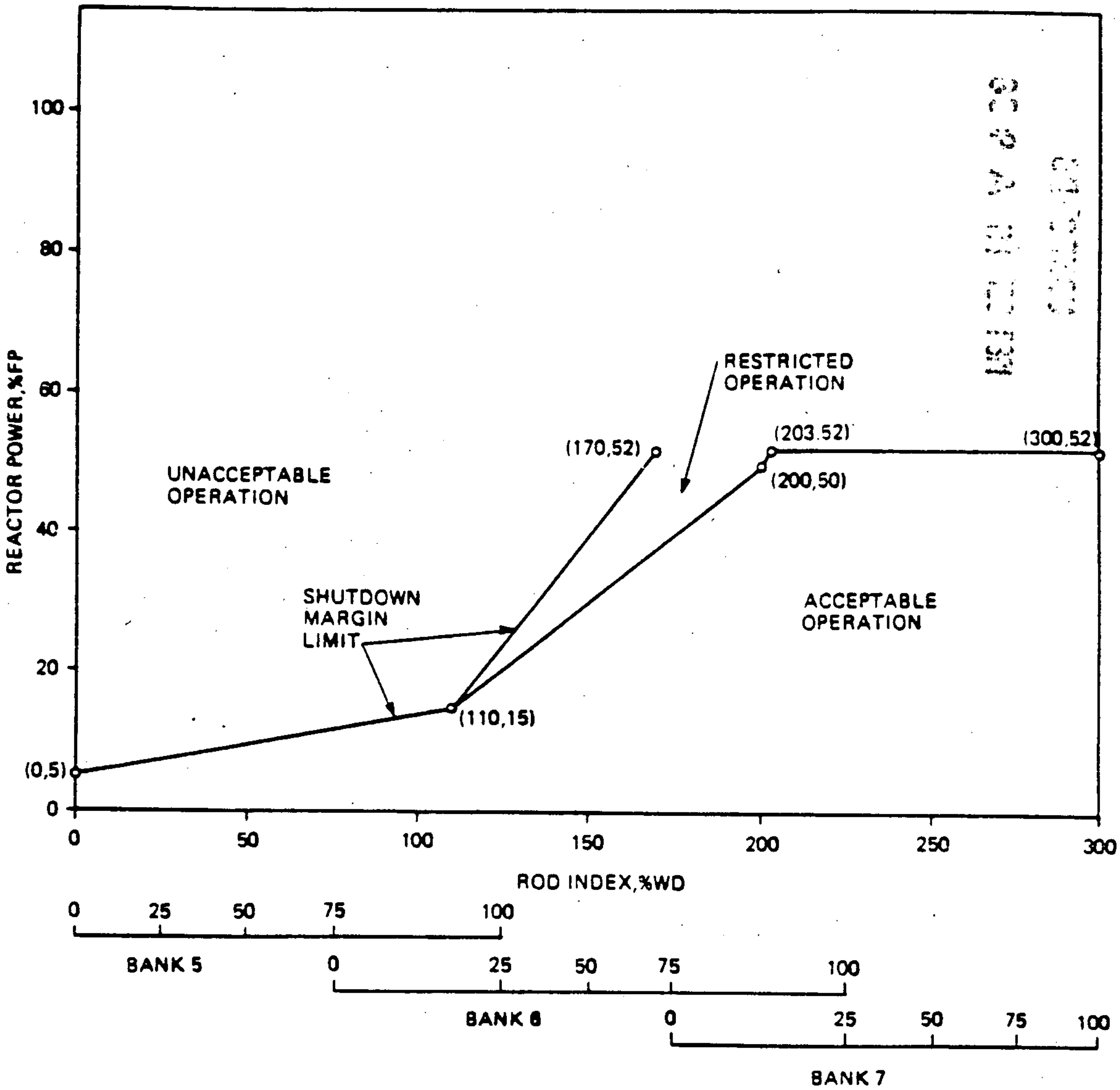
3.1.3.5
3.1.11
3.5.2.1.b
3.5.2.2.d.2.c
3.5.2.3
3.5.2.5.c

Control Rod Position Limits, 2 Pumps, 0 EFPD to EOC



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3.1.3.5
3.1.11
3.5.2.1.b
3.5.2.2.d.2.c
3.5.2.3
3.5.2.5.c