Docket

Docket Nos.: 50-413
and 50-414

August 7, 1987

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

Subject: Issuance of Amendment No. 29 to Facility Operating License NPF-35 and Amendment No. 20 to Facility Operating License NPF-52 - Catawba Nuclear Station, Units 1 and 2 (TACS 65551/65552)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License NPF-35 and Amendment No. 20 to Facility Operating License NPF-52 for the Catawba Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated June 10, 1987, as supplemented June 11 and 16, 1987.

The amendments modify the Technical Specifications to allow the extension, on a one-time basis, of several 18-month surveillance intervals until the first refueling outage for Unit 2.

A copy of the related safety evaluation supporting Amendment No. 29 to Facility Operating License NPF-35 and Amendment No. 20 to Facility Operating License NPF-52 is enclosed.

Notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/s/

Kahtan Jabbour, Project Manager
Project Directorate II-3
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 29 to NPF-35
2. Amendment No. 20 to NPF-52
3. Safety Evaluation

cc w/encl:
See next page

DISTRIBUTION: *SEE PREVIOUS CONCURRENCE
See attached page

PDII-3/DRPI/II	PDII-3/DRPI/II	PDII-3/DRPI/II
*MDuncan/rad	*KJabbour	*BJYoungblood
07/ /87	07/ /87	07/ /87

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Mr. H. B. Tucker
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Catawba Nuclear Station

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

DUKE POWER COMPANY

NORTH CAROLINA ELECTRIC MEMBERSHIP CORPORATION

SALUDA RIVER ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-413

CATAWBA NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. NPF-35

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-35 filed by the Duke Power Company acting for itself, North Carolina Electric Membership Corporation and Saluda River Electric Cooperative, Inc., (licensees) dated June 10, 1987, as supplemented June 11 and 16, 1987, 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, as amended, 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 rules and 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;

   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 attachments to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-35 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 29, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B.J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: August 7, 1987

ICSB
8/4/87

OTSB
8/6/87

PDII-3/DRPI/II	PDII-3/DRPI/II	OGC-Bethesda	PDII-3/DRPI/II
MDuncan/rad	KJabbour		BJYoungblood
07/23/87	07/23/87	07/27/87	07/29/87

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

DUKE POWER COMPANY

NORTH CAROLINA MUNICIPAL POWER AGENCY NO. 1

PIEDMONT MUNICIPAL POWER AGENCY

DOCKET NO. 50-414

CATAWBA NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 20
License No. NPF-52

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Catawba Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-52 filed by the Duke Power Company acting for itself, North Carolina Municipal Power Agency No. 1 and Piedmont Municipal Power Agency, (licensees) dated June 10, 1987, as supplemented June 11 and 16, 1987, 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, as amended, 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 rules and 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;

   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 attachments to this license amendment and Paragraph 2.C.(2) of Facility Operating License No. NPF-52 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 20, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. Duke Power Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B.J. Youngblood, Director
Project Directorate II-3
Division of Reactor Projects I/II

Attachment:
Technical Specification Changes

Date of Issuance: August 7, 1987

ICSB
8/4/87

OTSB
RLE 8/6/87

DSHm

PDII-3/DRPI/II	PDII-3/DRPI/II	OGC-Bethesda	PDII-3/DRPI/II
MDuncan/rad	KJabbour		BJYoungblood
07/23/87	07/23/87	07/ /87	07/ /87

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. NPF-35

DOCKET NO. 50-413

AND

TO LICENSE AMENDMENT NO. 20

FACILITY OPERATING LICENSE NO. NPF-52

DOCKET NO. 50-414

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. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 3-1	
3/4 3-14	3/4 3-13
3/4 3-45	3/4 3-46
3/4 3-50	3/4 3-49
3/4 6-54	3/4 6-53
3/4 7-5	3/4 7-6
3/4 8-5	

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be demonstrated to be within its limit at least once per 18 months.* Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

*This surveillance need not be performed for the primary RTD response time testing portion of items 7 and 8 from Table 3.3-2 for Unit 2 until prior to entering STARTUP following the Unit 2 first refueling.

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and interlocks shown in Table 3.3-3 shall be OPERABLE with their Trip Setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3

ACTION:

a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Values Column of Table 3.3-4, either:

   1. Adjust the Setpoint consistent with the Trip Setpoint value of Table 3.3-4, and determine within 12 hours that Equation 2.2-1 was satisfied for the affected channel, or

   2. Declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3.3 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value.

   Equation 2.2-1 $Z + R + S \leq TA$

Where:

Z = The value from Column Z of Table 3.3-4 for the affected channel,

R = The "as measured" value (in percent span) of rack error for the affected channel,

S = Either the "as measured" value (in percent span) of the sensor error, or the value from Column S (Sensor Error) of Table 3.3-4 for the affected channel, and

TA = The value from Column TA (Total Allowance) of Table 3.3-4 for the affected channel.

c. With an ESFAS instrumentation channel or interlock inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the Engineered Safety Features Actuation System Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months.* Each test shall include at least one train such that both trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once per N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

*This surveillance need not be performed for items 7.a, 7.b, and 8 from Table 3.3-5 on Unit 2 until prior to entering HOT SHUTDOWN following the Unit 2 first refueling.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. Feedwater Isolation (Continued)								
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. T_{avg}-Low (P-4 Interlock)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
d. Doghouse Water Level-High	N.A.	N.A.	N.A.	R(4)	N.A.	N.A.	N.A.	1, 2
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Turbine Trip								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
c. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
d. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	R(4)	N.A.	N.A.	N.A.	1, 2
e. Reactor Trip (P-4)	N.A.	N.A.	N.A.	R(4)	N.A.	N.A.	N.A.	1, 2, 3
f. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
7. Containment Pressure Control System								
a. Start Permissive	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Termination	S	R	M	N.A.	N.A	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A	N.A	1, 2, 3
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power	N.A.	R	N.A.	M(3)	N.A.	N.A.	N.A	1, 2, 3
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2
g. Auxiliary Feedwater Suction Pressure-Low	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
9. Containment Sump Recirculation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. Refueling Water Storage Tank Level - Low Coincident With Safety Injection	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUI( )
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Operation Nuclear Service Water Operation) (Continued)								
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Loss-of-Offsite Power	N.A.	R	N.A.	M(2)	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
16. Auxiliary Building Filtered Exhaust Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
17. Diesel Building Ventilation Operation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Surveillance Requirements.							
18. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Pressurizer Pressure, not P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A	1, 2, 3
c. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Generator Water Level, P-14	S	R	M	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Monthly testing shall consist of voltage sensor relay testing excluding actuation of load shedding diesel start, and time - delay timers.
(3) Monthly testing shall consist of relay testing excluding final actuation of the pumps or valves.
(4) This surveillance need not be performed on Unit 2 until prior to entering STARTUP or HOT STANDBY (as applicable) following the Unit 2 first refueling.

TABLE 3.6-3

DIVIDER BARRIER SEAL
ACCEPTABLE PHYSICAL PROPERTIES

MATERIAL	TENSILE STRENGTH
Membrane Type Seals	
Mk 10	39.7 lbs
Mk 11	39.7 lbs

CONTAINMENT VALVE INJECTION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 Both trains of the Containment Valve Injection Water System shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one train of the Containment Valve Injection Water System inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each train of the Containment Valve Injection Water System shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a (16.2 psig) and has adequate capacity to maintain system pressure for at least 30 days.

4.6.6.2 Each train of the Containment Valve Injection Water System shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve seal injection flow rate is less than 1.7 gpm (Unit 1), 1.6 gpm (Unit 2) for Train A and 1.4 gpm (Unit 1), 1.5 gpm (Unit 2) for Train B with a tank pressure greater than or equal to 45 psig and each automatic valve in the flow path actuates to its correct position on a Containment Pressure-High or a Containment Pressure-High-High test signal.**

**This surveillance which assures that "...the valve seal injection flow rate is less than...1.6 gpm (Unit 2) for Train A and...1.5 gpm (Unit 2) for Train B with a tank pressure greater than or equal to 45 psig" need not be performed on Unit 2 until prior to entering HOT SHUTDOWN following the Unit 2 first refueling.

SURVEILLANCE REQUIREMENTS (Continued)

3) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

4) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

5) Verifying that the isolation valves in the auxiliary feedwater pump suction lines are open and that power is removed from the valve operators on Valves CA-2, CA-7A, CA-9B, and CA-11A and that the respective circuit breakers are padlocked.

b. At least once per 18 months during shutdown by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Auxiliary Feedwater Actuation test signal,

2) Verifying that each motor-driven auxiliary feedwater pump starts as designed automatically upon receipt of an Auxiliary Feedwater Actuation test signal,

3) Verifying that the turbine-driven auxiliary feedwater pump steam supply valves open upon receipt of an Auxiliary Feedwater Actuation test signal,** and

4) Verifying that the valve in the suction line of each auxiliary feedwater pump from the Nuclear Service Water System automatically actuates to its full open position within less than or equal to 15 seconds* on a Loss-of-Suction test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 2 by verifying normal flow to each steam generator.

*Includes 5 second time delay.
**This surveillance need not be performed on Unit 2 until prior to entering HOT STANDBY following the Unit 2 first refueling.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.3 The specific activity of the Secondary Coolant System shall be less than or equal to 0.1 microCurie/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.1 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

f. At least once every 31 days by obtaining a sample of fuel oil in accordance with ASTM-D2276-78, and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM-D2276-78, Method A;

g. At least once per 18 months by:

1) Subjecting the diesel to an inspection, during shutdown, in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service;***

2) Verifying the generator capability to reject a load of greater than or equal to 825 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz;

3) Verifying the generator capability to reject a load of greater than or equal to 5600 kW but less than or equal to 5750 kW without tripping. The generator speed shall not exceed 500 rpm during and following the load rejection;

4) Simulating a loss-of-offsite power by itself,** during shutdown, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 11 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be at 4160 ± 420 volts and 60 ± 1.2 Hz within 11 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, during shutdown, and

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;**

**This surveillance need not be performed until prior to entering HOT SHUTDOWN following the Unit 1 first refueling.

***This surveillance need not be performed on Unit 2 until prior to entering HOT SHUTDOWN following the Unit 2 first refueling.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NPF-35

AND AMENDMENT NO. 20 TO FACILITY OPERATING LICENSE NPF-52

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DUKE POWER COMPANY, ET AL.

INTRODUCTION

By letters dated June 10, 11 and 16, 1987, Duke Power Company, et al., (the licensee) proposed changes to the Technical Specifications (TS) for Catawba Nuclear Station, Unit 2, which would extend, on a one-time basis, by a maximum of 4.5 months (until the first refueling outage currently scheduled for December 30, 1987) several 18-month TS surveillance intervals. This extension is needed because these surveillances can only be performed with the Unit in Hot Shutdown (Mode 4), Cold Shutdown (Mode 5), or Refueling (Mode 6). Although amendments will be issued for both Units 1 and 2, changes are proposed for Unit 2 only. Unit 1 is included only because the Technical Specifications are combined in one document for both Units.

Normally, since refueling outages occur about every 18 months, extension beyond the 18-month interval required by the TSs for such surveillances is usually not necessary. However, due to the extended length of the Unit 2 startup program and cycle 1, the licensee must either request and receive an extension or shut down prior to the first scheduled refueling outage. Similar extension was approved for Catawba Unit 1 by amendments issued July 3, 1986 (Amendment No. 8 for Unit 1 and No. 1 for Unit 2). Unit 2 is currently scheduled to enter its first refueling outage on December 30, 1987. Most of those surveillances must be performed on August 15, 1987, or later. Therefore, the longest extension entails a period of 4.5 months. Furthermore, the tests required will be performed if an outage of sufficient duration occurs prior to the first scheduled refueling outage.

The changes would be accomplished by adding a footnote usually stating that this surveillance need not be performed until prior to entering Hot Shutdown, Hot Standby or Startup, as applicable, following the Unit 2 first refueling outage, and clarifying that the footnote (i.e., the extension) applies to Unit 2 only.

EVALUATION

The particular surveillances and the time at which the surveillance interval (including the 25% grace period allowed by TS 4.0.2) will expire are discussed below.

1. Feedwater Isolation on receipt of a high doghouse water level signal, TS Table 4.3-2, item 5.d. The trip actuating device operational test

would be extended from August 15, 1987, and would be performed prior to entering startup (Mode 2) or Hot Standby (Mode 3), as applicable, following Unit 2 first refueling. There have been no failures of this circuitry and no actuations since preoperational testing.

2. Turbine Trip on loss of all main feedwater pumps, TS Table 4.3-2, item 6.d. The trip actuating device operational test would be extended from August 15, 1987, and would be performed prior to entering Startup (Mode 2) or Hot Standby (Mode 3), as applicable, following Unit 2 first refueling outage. This instrumentation is reliable and has operated satisfactorily due to one challenge after completion of preoperational testing. Therefore, the staff finds this change acceptable.

3. Turbine Trip on reactor trip, TS 4.3-2, item 6.e. The trip actuating device operational test would be extended from August 15, 1987, and would be performed prior to entering Startup (Mode 2) or Hot Standby (Mode 3), as applicable, following Unit 2 first refueling. This instrumentation is reliable and has responded satisfactorily in response to seven challenges. Therefore, the staff finds this change acceptable.

4. Turbine Trip on steam generator water level-high-high, TS 4.3.2.2, Table 3.3-5, item 7.a. The response time test would be extended from August 15, 1987, and would be performed prior to entering Hot Shutdown (Mode 4) following Unit 2 first refueling. This instrumentation is reliable and has responded satisfactorily in response to three challenges. Therefore, the staff finds this change acceptable.

5. Feedwater Isolation on steam generator water level-high-high, TS 4.3.2.2, Table 3.3-5, item 7.b. The response time test would be extended from August 15, 1987, and would be performed prior to entering Hot Shutdown (Mode 4) following Unit 2 first refueling. This instrumentation is reliable and has responded satisfactorily in response to three challenges. Therefore, the staff finds this change acceptable.

6. Feedwater Isolation on a reactor trip coincident with low reactor coolant system average temperature, TS 4.3.2.2, Table 3.3-5, item 8. The response time test would be extended from August 15, 1987, and would be performed prior to entering Hot Shutdown (Mode 4) following Unit 2 first refueling. This instrumentation is reliable and has responded satisfactorily in response to three challenges. Therefore, the staff finds this change acceptable.

7. Turbine-driven Auxiliary Feedwater Pump Steam Supply Valves Surveillance to verify that they open upon receipt of an auxiliary feedwater actuation test signal, TS 4.7.1.2.1b.3). The test would be extended from August 15, 1987, and would be performed prior to entering Hot Standby (Mode 3) following Unit 2 first refueling. These equipment and instrumentations are highly reliable and have responded successfully in response to eleven challenges. Therefore, the staff finds this change acceptable.

8. Containment Valve Injection Water System Surveillance to verify injection flow to containment isolation valves, TS 4.6.6.2. The test would be extended from October 29, 1987, and would be performed prior to entering Hot Shutdown (Mode 4) following Unit 2 first refueling. This system is reliable with a good operating history and only three valves have not been tested. Therefore, the staff finds this change acceptable.

9. Diesel Generator Inspection, during shutdown, in accordance with the manufacturer's recommendations, TS 4.8.1.1.2g.1). The inspection, which is required by TS at least once per 18 months, would be extended from January 9, 1988, and would be performed prior to entering Hot Shutdown (Mode 4) following Unit 2 first refueling. In its June 10, 1987, letter the licensee stated that the time at which the surveillance interval (including the 25% grace period allowed by TS 4.0.2) will expire for this surveillance is January 9, 1988. The currently scheduled shutdown date for Unit 2 is December 30, 1987, and the extension for surveillance 4.8.1.1.2g.1) is needed to allow the diesel inspections to be conducted during the outage and to process the paperwork. The present Unit 2 refueling outage schedule puts diesel generator 2A inspection at a 25 day duration and diesel generator 2B inspection at a 20 day duration. The licensee also stated that the inspections are of a nature that they could not be accomplished within the three day TS time limitation allowed for an inoperable diesel generator during power operation.

   Furthermore, the licensee stated that, of a total of 296 starts on diesel generator 2A there have been 6 valid failures, and of a total of 212 starts on diesel generator 2B there have been 2 valid failures. This makes for a reliability of approximately .98 on diesel generator 2A and approximately .99 on diesel generator 2B. To date, diesel generator 2A has a total run accumulation of 440 hours and diesel generator 2B a total run accumulation of 469 hours or approximately 350-400 hours since major teardown and inspection. The licensee also stated that all lube oil pressures and temperatures continue to be normal. Jacket water pressure and temperature continue to be normal. All cylinder temperatures, firing pressures, and cold compression pressures continue to be normal. Ferrograph and spectrograph results on engine oil continue to be satisfactory with no indication of unusual trends that would indicate abnormal conditions or wear. Routine surveillance activities will continue to be conducted on schedule and on time.

   Based upon the relatively short length of time required for this surveillance interval extension and the satisfactory operating history of the Unit 2 diesel generators described above, the staff finds that this one-time extension of the inspection interval for Unit 2 diesels will have a negligible impact on plant safety and is, therefore, acceptable.

10. System Response Time tests for the primary RTDs associated with the Overtemperature Delta T and Overpower Delta T Reactor Trips, TS 4.3.1.2, Table 3.3-2, items 7. and 8. These tests would be extended from September 11, 1987, and would be performed prior to entering Startup (Mode 2) following Unit 2 first refueling. Only the RTDs remain to be tested. The rest of the circuitry has been tested satisfactorily within the required surveillance interval. These RTDs are reliable and have successfully met the required response time during the last three previous tests (two on Unit 1 and one one Unit 2). Therefore, the staff finds this change acceptable.

Based on the above evaluations, the staff finds that the licensee's proposed one-time extension of the surveillance intervals for the tests listed above would not pose an undue risk to public health and safety, and therefore is acceptable.

ENVIRONMENTAL CONSIDERATION

The amendments involve a change in use of facility components located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The 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 exposures. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there have been no public comments on such finding. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (52 FR 24548) on July 1, 1987, and consulted with the state of South Carolina No public comments were received, and the state of South Carolina did not have any comments.

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

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