April 15, 1996

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

SUBJECT: ISSUANCE OF AMENDMENTS - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M94288, M94289, AND M94290)

Dear Mr. Hampton:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 216 , 216 , and 213 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55, respectively, for the Oconee Nuclear Station, Units 1, 2, and 3. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated December 7, 1995.

The amendments revise Secondary Decay Heat Removal TS 3.4.2 and TS Table 4.1-1 to delete the requirement of having the main feedwater pump discharge header pressure switch provide an input to actuate the Anticipatory Reactor Trip System and Emergency Feedwater System.

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,
Original signed by:

Leonard A. Wiens, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 216 to DPR-38
2. Amendment No. 216 to DPR-47
3. Amendment No. 213 to DPR-55
4. Safety Evaluation

cc w/encl: See next page

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

April 15, 1996

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Vice President, Oconee Site
Duke Power Company
P. O. Box 1439
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4. Safety Evaluation

cc w/encl: See next page

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

DUKE POWER COMPANY

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
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 December 7, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-38 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 216, 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 from the date of issuance.

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: April 15, 1996

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

DUKE POWER COMPANY

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 216
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 December 7, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Facility Operating License No. DPR-47 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 216, 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 from the date of issuance.

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: April 15, 1996

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

DUKE POWER COMPANY

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 213
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 December 7, 1995, 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:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 213, 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 from the date of issuance.

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: April 15, 1996

<u>ATTACHMENT TO LICENSE AMENDMENT NO. 216</u>

<u>FACILITY OPERATING LICENSE NO. DPR-38</u>

<u>DOCKET NO. 50-269</u>

<u>AND</u>

<u>TO LICENSE AMENDMENT NO. 216</u>

<u>FACILITY OPERATING LICENSE NO. DPR-47</u>

<u>DOCKET NO. 50-270</u>

<u>AND</u>

<u>TO LICENSE AMENDMENT NO. 213</u>

<u>FACILITY OPERATING LICENSE NO. DPR-55</u>

<u>DOCKET NO. 50-287</u>

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.

<u>Remove Pages</u>	<u>Insert Pages</u>
3.4-1	3.4-1
3.4-2	3.4-2*
3.4-3	3.4-3
4.1-8	4.1-8

*No change - overflow page

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

3.4.1 The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam driven pump capable of being driven from an operable steam supply system and two motor driven pumps) and associated manual initiation circuitry shall be operable.

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry associated with loss of main feedwater pumps as sensed by low hydraulic oil pressure shall be operable.

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

b. One turbine driven emergency feedwater pump or one emergency feedwater flow path may be inoperable for a period of up to 72 hours. If the inoperable pump or flow path is not restored to operable status within 72 hours the unit will be at hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

d. With three emergency feedwater pumps and/or both emergency feedwater flow paths inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump and associated emergency feedwater flowpath to operable status. The unit shall be at hot shutdown within 12 hours and below 250°F in another 12 hours if one emergency feedwater pump and associated flowpath are not restored to operable status.

e. If an emergency feedwater pump is inoperable due only to automatic initiation circuitry as specified by 3.4.2, the additional provisions of 3.4.3 a, b, c, and d which require cooldown of the RCS do not apply.

3.4.4 The 16 main steam safety relief valves shall be operable.

3.4.5 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell. A minimum of 6 ft. (=30,000 gal.) shall be available in the upper surge tank.

3.4.6 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

Bases

The Main Feedwater System and the Turbine Bypass System are normally used for decay heat removal and cooldown above 250°F. Feedwater makeup is supplied by operation of a hotwell pump, condensate booster pump, and a main feedwater pump.

Operability of the Emergency Feedwater System (EFW) assures the capability to remove decay heat and cool down the Reactor Coolant System to the operating conditions for switch over to decay heat removal by the Decay Heat Removal System, in the event that the Main Feedwater System is inoperable. The EFW system consists of a turbine driven pump (880 gpm), two motor driven pumps (450 gpm each), and associated flow paths to the steam generators.

The limiting transient requiring maximum EFW flow is the loss of main feedwater with offsite power available. For this transient, a minimum EFW flow rate equivalent to 400 gpm at 1050 psia and no more than 130°F is adequate. Each of the three EFW pumps is capable of delivering this flow.

A 100% flowpath is defined as: The flowpath to either steam generator including associated valves and piping capable of being supplied by either the turbine driven pump or the associated motor driven pump.

One flow indicator or steam generator level indicator per steam generator is sufficient to provide indication of emergency feedwater flow to the steam generators and to confirm emergency feedwater system operation. In the event that at least one indicator per steam generator is not available, then the flowpath to this steam generator is considered to be inoperable.

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as sensed by low hydraulic oil pressure. This specific automatic initiation logic is placed in service prior to criticality and may be bypassed when shutdown to present inadvertent actuation during startup and shutdown. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

Normally, decay heat is removed by steam relief through the Turbine Bypass System to the condenser. Decay heat also can be also removed from the steam generators by steam relief through the main steam safety relief valves. The total relief capacity of the 16 main steam safety relief valves is 13,105,000 lbs./hr. In this case the minimum amount of water in the upper surge tank, condensate storage tank, and hotwell is sufficient to remove decay heat for at least 4 hours at hot shutdown conditions. This provides adequate time to establish normal flow through the condenser by restarting a Condenser Cooling Water (CCW) pump in a loss of station power events. The turbine bypass valves can then be utilized to relieve steam to the condenser and commence a cooldown of the RCS.

A 6 foot level in the upper surge tank will ensure that 30,000 gallons of water are available to the EFW pumps from that source. The 6 foot level setpoint includes an allowance for instrument error and for the depletion of inventory while switching to an alternate suction source.

Table 4.1-1 (CONTINUED)

Channel Description		Check	Test	Calibrate	Remarks
49.	Emergency Feedwater Flow Indicators	MO	NA	RF	
50.	PORV and Safety Valve Position Indicators	MO	NA	RF	
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	RF	
52.	RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
	a) Control Oil Pressure Switches	NA	45 Days STB	RF	
53.	Emergency Feedwater Initiation Circuits				
	a) Control Oil Pressure Switches	NA	MO	RF	
54.	Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II.F.1.3

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 216 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 213 TO FACILITY OPERATING LICENSE DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

By letter dated December 7, 1995, Duke Power Company, (the licensee), submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3, Technical Specifications (TS). The requested changes would revise TS 3.4.2 and TS Table 4.1-1 of the Secondary Decay Heat Removal TS to delete the requirement of having the main feedwater pump discharge header pressure switch provide an input to actuate the Anticipatory Reactor Trip System (ARTS) and Emergency Feedwater System (EFDW).

2.0 BACKGROUND

The ARTS sub-system was added to the Reactor Protection System (RPS) in 1980 in response to NRC IE Bulletin 79-05B and NRC Commission orders. The ARTS sub-system monitors main feedwater and main turbine-generator parameters to provide anticipatory reactor shutdown following a loss of all main feedwater or a main turbine-generator trip.

The primary purpose of the ARTS is to reduce challenges to the pressurizer Power Operated Relief Valves (PORVs), to lift and then reset following a loss of main feedwater or a main turbine-generator trip transient. Both the main feedwater pump turbines and the main turbine-generator trips provide input to the ARTS via low trip oil pressure. Also, there is an input to the ARTS from main feedwater pumps low pump discharge header pressure.

The Emergency Feedwater System (EFDW) is designed to start automatically in the event of loss of main feedwater pumps as sensed by either:

- Main Feedwater Pumps Tripped (Control Oil pressure below 75 psig)
- Main Feedwater Pumps Discharge Pressure below 800 psig
- Main Turbine-Generator Tripped (Emergency Trip Header Oil Pressure below 800 psig), or
- Low water level in either steam generator for 30 seconds.

3.0 EVALUATION

The licensee proposes to remove the anticipatory reactor trip and emergency feedwater actuation signal from loss of main feedwater pumps sensed by feedwater discharge pressure of less than 800 psig. The low feedwater discharge header pressure switches inputs would be removed from both ARTS and EFDW in order to:

- Eliminate the D Heater Drain Pumps (HDPs) discharge pressure from preventing the main feedwater discharge pressure switches from providing input to the ARTS and EFDW systems. After Unit 3 tripped from a loss of feedwater, the D HDPs maintained a 770 psig pressure in the dead headed main feedwater discharge header. Therefore, the original pressure switch setpoint of 750 psig was not reached and there was no input to either ARTS or EFDW for a loss of main feedwater. (Reported by Licensee Event Report 91-009-00 and 91-009-01)

- Preclude spurious reactor trips during pressure switch calibration,

- Maintain a design similar to all other B&W design plants, and

- Save approximately $300,000 over the life of the plant.

The justification for removing the low feedwater pump discharge pressure switches as inputs to ARTS and EFDW, is that another diverse input is provided. This diverse input is the detection of the main feedwater pumps tripped (control oil pressure below 75 psig). After the TMI-2 incident, the NRC required all B&W plants to increase the reactor coolant pressure setpoint for lifting the PORVs and lower the pressure setpoint for tripping the reactor. These reactor coolant pressure setpoint changes also reduced the probability of challenging the PORVs. The main turbine-generator trip (Emergency Trip Header Oil Pressure below 800 psig) and the main feedwater pumps tripped (control oil pressure below 75 psig) input to ARTS will acceptably reduce the probability of challenging the PORVs. Further diverse input to EFDW is provided by:

- Main Turbine-Generator Tripped (Emergency Trip Header Oil Pressure below 800 psig), or

- Low water level in either steam generator for 30 seconds.

4.0 STAFF CONCLUSION

Based on our review of the licensee's request to remove the main feedwater pump discharge header pressure switch inputs to the ART and EFDW actuation, the staff concludes that sufficient diversity is provided to assure that the probability of challenging the PORVs is not changed and the EFDW system is not adversely affected. Therefore, this TS change is acceptable.

5.0 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.

6.0 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 (61 FR 1628 dated January 22, 1996). 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.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: F. Paulitz

Date: April 15, 1996