LICENSE AUTHORITY FILE COPY DO NOT REMOVE

January 5, 1993

Posted

Amdt. 197 to DPR-47

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

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

Dear Mr. Hampton:

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 (TAC NOS. M84088, M84089, AND M84090)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 197, 197 , and 194 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 13, 1992, as supplemented December 1, 1992.

The amendments delete cycle-dependent core operating limits from the TS to allow 10 CFR 50.59 reviews for future core reloads for all Oconee units. These limits will be relocated in the Core Operating Limits Report (COLR) in accordance with the guidance provided in Generic Letter 88-16.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance 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. 197 to DPR-38
2. Amendment No. 197 to DPR-47
3. Amendment No. 194 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

OFC	:PDII-3/LA	:PDII-3/PM	:OGC	:PDII-3/D
NAME	:L.BERRY	:L.WIENS	:[illegible]	:D.MATTHEWS
DATE	:12/15/92	:12/15/92	:12/21/92	:1/5/93

OFFICIAL RECORD COPY
File Name: **G:\OCONEE\OCO84088.AMD**

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

January 5, 1993

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

Mr. J. W. Hampton
Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
Seneca, South Carolina 29679

Dear Mr. Hampton:

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M84088, M84089, AND M84090)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 197, 197 , and 194 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 13, 1992, as supplemented December 1, 1992.

The amendments delete cycle-dependent core operating limits from the TS to allow 10 CFR 50.59 reviews for future core reloads for all Oconee units. These limits will be relocated in the Core Operating Limits Report (COLR) in accordance with the guidance provided in Generic Letter 88-16.

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

Sincerely,

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

Enclosures:
1. Amendment No. 197 to DPR-38
2. Amendment No. 197 to DPR-47
3. Amendment No. 194 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

Mr. J. W. Hampton
Duke Power Company

Oconee Nuclear Station

cc:

Mr. A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

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

Manager, LIS
NUS Corporation
2650 McCormick Drive, 3rd Floor
Clearwater, Florida 34619-1035

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, NW. 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. M. E. Patrick
Compliance
Duke Power Company
Oconee Nuclear Site
P. O. Box 1439
Seneca, South Carolina 29679

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
North Carolina Department of
Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. R. L. Gill, Jr.
Licensing
Duke Power Company
P. O. Box 1006
Charlotte, North Carolina 28201-1006

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. 197
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 July 13, 1992, as supplemented December 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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. 197, 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

DBMatthews

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: January 5, 1993

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. 197
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 July 13, 1992, as supplemented December 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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. 197, 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

DB Matthews

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: January 5, 1993

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. 194
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 July 13, 1992, as supplemented December 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 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:

<u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 194, 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

DB Matthews

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: January 5, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 194

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
vii	vii
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	---
2.1-4	---
2.1-5	---
2.3-2	2.3-2
2.3-3	2.3-3
2.3-5	2.3-5
3.2-1	3.2-1
3.2-2	3.2-2
3.3-3	3.3-3
3.3-6	3.3-6
6.9-1	6.9-1

LIST OF FIGURES

2 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow during power operation of the plant.

Objective

To maintain the integrity of the fuel cladding.

Specification

The maximum local fuel pin centerline temperature shall be less than 5080 - $(6.5\times10^{-3})\times$(Burnup, MWD/MTU) °F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

The DNBR shall be maintained greater than the correlation limits of 1.3 for BAW-2 and 1.18 for BWC. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits as specified in the Core Operating Limits Report.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions and anticipated transients. This is accomplished by operating within the nuclear boiling heat transfer regime where the heat transfer coefficient is large and the cladding temperature is only slightly greater than the coolant temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation, but neutron power and reactor coolant pressure and temperature can be related to DNB using a critical heat flux (CHF) correlation. The local DNB heat flux ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual local heat flux, is indicative of the margin to DNB.

The BAW-2 and BWC CHF correlations[1,2] have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The Variable Low RCS Pressure Protective Limits presented in the Core Operating Limits Report represent 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 provided in the Core Operating Limits Report.

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 Axial Power Imbalance Protective Limits presented in the Core Operating Limits Report 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 provided with the Axial Power Imbalance Protective Limits in the Core Operating Limits Report.

References

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

(2) Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143P, Part 2, August 1981.

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. For example, typical power level and flow rate combinations for different pump situations are as follows (actual values are given in the Core Operating Limits Report):

1. Assuming a flux/flow ration of 1.07, a reactor 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 the Axial Power Imbalance RPS Maximum Allowable Setpoints figure in the Core Operating Limits Report. The flux/flow ratio reduces the power level trip and associated power-imbalance boundaries to account for any reduction in RCS flow. The power-imbalance boundaries shown in the Axial Power Imbalance RPS Maximum Allowable Setpoints figure in the COLR 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 trip setpoints shown in the Variable Low RCS Pressure RPS Maximum Allowable Setpoints figure in the Core Operating Limits Report 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 the Variable Low RCS Pressure RPS Maximum Allowable Setpoints figure in the Core Operating Limits Report bound the pressure-temperature curves calculated for 4 and 3 pump operation.

The safety analyses use a variable low RCS pressure trip setpoint which accounts for calibration and instrumentation uncertainties.

Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618°F) shown in the Variable Low RCS Pressure RPS Maximum Allowable Setpoints figure in the Core Operating Limits Report 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).

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	Axial Power Imbalance RPS Maximum Allowable Setpoints in the Core Operating Limits Report	Bypassed
3.	Pump Monitors	At power operation >2.0% Rated Power and 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	Variable Low RCS Pressure RPS Maximum Allowable Setpoints in the Core Operating Limits Report	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.

3.2 HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS

Applicability

Applies to the high pressure injection and the chemical addition systems.

Objective

To provide for adequate boration under all operating conditions to assure ability to bring the reactor to a cold shutdown condition.

Specification

The reactor shall not be critical unless the following conditions are met:

3.2.1 Two high pressure injection pumps per unit are operable except as specified in 3.3.

3.2.2 One source per unit of concentrated soluble boric acid in addition to the borated water storage tank is available and operable.

This source will be the concentrated boric acid storage tank with the volume and boron concentration within the limits of the Core Operating Limits Report with a temperature at least 10°F above the crystallization temperature. System piping and valves necessary to establish a flow path from the tank to the high pressure injection system shall be operable and shall have the same temperature requirement as the concentrated boric acid storage tank. At least one channel of heat tracing capable of meeting the above temperature requirement shall be in operation. One associated boric acid pump shall be operable.

If the concentrated boric acid storage tank with its associated flowpath is unavailable, but the borated water storage tank is available and operable, the concentrated boric acid storage tank shall be restored to operability within 72 hours or the reactor shall be placed in a hot shutdown condition and be borated to a shutdown margin equivalent to 1% $\Delta k/k$ at 200°F within the next twelve hours; if the concentrated boric acid storage tank has not been restored to operability within the next 7 days the reactor shall be placed in a cold shutdown condition within an additional 30 hours.

If the concentrated boric acid storage tank is available but the borated water storage tank is neither available nor operable, the borated water storage tank shall be restored to operability within one hour or the reactor shall be placed in a hot shutdown condition within 6 hours and in a cold shutdown condition within an additional 30 hours.

Bases

The high pressure injection system and chemical addition system provide control of the reactor coolant system boron concentration.(1) This is normally accomplished by using any of the three high pressure injection pumps in series with a boric acid pump associated with either the boric acid mix tank or the concentrated boric acid storage tank. An alternate method of boration will be the use of the high pressure injection pumps taking suction directly from the borated water storage tank.(2)

The quantity of boric acid in storage in the concentrated boric acid storage tank or the borated water storage tank is sufficient to borate the reactor coolant system to a 1% $\Delta k/k$ subcritical margin at cold conditions (70°F) with the maximum worth stuck rod and no credit for xenon at the worst time in core life. The current cycles for each unit are analyzed with the limits presented in the Core Operating Limits Report. The cycle specific analyses determine the volume and boron concentration requirements for the BWST and CBAST necessary to borate to cold shutdown. The volume requirements include a 10% margin and, in addition, allow for a deviation of 10 EFPD in the cycle length. The specification assures that two supplies are available whenever the reactor is critical so that a single failure will not prevent boration to a cold condition.

The concentration of boron in the concentrated boric acid storage tank may be higher than the concentration which would crystallize at ambient conditions. For this reason, and to assure a flow of boric acid is available when needed, these tanks and their associated piping will be kept at least 10°F above the crystallization temperature for the concentration present. Once in the high pressure injection system, the concentrate is sufficiently well mixed and diluted so that normal system temperatures assure boric acid solubility.

REFERENCES

(1) FSAR, Sections 9.3.1, and 9.3.2
(2) FSAR, Figure 6.0.2
(3) Technical Specification 3.3

3.3.3 Core Flood Tank (CFT) System

When the RCS is in a condition with pressure above 800 psig both CFT's shall be operable with the electrically operated discharge valves open and breakers locked open and tagged; a minimum level of 13 ± .44 feet (1040 ± 30 ft.^{3}) and one level instrument channel per CFT; a minimum boron concentration within the limit specified in the Core Operating Limits Report in each CFT; and pressure at 600 ± 25 psig with one pressure instrument channel per CFT.

3.3.4 Borated Water Storage Tank (BWST)

When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:

a. The BWST shall have operable two level instrument channels.

(1) Tests or maintenance shall be allowed on one channel of BWST level instrumentation provided the other channel is operable.

(2) If the BWST level instrumentation is not restored to meet the requirements of Specification 3.3.4.a above within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the requirements of Specification 3.3.4.a are not met within 24 hours following hot shutdown, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of boron within the limit specified in the Core Operating Limits Report at a minimum temperature of 50°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

3.3.5 Reactor Building Cooling (RBC) System

a. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable.

Bases

Specification 3.3 assures that, for whatever condition the reactor coolant system is in, adequate engineered safety feature equipment is operable.

For operation up to 60% FP, two high pressure injection pumps are specified. Also, two low pressure injection pumps and both core flood tanks are required.

In the event that the need for emergency core cooling should occur, functioning of one high pressure injection pump, one low pressure injection pump, and both core flood tanks will protect the core, and in the event of a main coolant loop severance, limit the peak clad temperature to less than 2,200°F and the metal-water reaction to that representing less than 1 percent of the clad. (1) Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.

The requirement to have three HPI pumps and two HPI flowpaths operable during power operation above 60% FP is based on considerations of potential small breaks at the reactor coolant pump discharge piping for which two HPI trains (two pumps and two flow paths) are required to assure adequate core cooling. (2) The analysis of these breaks indicates that for operation at or below 60% FP only a single train of the HPI system is needed to provide the necessary core cooling.

The requirement for a flowpath from LPI discharge to HPI pump suction is provided to assure availability of long term core cooling following a small break LOCA in which the BWST is depleted and RCS pressure remains above the shutoff head of the LPI pumps.

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature above 50°F to lessen the potential for thermal shock of the reactor vessel during high pressure injection system operation. The boron concentration is set at the amount of boron required to maintain the core 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The minimum boron concentration is specified in the Core Operating Limits Report.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the Reactor Building design pressure will not be exceeded with one spray and two coolers operable. (4) Therefore, a maintenance period of seven days is acceptable for one Reactor Building cooling fan

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) Axial Power Imbalance Protective Limits and Variable Low RCS Pressure Protective Limits for Specification 2.1.

(2) Reactor Protective System Trip Setting Limits for the Flux/Flow/Imbalance and Variable Low Reactor Coolant System Pressure trip functions in Specification 2.3.

(3) Power Dependent Rod Insertion Limits for Specifications 3.1.3.5, 3.1.11, 3.5.2.1.b, 3.5.2.2.d.2.c, 3.5.2.3, and 3.5.2.5.c.

(4) Concentrated Boric Acid Storage Tank volume and boron concentration for Specification 3.2.2.

(5) Core Flood Tank boron concentration for Specification 3.3.3.

(6) Borated Water Storage Tank boron concentration for Specification 3.3.4.

(7) Quadrant Power Tilt Limits for Specification 3.5.2.4.a, 3.5.2.4.b, 3.5.2.4.d, 3.5.2.4.e, and 3.5.2.4.f.

(8) 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 applic able 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. 197 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 197 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 194 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 July 13, 1992, as supplemented December 1, 1992, Duke Power Company (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 Technical Specifications (TS). The requested changes would relocate some cycle-specific operating parameter limits currently specified in the TS to the Core Operating Limits Report (COLR) in accordance with the guidance provided in Generic Letter (GL) 88-16. Previously, TS Amendments 172/172/169 and 191/191/198 similarly revised Oconee TS to replace the values of other cycle-specific parameter limits with references to the COLR providing these values. The December 1, 1992, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The following TS were revised to replace the values of cycle-specific parameter limits with references to the COLR that provide these limits.

(1) TS 2.1

Axial power imbalance protective limits and variable low RCS pressure protective limits for this specification are specified in the COLR.

(2) TS 3.2.2

The concentrated boric acid storage tank volume and boron concentration limits for this specification are specified in the COLR.

(3) TS 3.3.3

The core flood tank boron concentration limit is specified in the COLR.

(4) TS 3.3.4

The borated water storage tank boron concentration limit is specified in the COLR.

We have reviewed the relocation of the cycle-specific values of the above core operating limits from the TS to the COLR and concluded that the relocation is acceptable in light of the guidance provided in GL 88-16.

The bases of the above specifications have been modified to include appropriate references to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

TS 6.9, Core Operating Limits Report, is revised to include the currently proposed TS changes in Specification 6.9.1. This clarifying revision is acceptable.

On the basis of its review of the above items, the NRC staff concludes that the licensee's proposed amendments are in accordance with the NRC guidance in GL 88-16 on modifying cycle-specific parameter limits in the TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC-approved methodologies, the NRC staff concludes that this change has no impact on plant safety. Accordingly, the NRC staff finds that the proposed changes are acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes recordkeeping or reporting requirements. The NRC 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 the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 40210, dated September 2, 1992). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Huang
S. Kirslis

Date: January 5, 1993