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. 234
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 July 15, 1997, as supplemented March 3, April 13, June 16, October 26, and November 5, 1998, 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 Appendices A and B, as revised through Amendment No. 234 , 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 coincident with implementation of the improved Technical Specification amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1998

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. 234
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 July 15, 1997, as supplemented March 3, April 13, June 16, October 26, and November 5, 1998, 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 Appendices A and B, as revised through Amendment No. 234 , 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 coincident with implementation of the improved Technical Specification amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1998

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. 233
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 July 15, 1997, as supplemented March 3, April 13, June 16, October 26, and November 5, 1998, comply 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 Appendices A and B, as revised through Amendment No. 233 , 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 coincident with implementation of the Improved Technical Specification amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: December 7, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 234

FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 233

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	Insert
iii	iii
3.5-1	3.5-1
3.5-5c	3.5-5c
3.5-5d	3.5-5d
-----	3.5-5e
-----	3.5-48
-----	3.5-49
-----	3.5-50
4.1-8b	4.1-8b
4.1-9	4.1-9
4.1-9a	4.1-9a

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

3.5.1.5 During startup when the wide range instruments come on scale, the overlap between the wide range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action if Conditions of Column C Cannot Be Met
20. Main Steam Header Pressure and MSLB detection (analog) channels per steam generator	3	2	3 (k)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
21. Feedwater isolation circuitry (digital) channels	2	1	2 (l)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
22. Feedwater isolation circuitry (digital) channels manual pushbutton	2	1	2 (l)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

(d) (Deleted)

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

(f)
1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or
2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

(g) (Deleted)

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

(i)
1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or
2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

(k) Requirement of 3 channels can be met with one of three channels placed in trip. The affected channel shall be placed in trip within 4 hours of discovery.

(l) 1 of 2 digital channels or manual pushbutton can be disabled for up to 72 hours and still meet the requirements of this column.

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability

Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F.

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside contaiment.

Specifications

3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22.

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

Bases

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner.

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main feedwater to both steam generators, and inhibit autostart/initiate autostop of the turbine driven emergency feedwater pump. In addition, to further decrease operator burden, this circuitry will initiate the same automatic actions if a MSLB occurs outside containment.

The MSLB detection and feedwater isolation circuitry is divided into two parts which consist of the MSLB detection circuitry and the feedwater isolation circuitry. The MSLB detection circuitry consists of three MSLB detection analog channels per main steam header (total of six). A detection analog channel consists of a pressure transmitter, a signal isolator(s) (if necessary), and a current switch(es). The feedwater isolation circuitry is divided into two redundant digital channels. Each digital channel consists of two parallel 2 out of 3 logic combinations. The three detection analog channels on each main steam header provide input to the two parallel 2 out of 3 logic combinations in each digital channel. Actuation of either 2/3 logic circuit in a digital channel will actuate that digital channel. Feedwater isolation will occur if either digital channel is actuated. Thus, low steam generator pressure in either steam generator fully actuates the system. In addition, each digital channel consists of a manual bypass pushbutton, an enable/disable switch, a seal-in, a time delay, and a master relay. The master relay is energized to cause the feedwater isolation.

MSLB detection and feedwater isolation circuitry is considered operable provided all of the following conditions are met:

a. Feedwater isolation digital channels are operable per Specification Table 3.5.1-1 Item # 21, enabled, and the MSLB manual initiation is functional per Table 3.5.1-1 Item # 22.

b. The main and startup Feedwater control valves are operable to close.

c. The Turbine Driven Emergency Feedwater pump (TDEFWP) is not in RUN or is in RUN but is not aligned to feed the steam generators.

d. MSLB detection analog channels are operable per Specification Table 3.5.1.1 Item # 20.

e. MS-93 (steam admission valve to TDEFWP) is operable to close or is isolated.

f. The associated Main Feedwater pump trip circuitry is operable.

g. The MSLB testing requirements of Technical Specification Tables 4.1-1 and 4.1-2 are met.

Main Feedwater main and startup control valves must remain operable to close even under conditions below the main steam header pressure of 700 psig. To protect against overpressurization of containment during a MSLB inside containment when the MSLB detection and Feedwater isolation circuitry is disabled, the specification requires that control valves be operable to close when RCS temperature is greater than 250 °F. 250 °F is a sufficiently low temperature to ensure that no significant energy release will occur in the event of a MSLB inside the reactor building.

The function of closing the Main Feedwater main and startup block valves is not credited in the MSLB analysis for mitigation of containment overpressurization. Therefore, no operability requirements for these valves are specified.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
60. Core Exit Thermocouples	MO	NA	18 months(1)	(1)A one-time extension of the calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
61. Subcooling Monitors	MO	18 months(1)	18 months(1)	(1)A one-time extension of the channel test and calibration frequency to a maximum of 24 months is allowed for Oconee Unit 2 during operating cycle 16.
62 Main Steam Header Pressure and MSLB detection (analog) channels	ES	MO(1)	18 months	(1) Testing will be performed every 18 months until modifications are implemented to allow for monthly testing.
63 Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	18 months	NA	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
STB - STAGGERED TEST BASIS

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	18 months(4)
3.	Main Steam Safety Valves	Setpoint	18 months(4)
4.	Refueling System Interlocks(5)	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System (2) Leakage	Evaluate	Daily
7.	Emergency Condenser(6) Circulating Water System Test	Functional	18 months
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	18 months
12.	MSLB Feedwater Isolation (7) Feature	Functional	18 months
13.	Essential Siphon Vacuum (8) System Test	Functional	Quarterly

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

(4) Number of safety valves to be tested every 18 months shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

(7) Verification that Main Feed Pumps, Main Feedwater Control Valves, and Turbine Driven Emergency Feedwater Pumps are appropriately actuated/inhibited by the MSLB Feedwater Isolation Feature.

(8) Applicability of these surveillances for each Oconee unit will begin following completion of the Service Water upgrade on the respective unit.