December 15, 1989

Dockets 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 AMENDMENTS NOS.180 , 180, AND 177 TO FACILITY OPERATING LICENSES DPR-38, DPR-47, and DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TACS 75074/75075/74887)

The Nuclear Regulatory Commission has issued the enclosed Amendments Nos. 180,180, and 177 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 Technical Specifications (TSs) in response to your request dated September 25, 1989, as supplemented October 18, 1989.

These amendments revise the TSs to account for minor changes in power peaking and control rod worths resulting from the Oconee Unit 3 core reload. In addition, the use of the VIPRE thermal hydraulic code is referenced and all specifications associated with two reactor coolant pump operations are deleted.

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

Sincerely,

/s/

Leonard A. Wiens, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.180 to DPR-38
2. Amendment No.180 to DPR-47
3. Amendment No.177 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

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Mr. H. B. Tucker
Duke Power Company

Oconee Nuclear Station
Units Nos. 1, 2 and 3

cc:
Mr. A. V. Carr, Esq.
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

J. Michael McGarry, III, Esq.
Bishop, Cook, Purcell & Reynolds
1400 L Street, N.W.
Washington, D.C. 20005

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Division
Suite 525
1700 Rockville Pike
Rockville, Maryland 20852

Manager, LIS
NUS Corporation
2536 Countryside Boulevard
Clearwater, Florida 34623-1693

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 610
Seneca, South Carolina 29678

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Heyward G. Shealy, Chief
Bureau of Radiological Health
South Carolina Department of Health
and Environmental Control
2600 Bull Street
Columbia, South Carolina 29201

Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

County Supervisor of Oconee County
Walhalla, South Carolina 29621

Mr. Paul Guill
Duke Power Company
Post Office Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Mr. Alan R. Herdt, Chief
Project Branch #3
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30323

Ms. Karen E. Long
Assistant Attorney General
N. C. Department of Justice
P.O. Box 629
Raleigh, North Carolina 27602

DATED: December 15, 1989

AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 180 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 177 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.180
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 25, 1989, as supplemented October 18, 1989, 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 attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-38 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 180, 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

Kahtan N. Jabbour for

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

Attachment:
Technical Specification
Changes

Date of Issuance: December 15, 1989

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. 180
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 25, 1989, as supplemented October 18, 1989, 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 attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 180, 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

Kalt N Jabbour for

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

Attachment:
Technical Specification
Changes

Date of Issuance: December 15, 1989

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. 177
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 25, 1989, as supplemented October 18, 1989, 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 attachment to this license amendment, and Paragraph 3.B. of Facility Operating License No. DPR-55 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 177, 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

Kahtan N. Jabbour for

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

Attachment:
Technical Specification
Changes

Date of Issuance: December 15, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 180

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 177

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 Pages	Insert Pages
2.1-2	2.1-2
2.1-3	2.1-3
2.1-5	2.1-5
2.3-1	2.3-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-4	2.3-4
2.3-6	2.3-6
2.3-7	2.3-7
3.1-1	3.1-1
3.1-2	3.1-2
3.5-9	3.5-9
6.9-1	6.9-1

The curve presented in Figure 2.1-1(3) represents the conditions at which the minimum allowable DNBR is predicted to occur for the limiting combination of thermal power and number of operating reactor coolant pumps. This curve is based upon the design nuclear peaking factors (4,6,7):

$$F_{\Delta H}^{N} = 1.714$$

$$F_{Z}^{N} = 1.50$$

Since power peaking is not a directly measurable quantity, DNBR limited power peaks and fuel melt limited power peaks are separately correlated to measurable reactor power and power imbalance. The reactor power imbalance limits,

Figure 2.1-2(5), define the values of reactor power as a function of axial imbalance that correspond to the more restrictive of two thermal limits - MDNBR equal to the DNBR limit or the linear heat rate equal to the centerline fuel melt limit.

The core protection safety limits are based on an RCS flow less than or equal to 385,440 gpm (4 pump operation). Three pump operation is analyzed assuming 74.7 percent of four pump flow. The maximum thermal power for three pump operation is 84.9 percent (Figure 2.1-2) due to a power level trip produced by the flux/flow ratio (74.7 percent flow x 1.07 = 79.9 percent power = 84.9 percent power adding the maximum calibration and instrument error).

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle cooled by Pressurized Water, BAW-10000, March 1970.

(2) Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, August 1981.

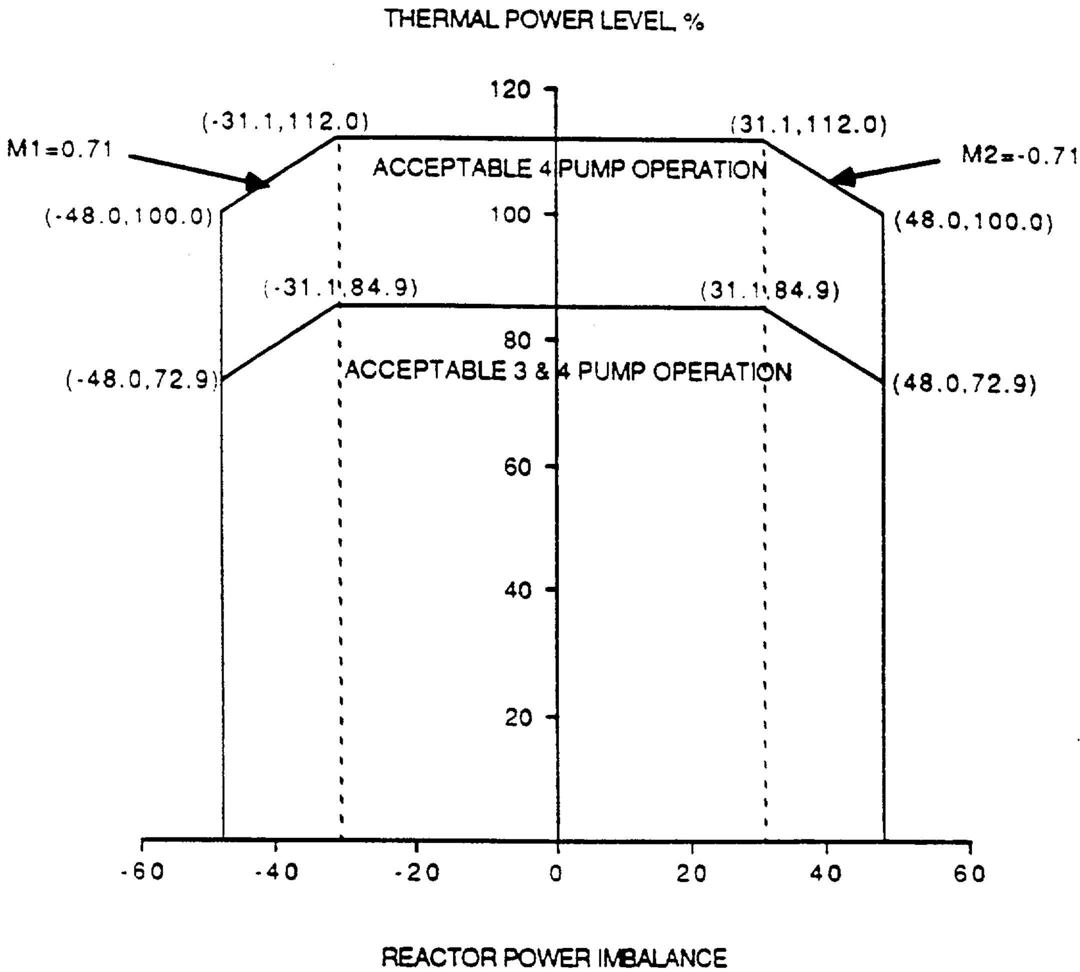
(3) Oconee Unit 3, Cycle 7 - Reload Report, DPC-RD-2001, Rev. 1, Duke Power Company, July 1982.

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

(5) Oconee Unit 2, Cycle 7 - Reload Report, DPC-RD-2002, Duke Power Company, September 1983.

(6) Oconee Nuclear Station Core Thermal Hydraulic Methodology using VIPRE-01, DPC-NE-2003A, Duke Power Company, July 1989.

(7) Oconee Nuclear Station Reload Design Methodology, NFS-1001A, Duke Power Company, April 1984.



CORE PROTECTION SAFETY LIMITS UNITS 1, 2, AND 3

Figure 2.1-2

OCONEE NUCLEAR STATION

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.

Specification

The reactor protective system trip setpoints and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and Figure 2.3-2.

The pump monitors shall produce a reactor trip when a loss of two pumps occurs and reactor power level is greater than 0.0% of rated power.

Bases

The reactor trip setpoints for reactor protective system (RPS) instrumentation are given in Table 2.3-1. The trip setpoints have been selected to ensure that the core and reactor coolant system are prevented from exceeding their safety limits. The various reactor trip circuits automatically open the reactor trip breakers whenever a parameter monitored by the RPS deviates from an allowed range. The RPS consists of four instrument channels for redundancy. The plant safety analyses are based on the trip setpoints given in Table 2.3-1 plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, a reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in the trip setpoint due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis. (1)

Overpower Trip Based on Flow and Imbalance

Following the loss of one or more reactor coolant pumps, the core is prevented from violating the minimum DNBR criterion by a reactor trip initiated by exceeding the allowable reactor power to reactor coolant flow (flux/flow) ratio setpoint. Loss of one or more reactor coolant pumps is also detected by the pump monitors. The power level trip produced by the flux/flow ratio provides DNB protection for all modes of pump operation.

The power level trip setpoint produced by the flux/flow ratio provides both high power level and low flow protection. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible flow rate. Typical power level and flow rate combinations for different pump situations are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is 107% and reactor flow rate is 100%, or flow rate is 93.46% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is 79.9% and reactor flow rate is 74.7% or flow rate is 70.09% and power level is 75%.

   The analysis to determine the flux/flow setpoint accounts for calibration and instrument errors and the variation in RC flow in such a manner as to ensure a conservative setpoint. Statistical methods are used to determine the combined effects of calibration and instrument uncertainties with the final string uncertainties used in the analysis corresponding to the 95/95 tolerance limits.

The reactor power imbalance (power in the top half of the core minus the power in the bottom half) reduces the power level trip produced by the flux/flow ratio as shown in Figure 2.3-2. The flux/flow ratio reduces the power level trip and associated power-imbalance boundaries by 1.07% for a 1% flow reduction. The power-imbalance boundaries shown in Figure 2.3-2 are established to prevent fuel thermal limits, DNBR and centerline fuel melt limits, from being exceeded.

Pump Monitors

The pump monitors trip the reactor due to the loss of reactor coolant pump(s) to ensure the DNBR remains above the minimum allowable DNBR. The pump monitors provide redundant trip protection of DNB; tripping the reactor on a signal diverse from that of the flux/flow trip. The pump monitors also restrict the power level depending on the number of operating reactor coolant pumps.

Reactor Coolant System Pressure

During a startup accident from low power or a slow rod withdraw from high power, the reactor coolant system (RCS) high pressure setpoint is reached before the nuclear overpower trip setpoint. The high RCS pressure trip setpoint (2355 psig) ensures that the pressure remains below the safety limit (2750 psig) for any design transient. (2) The low pressure (1800 psig) and variable low pressure (11.14 T_{out} - 4706) trip setpoints shown in Figure 2.3-1 ensure that the minimum DNBR is greater than or equal to minimum allowable DNBR for those accidents that result in a reduction in pressure. (3,4) The limits shown in Figure 2.3-1 bound the pressure-temperature curves calculated for 4 and 3 pump operation.

Accounting for calibration and instrumentation errors, the safety analyses used a variable low RCS pressure trip setpoint of (11.14 T_{out} - 4756).

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setpoint (618°F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analysis used a trip setpoint of 620°F.

Reactor Building Pressure

The high reactor building pressure trip setpoint (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

Shutdown Bypass

In order to startup the reactor and to be able to perform control rod drive tests and zero power physics tests (see Technical Specification 3.1.9), there is provision for bypassing certain segments of the reactor protective system (RPS). The RPS segments which can be bypassed are given in Table 2.3-1. Two conditions are imposed when the RPS is bypassed:

1. By administrative control the nuclear overpower trip setpoint is reduced to a value of $\leq$ 5.0% of rated power.

2. The high reactor coolant system pressure trip setpoint is automatically lowered to 1720 psig.

The high RCS pressure trip setpoint is reduced to prevent normal operation with part of the RPS bypassed. The reactor must be tripped before the bypass is initiated since the high pressure trip setpoint is lower than the normal low pressure trip setpoint (1800 psig).

The overpower trip setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced when performing physics tests. If no reactor coolant pumps are operating, sufficient natural circulation would be available to remove 5.0% of rated power.(5)

REFERENCES

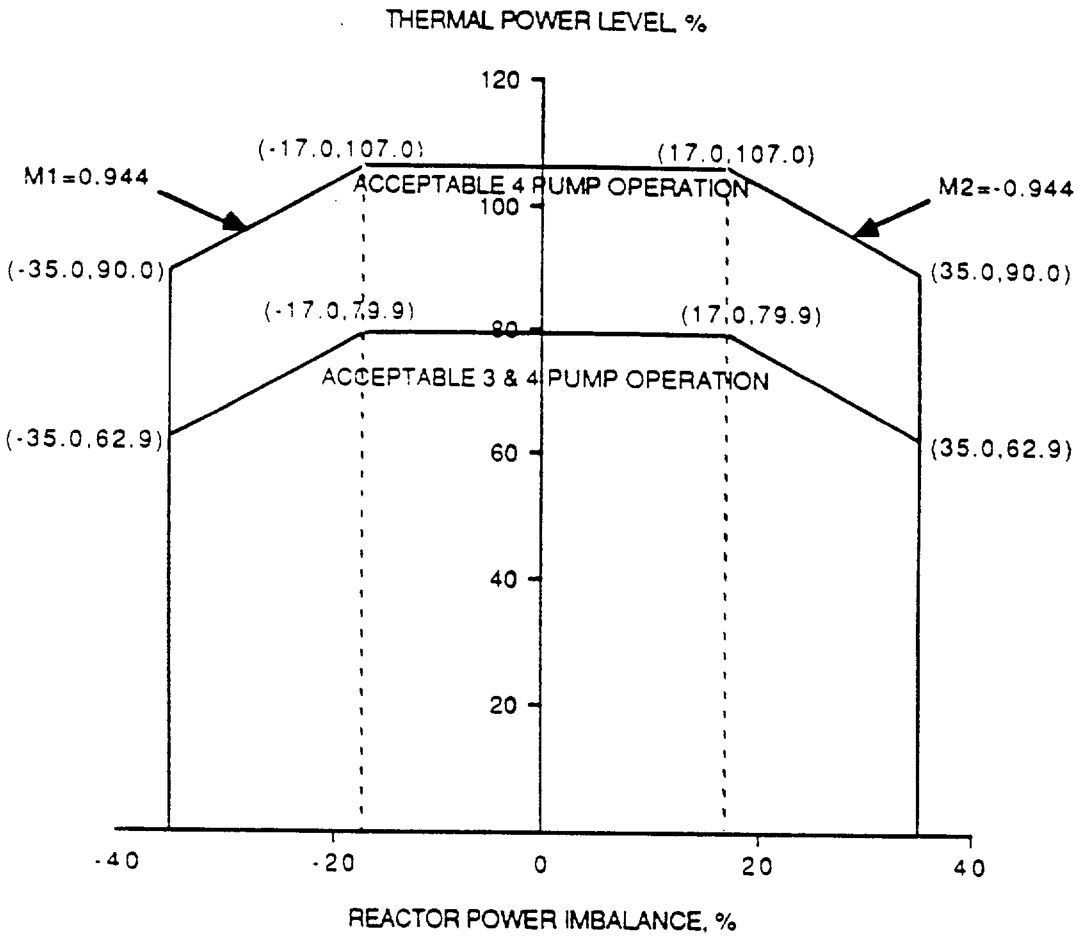
(1) FSAR, Section 15.3

(2) FSAR, Section 15.2

(3) FSAR, Section 15.7

(4) FSAR, Section 15.8

(5) FSAR, Section 15.6



PROTECTIVE SYSTEM MAXIMUM ALLOWABLE SETPOINTS UNITS 1, 2, AND 3

Figure 2.3-2

OCONEE NUCLEAR STATION

TABLE 2.3-1

Reactor Protective System Trip Setting Limits

	RPS Trip	RPS Trip Setpoint	Shutdown Bypass
1.	Nuclear Overpower	105.5% Rated Power	5.0% Rated Power (1)
2.	Flux/Flow/Imbalance	1.07	Bypassed
3.	Pump Monitors	> 0% Rated Power loss of two pumps	Bypassed
4.	High Reactor Coolant System Pressure	2355 psig	1720(2)
5.	Low Reactor Coolant System Pressure	1800 psig	Bypassed
6.	Variable Low Reactor Coolant System Pressure	P (psig) = (11.14 T_{out} - 4706)(3)	Bypassed
7.	High Reactor Coolant Temperature	618°F	618°F
8.	High Reactor Building Pressure	4 psig	4 psig

(1) Administratively controlled reduction set only during reactor shutdown.

(2) Automatically set when other segments of the RPS are bypassed.

(3) T_{out} is in degrees Fahrenheit (°F).

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system components which must be met to ensure safe reactor operation.

Specification

3.1.1 Operational Components

a. Reactor Coolant Pumps

1. Whenever the reactor is critical, one and two pump operation shall be prohibited, single-loop operation shall be restricted to testing, and other pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

2. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one low pressure injection pump is circulating reactor coolant.

b. Steam Generator

1. One steam generator shall be operable whenever the reactor coolant average temperature is above 250°F.

c. Pressurizer Safety Valves

1. All pressurizer code safety valves shall be operable whenever the reactor is critical.

2. At least one pressurizer code safety valve shall be operable whenever all reactor coolant system openings are closed, except for hydrostatic tests in accordance with the ASME Section III Boiler and Pressure Vessel Code.

Bases

A reactor coolant pump or low pressure injection pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One low pressure injection pump will circulate the equivalent of the reactor coolant system volume in one-half hour or less. (1)

The low pressure injection system suction piping is designed for 300°F and 370 psig; thus the system with its redundant components can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident at hot shutdown. (5) The pressurizer code safety valve lift setpoint shall be set at 2500 psig ±1% allowance for error and each valve shall be capable of relieving 300,000 lb/hr of saturated steam at a pressure no greater than 3% above the set pressure.

REFERENCES

(1) FSAR, Section 6.3.3.2, and Tables 5.3-1, 5.4-2, 5.4-3, 5.4-6, 5.4-7, 5.4-8 and 6.3-2.

(2) FSAR, Sections 5.4.7-1 and 9.3.3.2.3.

(3) FSAR, Sections 5.4.7.4 and 6.3.3.2

(4) FSAR, Sections 5.2.3.10.4 and 5.4.6.

(5) FSAR, Sections 5.2.3.7 and 15.2.3.

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).

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.

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 REPORTS.

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.

(3) DPC-NE-2003A, Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01, July 1989.

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.180 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO.180 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO.177 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

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

1.0 INTRODUCTION

By letter dated September 25, 1989, Duke Power Company, the licensee for operation of Oconee Nuclear Station (ONS), Units 1, 2 and 3, requested an amendment to the ONS Units 1, 2 and 3 Technical Specifications (TSs) to support operation of Unit 3 at full rated power during Cycle 12. Specifically, the TS change request is to (1) revise the flux/flow/imbalance safety limits in Figure 2.1-2 by reducing the allowable thermal power level and imbalance for the three- and four-reactor coolant pump operations, (2) remove specifications associated with two-pump operation from the TSs including Figures 2.1-2 and 2.3-2, and Table 2.3-1, and (3) minor editorial changes to reflect the use of the VIPRE code for the core thermal hydraulic analysis and revise the radial peaking factor from 1.71 to 1.714.

2.0 EVALUATION

In support of the Unit 3, Cycle 12 operation, the licensee, in Attachment 3 of the September 25, 1989 letter, provided the Oconee Unit 3, Cycle 12 Reload Report, DPC-RD-2014. The Cycle 12 core will use 52 fresh Batch 14 Mark-B8 fuel assemblies, and 125 fuel assemblies from the previous cycles with the majority being Batches 12B and 13 fuel assemblies, i.e., Mark-B5Z and Mark-B7 assemblies. Forty-four of the 52 Batch 14 assemblies have burnable poison rod assemblies (BPRAs) inserted. Thirty-six (36) of the BPRAs are new, and the remaining 8 are once burned.

The Batch 14 Mark-B8 assemblies are similar to the Mark-B7 fuel previously reloaded into Cycle 11. Both fuel designs have intermediate zircaloy spacer grids. New features for Mark-B8 include a debris fretting resistant fuel rod design, which utilizes a lengthened solid lower end plug extending below the bottom end grid, and a slightly reduced fuel rod prepressurization level to compensate for the reduction in plenum volume. The Mark-B8 fuel also has higher initial enrichment than that loaded in the previous cycles. Because of variation in the shuffle pattern and changes in the radial flux and burnup distributions, the physics parameters and the ejected and stuck rod worths are different between Cycles 11 and 12. These physics characteristics were calculated with the approved CASMO-2 based reload design methods. The resulting rod worth and shutdown margin are shown in Tables 5-1 and 5-2 of the Reload Report.

With the recent approval of Topical Report DPC-NE-2003A, "Oconee Nuclear Station Core Thermal Hydraulic Methodology Using VIPRE-01," the Cycle 12 thermal hydraulic analysis was done with the VIPRE-01 code. Of the 177 fuel assemblies in the core, only 4 assemblies have Inconel spacer grids and the remainder (Mark-BZ design) have zircaloy spacer grids. The Mark-BZ fuel assemblies have a slightly higher pressure drop than the other assemblies as a result of the increased flow resistance of the zircaloy spacer grids. The Cycle 12 transition core was conservatively analyzed at the limiting thermal design conditions for the limiting two-pump coastdown transient and a flux to flow trip setpoint of 1.07. The resulting minimum departure from nucleate boiling ratio (DNBR) is greater than the BWC critical heat flux correlation limit of 1.18.

The licensee performed the reload analysis using the approved reload design methodology described in DPC-NE-1002A, "Oconee Nuclear Station Reload Design Methodology II." The flux/flow/imbalance safety limit change is necessary because of the minor changes in the power peaking and control rod worth. The changes in safety limits also affect the limiting safety system settings. The flux/flow/imbalance safety limit change is a result of maneuvering analysis on fuel depletion, integral rod worth, and power maneuver. For the four-pump operation, the maximum power imbalance is reduced from the current value of 49.5 percent to 48 percent. For three-pump operation, the maximum allowable power level is reduced from 88.07 percent to 84.9 percent. The calculation of the allowable power level for the three-pump operation is based on an assumed reactor coolant system flow rate of 74.7 percent of the rated flow, a flux/flow trip setpoint of 1.07, and a power measurement uncertainty of 5 percent. This method is the same as that for the previous cycles except that the power uncertainty of 5 percent is obtained using the square root of sum of the squares method to account for the uncertainties associated with heat balance error, TS allowance for calibration of the excore detectors to the heat balance, transient nuclear instrument error, and an allowance for the uncertainty of the flux/flow imbalance trip function hardware. Since the error adjustment of the flux/flow/imbalance safety limits would result in a setpoint of 79.9 percent, the same as the current Reactor Protection System (RPS) maximum allowable setpoint, the RPS allowable setpoint in Figure 2.3-2 was unchanged.

The analysis employs analytical techniques and design bases established in reports that were previously accepted by NRC. All of the accidents analyzed in the Final Safety Analysis Report (FSAR) have been reviewed for Cycle 12 operation. A comparison of the key parameters for accident analysis, such as Doppler and moderator coefficients, rod worth and boron reactivity worth, is provided in Table 7.1 of the reload report. A review of these key parameters by the licensee has determined that the Cycle 12 characteristics were conservative compared to those analyzed for previous cycles. Therefore, no new accident analyses were performed. The TS modifications required for Cycle 12 operation are justified.

Other TS changes are essentially editorial changes. The change of the enthalpy rise factor from 1.71 to 1.714 is to be consistent with the actual value used in the analysis. Because the licensee's proposal does not include two-pump operation, TS items associated with two-pump operations are to be deleted. The Cycle 12 reload report also indicated that the figures for operating limits on rod index and axial power imbalance have been removed from the TSs and included in the cycle-specific Core Operating Limits Report (COLR). This removal was approved by NRC in Amendments Nos. 172, 172, and 169 to the operating licenses for ONS, Units 1, 2 and 3, issued January 26, 1989.

Since Oconee Units 1, 2 and 3 have common TSs, the changes on Unit 3 also affect Units 1 and 2. The September 25, 1989, letter indicated that changes which affect Units 1 and 2 will be implemented upon Unit 3 Cycle 12 startup. Even though there is no supporting analysis to justify the TS changes for Units 1 and 2, we find that they are acceptable because (1) the flux/flow/imbalance safety limit change is in a more restrictive, conservative direction, and (2) other changes are only editorial changes.

The NRC staff has reviewed the licensee's request for TS changes to support the Unit 3, Cycle 12 operation. We have found that the TS changes are either editorial or supported by the reload safety analysis performed with approved methods, and therefore are acceptable. The TS changes are also acceptable for Units 1 and 2.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. We have 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.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 46146) on November 1, 1989, and consulted with the State of South Carolina. No public comments were received, and the State of South Carolina did not have any comments.

We have 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.

Principal Contributor: Y. Hsii, SRXB/DST

Dated: December 15, 1989