January 26, 1989

Docket Nos.: 50-269, 50-270
and 50-287

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 172,172, AND 169 TO FACILITY OPERATING LICENSES DPR-38, DPR-47, and DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TACS 66430/66431/66432)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 172,172, and 169 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3. These amendments consist of changes to the Station's Technical Specifications (TS) in response to your request dated September 3, 1987, as supplemented on February 27, September 9, and September 20, 1988.

The amendments revise the TS to replace the values of cycle-specific parameter limits with a reference to the Core Operating Limits Report which contains the values of those limits.

A copy of our Safety Evaluation is also enclosed. Notice of issuance of the enclosed amendments will be included in the Commission's bi-weekly Federal Register notice.

Sincerely,

/s/

Helen N. Pastis, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II

Enclosures:
1. Amendment No. 172 to DPR-38
2. Amendment No. 172 to DPR-47
3. Amendment No. 169 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

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MRood	HPastis:	DMatthews
12/5/88	12/12/88	1/26/88

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DATED: January 26, 1989

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE DPR-55 - Oconee Nuclear Station, Unit 3

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Facility Operating License No. DPR-38 filed by the Duke Power Company (the licensee) dated September 3, 1987, as supplemented on February 27, September 9, and September 20, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-38 is hereby amended to read as follows:

3.B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: January 26, 1989

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LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HRasmis	MYoung	DMatthews
12/5/88	12/12/88	1/18/89	1/26/88

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Facility Operating License No. DPR-47 filed by the Duke Power Company (the licensee) dated September 3, 1987, as supplemented on February 27, September 9, and September 20, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-47 is hereby amended to read as follows:

3.B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/ S /

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: January 26, 1989

OFFICIAL RECORD COPY

MY/ok w/ noted revisions to SE, SECY and date to TS bef issuance

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HPastis:	MYoung	DMatthews
12/5/88	12/12/88	1/18/89	1/26/88

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Facility Operating License No. DPR-55 filed by the Duke Power Company (the licensee) dated September 3, 1987, as supplemented on February 27, September 9, and September 20, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations, and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachments to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-55 is hereby amended to read as follows:

3.B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 169, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/s/

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects - I/II

Attachment:
Technical Specification
Changes

Date of Issuance: January 26, 1989

OFFICIAL RECORD COPY

LA:PDII-3	PM:PDII-3	OGC-WF	D:PDII-3
MRood	HPastis:	MYoung	DMatthews
12/5/88	12/12/88	1/16/88	1/26/88

MY wanted revision to SE; ck SECY + TS edits before issuance

ATTACHMENT TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 172

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the areas of change.

Remove Page	Insert Page
ii	ii
v-a	v-a
vii	vii
viii	viii
1.-6	1.-6
3.1-8	3.1-8
3.1-9	3.1-9
3.1-23	3.1-23
3.5-7	3.5-7
3.5-8	3.5-8
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-12	3.5-12

Remove Page	Insert Page
3.5-15	3.5-15
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3.5-24	
3.5-25	
3.5-26	
3.5-27	
3.5-28	
3.5-29	
	6-9.1

LIST OF FIGURES

LIST OF FIGURES (CONT'D)

1.8.8 VENTING

Venting is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is not provided or required during Venting. Vent, used in system names, does not imply a venting process.

1.8.9 MEMBER(S) OF THE PUBLIC

Member(s) Of The Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

1.8.10 UNRESTRICTED AREA

An Unrestricted Area shall be any area at or beyond the site boundary to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial institutional and/or recreational purposes.

1.9 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9. Plant operation within these core operating limits is addressed in individual specifications.

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%Δk/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 the acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

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 or 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% Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[1] and the small integrated Δk/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 relationship between primary coolant pressure and temperature will be maintained relative to the NDTT 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 the 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 the 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 for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

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 provided in the CORE OPERATING LIMITS REPORT.

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 of Table 3.5-1, 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 of Table 3.5-1, 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 for regulating rod position provided in the CORE OPERATING LIMITS REPORT 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 acceptable operating limits for reactor power imbalance provided in the CORE OPERATING LIMITS REPORT.

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 3.5.2-16) 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 (Units 1 and 2 only)
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	Xenon transient override
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% Δk/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% Δk/k is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% Δk/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% Δk/k ejected rod worth at rated power.

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 Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

(1) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(2) Power Imbalance Limits for Specification 3.5.2.6.

and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

6.9.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

DOCKET NOS. 50-269, 50-270 AND 50-287

1.0 INTRODUCTION

By letter dated September 3, 1987 (Ref. 1), as supplemented on February 27 (Ref. 5), September 9 (Ref. 6), and September 20, 1988 (Ref. 8), Duke Power Company (DPC or the licensee) proposed revisions to the Technical Specifications (TS) for Oconee Nuclear Station, Units 1, 2, and 3. The revisions to the TS would provide an alternative method for specifying the values of cycle-specific control rod position limit curves and axial imbalance (with imbalance being related to the axial flux difference between the top and bottom of the core) limit curves for affected TS. The purpose of the submittal was to obtain approval of the concept and the wording of affected TS, not to obtain approval of specific limiting curves for any Oconee Station current or future reload fuel cycle. The Oconee units are being used as the lead-plants to develop an acceptable alternative to specifying the values of cycle-specific parameters limits in the TS.

The elements of the DPC concept consist of three separate actions to revise the station's TS: (1) the addition of the definition of a named formal report called the Core Operating Limits Report (COLR) that includes the values of cycle-specific parameter limits that have been established using an NRC-approved methodology and consistent with all applicable limits of the safety analysis; (2) the addition of an administrative reporting requirement to submit the formal report on cycle-specific parameter limits to the Commission for information; and (3) the modification of individual TS to note that cycle-specific parameters shall be maintained within the limits provided in the defined formal report. There would be no prior staff review of the COLR. Thus, no staff amendment to the station's TS would be required for future reloads which alter these parameters. The COLR would be submitted to the NRC, the Regional Administrator, and the Resident Inspector upon issuance for each reload cycle.

This concept and procedure would only apply to those TS specified by TS 6.9, which are included in the COLR, and whose limit values and limit curves exhibit only nominal cycle-to-cycle variations. These limit values and limit curves may be generated, prior to implementation, as a function of cycle burnup or generated, after cycle operation begins, for implementation during a particular portion of the cycle. Revisions to TS that are not included in this concept, but which would be required by a core reload or other considerations, would be processed in accordance with the current license amendment procedures.

The analytical methods and procedures, which will be used by DPC, have been documented in topical reports (Refs. 2 and 3). These topical reports have been reviewed and approved by the NRC. DPC currently uses these analytical methods and procedures to calculate cycle-specific control rod position limit curves and axial imbalance limit curves for the Oconee Units' TS. The staff review of these cycle specific limit curves consists of confirmation that the new limits have been calculated using approved methods. The new limits are reviewed by the staff by also noting the trends from previous cycles and staff experience with other reloads. The proposed revisions to the form of the TS and associated COLR permit the staff to continue to trend cycle-specific limit values. It should be noted that, in some cases, audit calculations are performed for the staff by consultants to independently verify a vendor's or licensee's determination of TS limit values or curves (see Reference 4 for an example).

The staff evaluation of this DPC proposal regarding certain cycle-specific TS and the COLR follows.

2.0 EVALUATION

There are two regulations which must be considered in the staff's evaluation of the DPC proposal. These are 10 CFR 50.36 and 10 CFR 50.59. The first of these, 10 CFR 50.36, is directly concerned with TS. This regulation specifies that an applicant must submit proposed TS with an application for a license authorizing operation of a production or utilization facility. The licensee application must meet the guidance of this regulation. The second regulation, 10 CFR 50.59, provides guidance on making changes, performing tests, and performing experiments by the holder of a license authorizing operation of a production or utilization facility.

The pertinent sections of 10 CFR 50.36 for this evaluation are:

> 10 CFR 50.36(b)
>
> "... the technical specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to Section 50.34 ...."
>
> 10 CFR 50.36(c)
>
> "Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility ...."

10 CFR 50.36.(c)(3)

"Surveillance requirements are requirements relating to test, calibration or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met."

10 CFR 50.36(c)(5)

"Administrative controls are provisions relating to ... and reporting necessary to assure operation of the facility in a safe manner ...."

The current method of controlling reactor physics parameters to assure conformance to 10 CFR 50.36 is to specify the specific value(s) determined to be within specified acceptance criteria (usually the limits of the safety analyses) using an approved calculation methodology. The alternative contained in this DPC concept controls the values of cycle-specific parameters and assures conformance to 10 CFR 50.36, which calls for specifying the lowest functional performance levels acceptable for continued safe operation, by specifying the calculation methodology and acceptance criteria. This permits operation at any specific value determined by the licensee, using the specified methodology, to be within the acceptance criteria. The COLR will document the specific values of parameter limits resulting from DPC's calculations including any mid-cycle revisions to such parameter values.

The staff concludes that the DPC concept meets the intent of this regulation because 10 CFR 50.36 (1) does not specify that numerical values are required to be used for Limiting Conditions for Operation (LCO), (2) does not prohibit the specification of an LCO limit by referencing another document, and (3) does allow for the submittal of reports to assure the safe operation of the facility.

The regulation embodied in 10 CFR 50.59 allows the holder of a license authorizing operation of a production or utilization facility to do the following:

"... (i) make changes in the facility as described in the safety analysis report, (ii) make changes in the procedures as described in the safety analysis report, and (iii) conduct tests or experiments not described in the safety analysis report, without prior Commission approval, unless the proposed change, test or experiment involves a change in the technical specifications incorporated in the license or an unreviewed safety question."

The DPC proposal does not involve an unreviewed safety question because all limit curves in COLR will be generated by using analytical methods and procedures that are specified in the TS and have been reviewed and approved by the staff. The proposal does not involve a TS revision because the TS need not be revised when cycle-specific limit curves are changed. Only the COLR will be revised so that the appropriate COLR for the cycle in question will be applicable in accordance with the definition of the COLR which would be provided in the TS Definition Section. No prior approval of COLR would be required when cycle-specific revisions are necessary. Therefore, the staff concludes that the DPC concept, concerning the use and revisions to COLR, does not constitute a TS revision and is fully in conformance with the provisions of 10 CFR 50.59 and does not require the staff to issue a future license amendment.

In addition to meeting the requirements of the regulations specified in 10 CFR 50.36 and 10 CFR 50.59, the DPC concept on an alternative formulation of TS meets the characteristics listed below:

(1) The TS provide a limit value or curve which has been assumed to be an initial condition of, or is the result of, the plant safety analysis. Plant operation beyond the TS limit would place the plant in an unanalyzed state and thus result in an unreviewed safety question. Thus there is a need for the limit value or limit curves to appear in the TS.

(2) The limit values or limit curves are cycle-specific and are usually changed each cycle.

(3) The limit values or limit curves are calculated by methods and procedures that have been reviewed and approved by the NRC for use at the plant.

(4) The limit values or curves are the result of usually complex calculations. The NRC review of such limit values or curves consists of comparison with previous values and observing trends from cycle-to-cycle and plant-to-plant. In some instances, the staff performs audit calculations to independently confirm the accuracy of calculation of a particular parameter by a licensee or vendor.

These characteristics recognize that (1) there is a need to provide the limit value or curves in the TS, (2) the staff performs only a limited review of the limit values or curves, (3) the limit values or curves may change each cycle, and (4) approved methods and procedures are used to calculate the limit values or curves. Thus the staff concludes that these characteristics provide an acceptable basis for determining which TS should be treated by DPC's alternate formulation of TS.

In sum, the staff finds the DPC approach acceptable as follows:

(1) The limit values or limit curve remain quantitatively identified in TS as limits on plant operation.

(2) A TS revision is not required each cycle. This assumes that there are no other TS revision for the cycle or unreviewed safety questions. The implication of this is that a significant savings in resources can be made by the NRC in not having to review and issue a license amendment for a reload in which only minor changes to cycle-specific parameters are needed. There is also a significant resource savings to the licensee in not having to support a reload licensing effort.

(3) The report containing the cycle specific parameters will still be available to the staff, after its implementation at a plant, for use in trending parameters.

(4) Licensees no longer need to operate their plants with restrictive bounding limit values or curves. Use of the Duke concept will allow more flexible and optimum cycle design and operation.

(5) Licensees will perform a safety analysis for each cycle regardless of whether or not the alternative TS formulation is used.

(6) The new alternative forms of the TS are considered to be improvements to the TS and thus in line with the Commission's stated policy for improving TS (see 52FR3788 of February 6, 1987).

(7) The DPC concept recognizes the limited nature of the NRC's review of some of the more complex cycle dependent TS and the trivial changes encountered, in most cases, in some TS limit values.

3.0 TECHNICAL SPECIFICATIONS

The acceptable wording for the revisions to the affected TS for the Oconee station are as follows:

(1) Definition

1.9 CORE OPERATING LIMITS REPORT

The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9. Plant operation within these core operating limits is addressed in individual specifications.

(2) Technical Specification 3.1.3.5 and Bases

3.1.3.5

Except for physics tests ... during the approach to criticality. The regulating rods shall then be positioned within the acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

Bases

(The Duke Power Company's submittal of September 3, 1987 is acceptable with regard to the wording of the last sentence of the last paragraph on Technical Specification Page 3.1-9.)

(3) Technical Specification 3.1.11 Bases

(The Duke Power Company's submittal of September 3, 1987 is acceptable with regard to the wording of the second paragraph of the Bases on Technical Specification Page 3.1-23.)

(4) Technical Specification 3.5.2.2.d.2.c

3.5.2.2.d.2.c

Position the remaining rods ... limits of Specification 3.5.2.2.a and within the acceptable operating rod position withdrawal/insertion limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT.

(5) Technical Specification 3.5.2.3

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 provided in the CORE OPERATING LIMITS REPORT.

(6) Technical Specification 3.5.2.5.c

3.5.2.5.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 the acceptable operating limits for regulating rod position provided in the CORE OPERATING LIMITS REPORT for the particular number of operating reactor coolant pumps (4, 3, 2).

(7) Technical Specification 3.5.2.6

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 acceptable operating limits for reactor power imbalance provided in the CORE OPERATING LIMITS REPORT.

(8) Technical Specification 3.5.2.6 Bases

(The Duke Power Company's submittal of September 3, 1987 is acceptable with regard to the wording of the first sentence of the second paragraph of Technical Specification Page 3.5-11.)

(9) Technical Specification 3.5.2 Bases

(The Duke Power Company's submittal of September 3, 1987 is acceptable with regard to the wording of (1) the first sentence of the first paragraph and (2) the last paragraph of Technical Specification Page 3.5-12.)

(10) Technical Specification 6.9

6.9 CORE OPERATING LIMITS REPORT

Specification

6.9.1

Core operating limits shall be established prior to each reload cycle, or prior to any remaining part of a reload cycle, for the following:

(1) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(2) Power Imbalance Limits for Specification 3.5.2.6.
and shall be documented in the CORE OPERATING LIMITS REPORT.

6.9.2

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically:

(1) DPC-NE-1002A, Reload Design Methodology II, October 1985.

(2) NFS-1001A, Reload Design Methodology, April 1984.

6.9.3

The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.4

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

The staff has reviewed the DPC's alternative method for specifying the values of cycle-specific control rod position limit curves and axial imbalance limit curves for affected TS. Affected TS (control rod position limit curves and axial imbalance limit curves) would no longer contain the numerical values for these limits but would reference the values in a COLR. The COLR will be a defined term, and its requirements will be specified by Specification 6.9. Based on the evaluation discussed above, the staff concludes that DPC's alternative method for formulating cycle-specific TS is acceptable providing that the formulation and wording of Section 3 for the definition of the COLR, affected Specifications, and Specification 6.9 are followed.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that these amendments involve no significant hazards consideration, and there has been no public comment on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (53 FR 50325) on December 14, 1988, and consulted with the state of South Carolina. No public comments were received, and the state of South Carolina did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from Hal B. Tucker (DPC) to USNRC, dated September 3, 1987.

2. "Reload Design Methodology II," DPC-NE-1002A, Duke Power Company, October 1985.

3. "Reload Design Methodology," NFS-1001A, Duke Power Company, April 1984.

4. "Implementation of LOCA Linear Heat Generation Limits by Restricting Control Rod Operation," BNL Memorandum from P. Neogy to J. F. Carew, dated March 26, 1986.

5. Letter from Hal B. Tucker (DPC) to USNRC, dated February 27, 1988.

6. Letter from Hal B. Tucker (DPC) to USNRC, dated September 9, 1988.

7. Letter from D. B. Matthews (NRC) to H. B. Tucker (DPC), dated August 23, 1988.

8. Letter from Hal B. Tucker (DPC) to USNRC, dated September 20, 1988.

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