UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 21, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3 RE: ISSUANCE OF AMENDMENTS (TAC NOS. MA8674, MA8675 AND MA8676)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 313, 313, and 313 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 in response to your application dated April 13, 2000, as supplemented by letter dated May 30, 2000.

The amendments revise the Technical Specifications and associated Bases pages to accommodate the use of Mark-B11 fuel with M5 cladding.

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,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 313 to DPR-38
2. Amendment No. 313 to DPR-47
3. Amendment No. 313 to DPR-55
4. Safety Evaluation

cc w/encls: See next page

June 21, 2000

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

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/RA/

David E. LaBarge, Senior Project Manager, Section 1
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4. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\PDII-1\OCONEE\ma8674amd.wpd

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 313
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 Energy Corporation (the licensee) dated April 13, 2000, as supplemented by letter dated May 30, 2000, 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:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 313 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raghavan, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 21, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 313
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 Energy Corporation (the licensee) dated April 13, 2000, as supplemented by letter dated May 30, 2000, 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:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 313 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raghavan, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 21, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 313
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 Energy Corporation (the licensee) dated April 13, 2000, as supplemented by letter dated May 30, 2000, 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:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 313 , 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 and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

L. Raghavan, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: June 21, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 313

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 313

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 313

FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Appendix A Technical Specifications and associated Bases pages with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
TS LOEP1	TS LOEP1
TS LOEP8	TS LOEP8
TS LOEP9	TS LOEP9
BASES LOEP1	BASES LOEP1
BASES LOEP2	BASES LOEP2
BASES LOEP7	BASES LOEP7
2.0-1	2.0-1
4.0-1	4.0-1
5.0-31	5.0-31
B 2.1.1-1	B 2.1.1-1
B 2.1.1-4	B 2.1.1-4
B 3.1.4-6	B 3.1.4-6
B 3.4.1-1	B 3.4.1-1

OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS
LIST OF EFFECTIVE PAGES

PAGE	AMENDMENT	REVISION DATE
LOEP1	312/312/312	06/06/00
LOEP2	300/300/300	12/16/98
LOEP3	304/304/304	04/28/99
LOEP4	309/309/309	1/18/00
LOEP5	300/300/300	12/16/98
LOEP6	309/309/309	1/18/00
LOEP7	312/312/312	06/06/00
LOEP8	310/310/310	1/18/00
LOEP9	310/310/310	1/18/00
i	300/300/300	12/16/98
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iii	309/309/309	1/18/00
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1.1-1	300/300/300	12/16/98
1.1-2	300/300/300	12/16/98
1.1-3	300/300/300	12/16/98
1.1-4	300/300/300	12/16/98
1.1-5	300/300/300	12/16/98
1.1-6	300/300/300	12/16/98
1.2-1	300/300/300	12/16/98
1.2-2	300/300/300	12/16/98
1.2-3	300/300/300	12/16/98
1.3-1	300/300/300	12/16/98
1.3-2	300/300/300	12/16/98
1.3-3	300/300/300	12/16/98
1.3-4	300/300/300	12/16/98
1.3-5	300/300/300	12/16/98
1.3-6	300/300/300	12/16/98
1.3-7	300/300/300	12/16/98
1.3-8	300/300/300	12/16/98
1.3-9	300/300/300	12/16/98
1.3-10	300/300/300	12/16/98
1.3-11	300/300/300	12/16/98
1.3-12	300/300/300	12/16/98
1.3-13	300/300/300	12/16/98
1.4-1	300/300/300	12/16/98
1.4-2	300/300/300	12/16/98
1.4-3	300/300/300	12/16/98
1.4-4	300/300/300	12/16/98
2.0-1	313/313/313	6/21/00

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OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS
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3.9.3-2	300/300/300	12/16/98
3.9.4-1	300/300/300	12/16/98
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3.9.5-1	300/300/300	12/16/98
3.9.5-2	300/300/300	12/16/98
3.9.6-1	300/300/300	12/16/98
3.9.6-2	300/300/300	12/16/98
3.9.7-1	309/309/309	1/18/00
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3.10.1-3	300/300/300	12/16/98
3.10.1-4	300/300/300	12/16/98
3.10.1-5	300/300/300	12/16/98
3.10.2-1	300/300/300	12/16/98
3.10.2-2	300/300/300	12/16/98
3.10.2-3	300/300/300	12/16/98
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OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS
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	310/310/310	
5.0-24	310/310/310	1/18/00
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5.0-26	310/310/310	1/18/00
5.0-27	310/310/310	1/18/00
5.0-28	310/310/310	1/18/00
5.0-29	310/310/310	1/18/00
5.0-30	310/310/310	1/18/00
5.0-31	313/313/313	6/21/00
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OCONEE NUCLEAR STATION
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LOEP3	BASES REVISION	12/16/98
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LOEP5	BASES REVISION	06/02/99
LOEP6	309/309/309	1/18/00
LOEP7	309/309/309	1/18/00
LOEP8	309/309/309	1/18/00
LOEP9	BASES REVISION	12/10/99
LOEP10	BASES REVISION	01/31/00
LOEP11	309/309/309	1/18/00
LOEP12	BASES REVISION	01/31/00
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LOEP15	BASES REVISION	01/31/00
LOEP16	309/309/309	1/18/00
LOEP17	BASES REVISION	12/16/98
i	BASES REVISION	03/27/99
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iii	309/309/309	1/18/00
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B 2.1.1-1	313/313/313	6/21/00
B 2.1.1-2	300/300/300	12/16/98
B 2.1.1-3	300/300/300	12/16/98
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B 3.0-14	300/300/300	12/16/98
B 3.1.1-1	300/300/300	12/16/98
B 3.1.1-2	BASES REVISION	05/11/99
B 3.1.1-3	300/300/300	12/16/98
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B 3.1.2-1	300/300/300	12/16/98
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B 3.1.3-2	BASES REVISION	03/27/99
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B 3.1.3-4	300/300/300	12/16/98
B 3.1.4-1	300/300/300	12/16/98
B 3.1.4-2	300/300/300	12/16/98
B 3.1.4-3	300/300/300	12/16/98
B 3.1.4-4	300/300/300	12/16/98
B 3.1.4-5	300/300/300	12/16/98
B 3.1.4-6	313/313/313	6/21/00
B 3.1.4-7	300/300/300	12/16/98
B 3.1.4-8	300/300/300	12/16/98
B 3.1.4-9	300/300/300	12/16/98
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B 3.1.6-2	300/300/300	12/16/98
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OCONEE NUCLEAR STATION
TECHNICAL SPECIFICATIONS - BASES
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B 3.3.19-2	300/300/300	12/16/98
B 3.3.19-3	300/300/300	12/16/98
B 3.3.19-4	300/300/300	12/16/98
B 3.3.20-1	300/300/300	12/16/98
B 3.3.20-2	300/300/300	12/16/98
B 3.3.20-3	300/300/300	12/16/98
B 3.3.20-4	300/300/300	12/16/98
B 3.3.21-1	300/300/300	12/16/98
B 3.3.21-2	300/300/300	12/16/98
B 3.3.21-3	300/300/300	12/16/98
B 3.3.22-1	BASES REVISION	03/27/99
B 3.3.22-2	300/300/300	12/16/98
B 3.3.23-1	300/300/300	12/16/98
B 3.3.23-2	300/300/300	12/16/98
B 3.3.23-3	300/300/300	12/16/98
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B 3.4.1-4	309/309/309	1/18/00
B 3.4.1-5	300/300/300	12/16/98
B 3.4.2-1	300/300/300	12/16/98
B 3.4.2-2	300/300/300	12/16/98
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B 3.4.3-2	307/307/307	10/01/99
B 3.4.3-3	307/307/307	10/01/99
B 3.4.3-4	307/307/307	10/01/99
B 3.4.3-5	307/307/307	10/01/99
B 3.4.3-6	307/307/307	10/01/99
B 3.4.3-7	307/307/307	10/01/99
B 3.4.3-8	Delete	10/01/99
B 3.4.4-1	300/300/300	12/16/98
B 3.4.4-2	300/300/300	12/16/98
B 3.4.4-3	300/300/300	12/16/98

6/21/00

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be $\leq$ 4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU))° F. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits as specified in the Core Operating Limits Report.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limit of 1.18 for the BWC correlation and 1.19 for the BWU correlation. Operation within this limit is ensured by compliance with the Axial Power Imbalance Protective Limits and RCS Variable Low Pressure Protective Limits as specified in the Core Operating Limits Report.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained $\leq$ 2750 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits and be in MODE 3 within 1 hour.

2.2.3 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to $\leq$ 2750 psig within 5 minutes.

4.0 DESIGN FEATURES

4.1 Site Location

The Oconee Nuclear Station is approximately eight miles northeast of Seneca, South Carolina. The minimum distance from the reactor center line to the boundary of the exclusion area and to the outer boundary of the low population zone, as defined in 10 CFR 100.3, shall be one mile and six miles respectively.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of zirconium alloy or M5 clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy, M5, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Assemblies

The reactor core shall contain 61 full-length CONTROL ROD Assemblies (CRAs) and 8 APSR assemblies. The full-length CRAs and APSR assemblies shall conform to the design described in the UFSAR or reload report.

4.3 Fuel Storage

4.3.1 Criticality

The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum nominal U-235 enrichment of 5.0 weight percent;

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

(7) DPC-NE-3000P-A, Thermal Hydraulic Transient Analysis Methodology, Rev. 2, (SER dated October 14, 1998);

(8) DPC-NE-2005P-A, Thermal Hydraulic Statistical Core Design Methodology, Rev. 2, (SER dated June 8, 1999); and

(9) DPC-NE-3005-PA, UFSAR Chapter 15 Transient Analysis Methodology, Rev. 1, (SER dated May 25, 1999).

(10) BAW-10227-PA, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, (SER dated December 14, 1999).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) and Main Feeder Bus Monitor Panel (MFPMP) Report

When a report is required by Condition B or G of LCO 3.3.8, "Post Accident Monitoring (PAM) Instrumentation" or Condition D of LCO 3.3.23, "Main Feeder Bus Monitor Panel," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring (PAM only), the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.7 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

ONS Design Criteria (Ref. 1) require that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and anticipated transients. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

DNB is not a directly measurable parameter during operation, but neutron power and Reactor Coolant System (RCS) temperature, flow and pressure can be related to DNB using a critical heat flux (CHF) correlation. The BWC (Ref. 2) and the BWU (Ref. 4) CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The BWC correlation applies to Mark-BZ fuel. The BWU correlation applies to the Mark-B11 fuel. 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 minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.18 (BWC) and 1.19 (BWU).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film,

BASES (continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1

If SL 2.1.1.1 or SL 2.1.1.2 is violated, the requirement to go to MODE 3 places the unit in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the unit to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

REFERENCES

1. UFSAR, Section 3.1.

2. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," April 1995.

3. UFSAR, Chapter 15.

4. BAW-10199P, "The BWU Critical Heat Flux Correlations," Addendum 1, April 2000

BASES

ACTIONS (continued)

A.2.4

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.18% $\Delta k/k$ at RTP, 0.36% $\Delta k/k$ at 80% RTP, or 0.7% $\Delta k/k$ at zero power. This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the duration of time that operation is expected to continue with a misaligned rod. Should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation.

B.1

If the Required Actions and associated Completion Times for Condition A are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

C.1.1

More than one trippable CONTROL ROD becoming inoperable or misaligned, or both inoperable but trippable and misaligned from their group average position, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit, then the restoration of the required SDM requires increasing the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated transients assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with operation within the nominal operating envelope and is above that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher minimum DNBR. A pressure lower than the minimum specified will cause the unit to approach the DNB limit.

The LCO for maximum RCS coolant loop average temperature is consistent with full power operation within the nominal operating envelope and is lower than the initial loop average temperature in the analyses. A loop average temperature lower than that specified will produce a higher minimum DNBR. A loop average temperature higher than that specified will cause the unit to approach the DNB limit.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. A higher RCS flow rate will produce a higher DNBR. A lower RCS flow will cause the unit to approach the DNB limit.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criterion of $\geq$ 1.18 for BWC correlation, $\geq$ 1.19 for BWU correlation, or an equally valid limit when the statistical DNBR limit is employed (SCD methodology). This is the acceptance limit for the RCS DNBR parameters.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 313 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 313 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 313 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated April 13, 2000, as supplemented by letter dated May 30, 2000, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS) and associated Bases. The requested changes would accommodate the use of Mark-B11 fuel with M5 cladding. The supplement dated May 30, 2000, provided clarifying information that did not change the scope of the April 13, 2000, application and the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

By letter dated December 7, 1997, the licensee stated that it intended to transition all three Oconee units to Framatome Cogema Fuels Mark-B11 fuel. By letter dated January 31, 2000, the licensee updated the information provided in its December 7, 1997, letter, and included its intention to use M5 cladding. The proposed amendments make several changes to the TS and associated Bases to accommodate the use of the Mark-B11 fuel with M5 cladding.

3.0 EVALUATION

The first proposed change revises TS 2.1.1.2 by adding the BWU correlation and its departure from nucleate boiling ratio limit of 1.19. The BWU correlation is used to analyze the Mark-B11 fuel and it has been approved by the staff in a letter dated April 6, 2000, "Acceptance for Referencing of Licensing Topical Report BAW-10199P, Addendum 1, The BWU Critical Heat Flux Correlations (TAC No. M96728)."

During upcoming refueling outages, the present Mark-BZ fuel will be replaced with Mark-B11 fuel that uses M5 cladding. As a result, the proposed TS change contains two critical heat flux correlations, one for the present Mark-BZ fuel design (BWC, 1.18) and another for the new Mark-B11 fuel design (BWU, 1.19). Therefore, correlations corresponding to both BWC and BWU would be specified and applied to the fuel type as appropriate.

The second proposed change revises TS 4.2.1 to allow the use of M5 cladding. The use of M5 cladding for Oconee has been approved by the staff in a letter dated March 23, 2000, "Exemption from Fuel Cladding Requirements (TAC Nos. MA6466, MA6467, and MA6468)."

The third proposed change revises TS 5.6.5 b(8) to update the thermal hydraulic statistical core design topical report, DPC-NE-2005P-A, to the latest revision, Revision 2. DPC-NE-2005P-A, Revision 2, was approved by the staff in its safety evaluation dated June 3, 1999.

The last proposed change revises TS 5.6.5 b to add a reference to BAW-10227-PA, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel." BAW-10227-PA describes the loss-of-coolant accident evaluation model changes that are necessary to analyze M5 cladding. BAW-10227-PA was approved by the staff in its safety evaluation dated December 14, 1999.

Thus, the proposed changes are administrative in nature since they incorporate provisions that have previously been approved by the staff. In addition, the proposed Bases changes support the TS changes. Therefore, the staff finds them 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 (65 FR 31356). 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: June 21, 2000

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