September 16, 19

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

Distribution
See next page

Mr. M. S. Tuckman
Vice President -
Nuclear Operations
Duke Power Company
P. O. Box 1007
Charlotte, North Carolina 28201-1007

See Correction letter of 11/18/91

Dear Mr. Tuckman:

SUBJECT: ISSUANCE OF AMENDMENT NOS. 191 , 191 AND 188 TO FACILITY OPERATING LICENSES DPR-38, DPR-47 AND DPR-55 - OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 (TACS 80378/80379/80380)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 191 , 191 and 188 to Facility Operating Licenses Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2 and 3. The amendments consist of changes to the Station's TSs in response to your request dated May 7, 1991, as supplemented May 13, August 1, and August 15, 1991.

The amendments modify specifications having cycle-specific parameter limits by transferring these limits to a Core Operating Limits Report (COLR). In addition, the specified height of the active fuel assembly is revised to incorporate a new fuel design.

A copy of the related 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. 191 to DPR-38
2. Amendment No. 191 to DPR-47
3. Amendment No. 188 to DPR-55
4. Safety Evaluation

cc w/enclosures:
See next page

LA:PDII3	PM:PDII3	OGC	D:PDII3
LBerry	LWiens	MYoung	DMatthews
8/29/91	9/3/91	9/10/91	9/16/91

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Document Name: CYCLE AMEND

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Oconee Nuclear Station
Units Nos. 1, 2 and 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. 191
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 May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, 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. 191, 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: September 16, 1991

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. 191
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 May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, 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. 191, 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: September 16, 1991

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. 188
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 May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, 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. 188, 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
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: September 16, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 188

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
vi, vii	vi, vii
2.1-1	2.1-1
2.1-2	2.1-2
2.1-3	2.1-3
2.1-4	2.1-4
2.3-5	2.3-5
2.3-6	--
2.3-7	--
3.5-7 - 3.5-14	3.5-7 - 3.5-14
5.3-1	5.3-1
6.9-1	6.9-1

LIST OF TABLES

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 - (6.5×10^{-3})x(Burnup, MWD/MTU) ^{o}F. Operation within this limit is assured by compliance with the Axial Power Imbalance Protective Limits as specified in Figure 2.1-2.

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 limits as specified in Figures 2.1-2 and 2.1-1 respectively.

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 curve presented in Figure 2.1-1 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 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 reactor power imbalance limits, Figure 2.1-2, 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 in Figure 2.1-2.

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.

Figure 2.1-1

Variable Low Pressure

Protective Limits

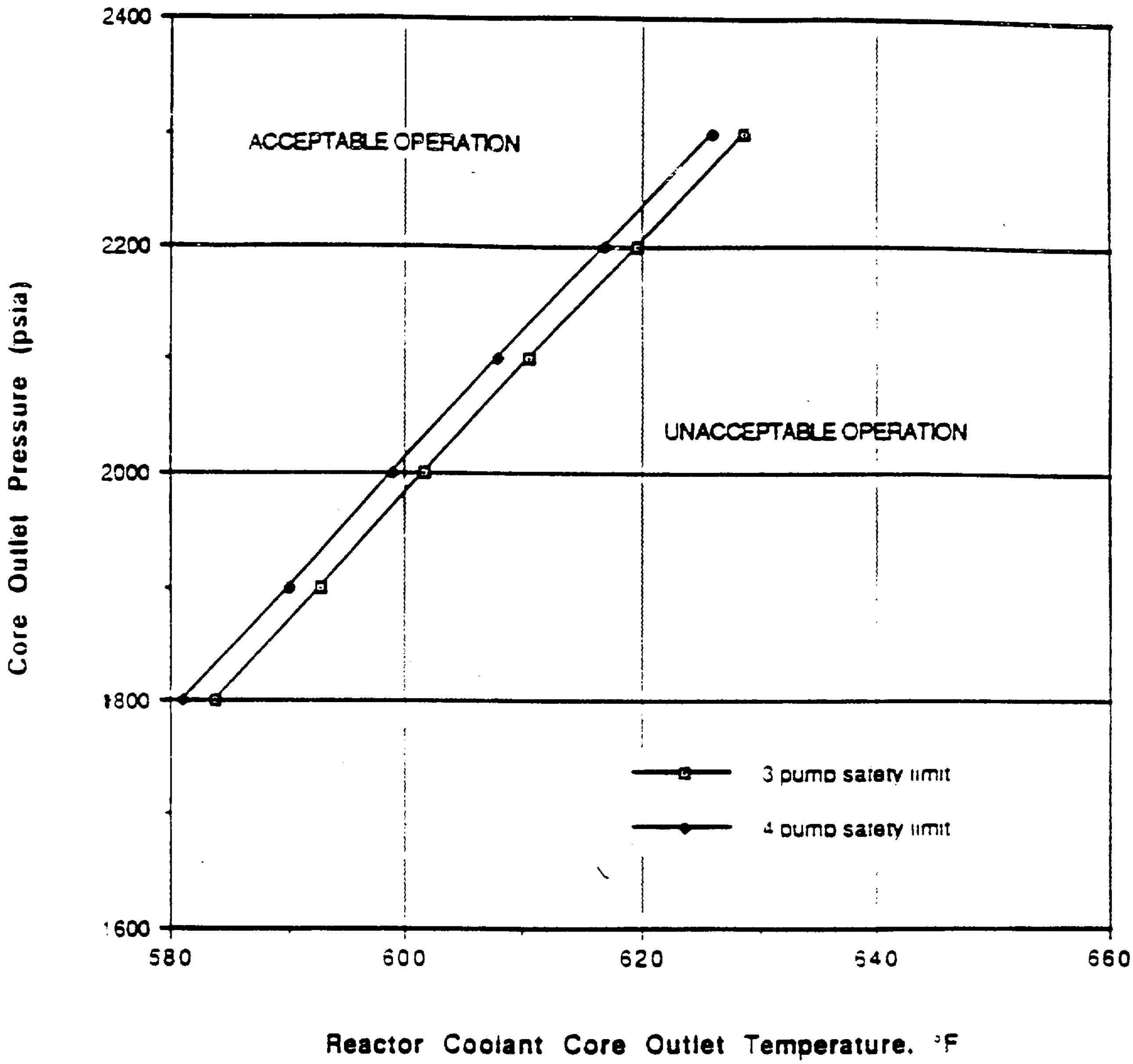
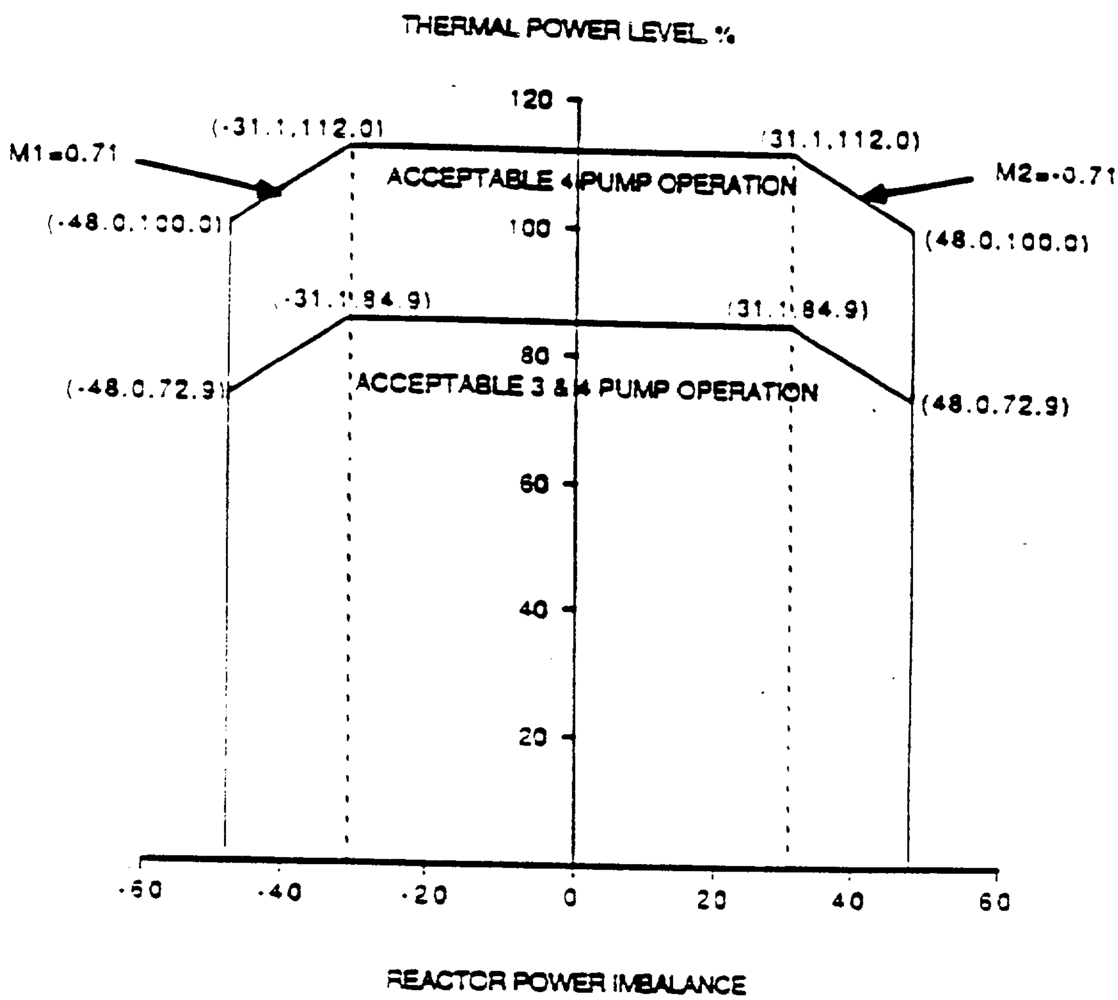


Figure 2.1-2

Axial Power

Imbalance Protective Limits



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.

The pump monitors shall produce a reactor trip when a loss of two pumps occurs and the reactor is at power operation greater than 2.0% 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 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. 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 ratio 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 Figure 1.3 of 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 Figure 1.3 of 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 Figure 1.4 of the Core Operating Limits Report ensure that the minimum DNBR is greater than or equal to the minimum allowable DNBR for those accidents that result in a reduction in pressure. (3,4) The limits shown in Figure 1.4 of 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 Figure 1.4 of the Core Operating Limits Report has been established to prevent excessive core coolant temperatures. Accounting for calibration and instrumentation errors, the safety analyses use 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 ≤ 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 ≤ 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

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	Figure 1.3 of 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	Figure 1.4 of 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.

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 provided in the Core Operating Limits Report 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 provided in the Core Operating Limits Report, then:

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

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

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

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

d. If the quadrant power tilt exceeds the Transient Limit but is less than the Maximum Limit provided in the Core Operating Limits Report 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 provided in the Core Operating Limits Report, 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 provided in the Core Operating Limits Report, 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.

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. Hot rod manufacturing tolerance 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% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2) of hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning-of-life, hot zero power would result in a lower transient peak thermal power and, therefore, less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

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 the values assumed in the reload design analyses. 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 given in Specification 6.9.2.

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 15.12

Page 3.5-14 Not Used

5.3 REACTOR

Specification

5.3.1 Reactor Core

5.3.1.1 The reactor core contains approximately 93 metric tons of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 177 fuel assemblies, all of which are prepressurized with Helium. (1)

5.3.1.2 The fuel assemblies shall form an essentially cylindrical lattice with an active height range of 140.5 in. to 142 in. and an equivalent diameter of 128.9 in. (1)

5.3.1.3 There are 61 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSR) distributed in the reactor core as shown in FSAR Figure 4.3-3. The full-length CRA and the APSR shall conform to the design described in the FSAR or reload report. (1)

5.3.1.4 Initial core and reload fuel assemblies and rods shall conform to design and evaluation described in the FSAR.

5.3.2 Reactor Coolant System

5.3.2.1 The design of the pressure components in the reactor coolant system shall be in accordance with the code requirements. (2)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, shall be designed for a pressure of 2,500 psig and a temperature of 650°F. The pressurizer and pressurizer surge line shall be designed for a temperature of 670°F. (3)

5.3.2.3 The maximum reactor coolant system volume shall be 12,200 ft^3.

REFERENCES

(1) FSAR Section 4.2.2

(2) FSAR Section 5.2.3.1

(3) FSAR Section 5.2.1

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

(2) 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.

(3) 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.

(4) 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. 191TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO.191 TO FACILITY OPERATING LICENSE DPR-47

AMENDMENT NO. 188TO 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 May 7, 1991, as supplemented May 13, August 1, and August 15, 1991, the Duke Power Company (DPC or the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 (ONS), Technical Specifications (TS). The requested changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to a Core Operating Limits Report (COLR) for the values of those limits in accordance with the guidance provided in Generic Letter 88-16. The use of the COLR for ONS was previously approved by the NRC. Also, the proposed Technical Specification 5.3.1.2 revises the height of the active fuel assembly to incorporate a new fuel design for Oconee Unit 1, Cycle 14 and future reload designs. The May 13, August 1 and August 15, 1991, letters provided clarifying information that did not change the action noticed in the Federal Register on June 26, 1991, and did not affect the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 Core Operating Limits Report

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) In addition to the approved cycle-specific core operating limits which were transferred to a COLR in accordance with a TS amendment dated January 26, 1989, the following specifications for Unit 1 Cycle 14 operation were revised to replace the value of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 2.3

The values of the core flux/flow/imbalance and variable low reactor coolant system pressure trip functions for the reactor protective system (RPS) setpoint are specified in the COLR.

(b) Specifications 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

The steady state, transient, and maximum quadrant power tilt limits are specified in the COLR.

The staff has made several conference calls with the licensee in August 1991 to resolve concerns relating to its May 7, 1991 submittal proposing to relocate two TS 2.1 figures to the COLR (figures for the combination of the reactor system pressure and temperature and the reactor power imbalance). The concern involved the proposed inclusion of safety limits in the operating limits report. The licensee responded to the staff concern in its August 15, 1991 submittal by returning those two relocated figures to TS 2.1 as Figures 2.1-1 and 2.1-2. We have found this supplement in the August 15, 1991 submittal acceptable.

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

(2) Specification 6.9 Core Operating Limits Report of the Administrative Controls section of the TS is revised to include currently proposed TS changes in Specification 6.9.1.

On the bases of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in 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 is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

2.2 Active Fuel Height Range

The licensee proposed changes to TS 5.3.1.2 to establish new dimensional requirements for the fuel assemblies specifying an active fuel height range of 140.5 inches to 142 inches. The Oconee Unit 1, Cycle 14 reload design includes a new batch of fuel assemblies with an active height of 140.5 inches. Therefore, a revision to TS 5.3.1.2 is necessary to accommodate both the old and the new fuel design. The licensee intends to use the new fuel design in all future Oconee reloads.

The new fuel assembly designs, the MK-B10, are an evolution from the existing MK-B8 fuel assembly series presently in service. They are 15x15 fuel rod lattices with Zircaloy spacer grids and are identical in every regard to the earlier MK-B8 designs with the exception of the following improvements:

(a) Cruciform hold-down spring (B10 only)
(b) Reduced pellet/clad gap fuel rod
(c) Reduced bypass flow guide tubes
(d) Skirtless, removable lower end fitting

2.2.1 Cruciform Hold-down Spring (B10 only)

Batch No. 16 fuel assemblies destined to be implemented in the upcoming Cycle 14, consists of 48 MK-B9 assemblies and 4 MK-B10 lead test assemblies (LTAs). The only difference between the MK-B10 and MK-B9 is the hold-down spring in the upper end fitting. The hold-down spring in the MK-B10 is of the cruciform design as opposed to the traditional helical springs found in the MK-B9 assemblies. The implementation of the hold-down spring significantly improves reliability and hold-down capacity such that current fuel assembly lift analyses based on the helical spring design are bounding.

2.2.2 Reduced Pellet/Clad Gap Fuel Rod

Data supplied by the licensee describes design differences between the old and the new fuel type. The MK-B10/MK-B9 fuel rod designs were based on the MK-B8 design. The basic differences lie in pellet diameter increase (from 0.3686 to 0.370 inches), pellet-cladding diameter gap decrease (from 0.0084 to 0.007 inches), a slight change in pellet dish geometry configuration, and fuel stack length decrease from 141.8 to 140.595 inches). The consequences of these small changes will result in a fuel rod with the same loading as the previous design, but with a smaller pellet-cladding gap and a larger plenum volume, leading to lower fuel temperatures and higher pin pressure burnup limits.

2.2.3 Reduced Bypass Flow Guide Tubes

The MK-B9/MK-B10 fuel assembly designs have optimized control component insertion guide tubes to reduce bypass flow and improve fuel thermal hydraulic performance (more heat removal is available in the core). Although too much reduction in flow could adversely effect control rod drop time and core cooling, the licensee has performed control rod drop tests confirming that Technical Specification limits are not exceeded. The licensee has also performed component analyses to ensure bulk boiling or centerline melt do not occur, and to ensure that component internal pressures are acceptable. The conclusion of these analyses indicate that reduced guide tube flow is acceptable.

The licensee analyzed the linear rate (LHR) for the upcoming Cycle 14. All fuel assemblies in the Cycle 14 core are thermally similar except for fuel batch #16 containing fuel assemblies MK-B9 and MK-B10. Data submitted by the licensee indicates that the fresh fuel assemblies, MK-B9/MK-B10, differ significantly in performance from the remaining fuel in the core. The thermal performance calculations were carried out via the TACO2 computer code and compared to the licensee's in-house generic analyses. Data supplied by the licensee indicates that the current LHR limits are bounding.

2.2.4 Skirtless, Removable Lower End Fitting

The MK-B9 and MK-B10 fuel assemblies differ from the MK-B8 fuel assembly in that the MK-B9 and MK-B10 feature a removable lower end fitting to provide added flexibility to accomplish field maintenance. This design improvement will complement the presently existing removable fitting. Both upper and lower removable end fittings facilitate field repairs and fuel loading.

On the basis of the review of the above items, the NRC staff concludes that the licensee's proposed changes to TS 5.3.1.2 are acceptable since methodologies uesd for the reload analyses and the new batch of fuel assemblies used for Oconee Unit 1, Cycle 14 operation are approved by the staff.

3.0 FINDINGS

The NRC staff has reviewed the licensee's request to modify the Technical Specifications of the Oconee Nuclear Station that would remove the specific values of additional cycle-dependent parameters from the specifications and place the values in a Core Operating Limit Report that would be referenced by the specifications, and to establish a fuel assembly active height range in TS 5.3.1.2. Based on this review, the NRC staff concludes that these Technical Specification changes are acceptable.

4.0 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.

5.0 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. 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 (56 FR 29272). 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 the amendments.

6.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Huang, SRXB
L. Wiens, PDII-3

Date: September 16, 1991

REFERENCES

1. Letter from M. S. Tuckman (DPC) to NRC, dated May 7, 1991.
2. Letter from M. S. Tuckman (DPC) to NRC, dated May 13, 1991.
3. Letter from M. S. Tuckman (DPC) to NRC, dated August 1, 1991.
4. Letter from M. S. Tuckman (DPC) to NRC, dated August 15, 1991.

DATED: September 16, 1991

AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE DPR-38 - Oconee Nuclear Station, Unit 1
AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE DPR-47 - Oconee Nuclear Station, Unit 2
AMENDMENT NO. 188 TO FACILITY OPERATING LICENSE DPR-55 - Oconee Nuclear Station, Unit 3

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