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

February 29, 1984

Docket Nos.: 50-369
and 50-370

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-9 and Amendment No. 10 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License NPF-9 and Amendment No. 10 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated August 2, 1983. The remaining changes requested in that letter will be addressed in a later amendment.

The amendments change the Technical Specifications related to the response time for steam line isolation, deletion of mechanical snubbers, inoperable steam generator instrumentation channels, surveillance requirements for sprinkler system valves, and correction of typographical errors.

A copy of the related safety evaluation report supporting Amendment No. 29 to Facility Operating License NPF-9 and Amendment No.10 to Facility Operating License NPF-17 is enclosed.

Sincerely,

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Enclosures:
1. Amendment No. 29 to NPF-9
2. Amendment No. 10 to NPF-17
3. Safety Evaluation

cc w/encl:
See next page

McGuire

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. NPF-9

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 1 (the facility) Facility Operating License No. NPF-9 filed by the Duke Power Company (licensee) dated August 2, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 regulations of the Commission;

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

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

   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-9 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No.29, are hereby incorporated into this license. The licensee 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

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: February 29, 1984

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

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. NPF-17

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

   A. The application for amendment to the McGuire Nuclear Station, Unit 2 (the facility) Facility Operating License No. NPF-17 filed by the Duke Power Company (licensee) dated August 2, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's 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 regulations of the Commission;

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

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

   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-17 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 10, are hereby incorporated into this license. The licensee 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

Elinor G. Adensam

Elinor G. Adensam, Chief
Licensing Branch No. 4
Division of Licensing

Attachment:
Technical Specification
Changes

Date of Issuance: February 29, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 29

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. NPF-17

DOCKET NO. 50-370

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 3-3	3/4 3-4
3/4 3-7	3/4 3-8
3/4 3-24	3/4 3-23
3/4 3-31	
3/4 3-32	
3/4 3-77	3/4 3-78
3/4 4-30	3/4 4-29
3/4 7-27	
3/4 7-28	
3/4 7-35	3/4 7-36
3/4 7-38	3/4 7-37
3/4 7-42	3/4 7-41
3/4 10-2	3/4 10-1

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Overpower ΔT					
Four Loop Operation	4	2	3	1, 2	6#
Three Loop Operation	(**)	(**)	(**)	(**)	(**)
9. Pressurizer Pressure-Low	4	2	3	1	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1	7#
12. Low Reactor Coolant Flow					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	7#
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	7#
13. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
14. Undervoltage-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6#
15. Underfrequency-Reactor Coolant Pumps (above P-7)	4-1/bus	2	3	1	6#
16. Turbine Trip					
a. Low Fluid Oil Pressure	3	2	2	1	7#
b. Turbine Stop Valve Closure	4	4	1	1	11#
17. Safety Injection Input from ESF	2	1	2	1, 2	9
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Low Setpoint Power Range Neutron Flux, P-10	4	2	3	1, 2	8
e. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement suspend all operations involving positive reactivity changes.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1 and Specification 4.3.2.1.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor trip breakers within the next hour.

ACTION 11 - With the number of OPERABLE channels less than the Total Number of Channels, operation may continue provided the inoperable channels are placed in the tripped condition within 1 hour.

TABLE 3.3-3 (Continued)

TABLE NOTATION

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##Trip function automatically blocked above P-11 and may be blocked below P-11 when Safety Injection on low steam pressure is not blocked.

*The provisions of Specification 3.0.4 are not applicable.

**These values left blank pending NRC approval of three loop operation.

ACTION STATEMENTS

ACTION 14 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 17 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

ACTION 18 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 1 hour, and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 2 hours for surveillance testing of other channels per Specification 4.3.1.1 and Specification 4.3.2.1.

ACTION 20 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

ACTION 22 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 23 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the action required by Specification 3.7.1.4.

ACTION 24 - With the number of OPERABLE channels less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated auxiliary feedwater pump inoperable and take the action required by Specification 3.7.1.2. With the channels associated with more than one auxiliary feedwater pump inoperable, immediately declare the associated auxiliary feedwater pumps inoperable and take the action required by Specification 3.7.1.2.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low-Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(3)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"[(2)]	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	N.A.
f. Auxiliary Feedwater[(5)]	N.A.
g. Nuclear Service Water System	$\leq 76^{(1)}/65^{(3)}$
h. Component Cooling Water	$\leq 76^{(1)}/65^{(3)}$
i. Start Diesel Generators	≤ 11
4. Steam Line Pressure-Low	
a. Safety Injection (ECCS)	$\leq 12^{(3)}/22^{(4)}$
b. Reactor Trip (from SI)	≤ 2
c. Feedwater Isolation	≤ 9
d. Containment Isolation-Phase "A"[(2)]	$\leq 18^{(3)}/28^{(4)}$
e. Containment Purge and Exhaust Isolation	N.A.
f. Auxiliary Feedwater[(5)]	N.A.
g. Nuclear Service Water	$\leq 65^{(3)}/76^{(4)}$
h. Steam Line Isolation	≤ 7
i. Component Cooling Water	$\leq 65^{(3)}/76^{(4)}$
j. Start Diesel Generators	≤ 11
5. Containment Pressure--High-High	
a. Containment Spray	≤ 45
b. Containment Isolation-Phase "B"	N.A.
c. Steam Line Isolation	≤ 7
6. Steam Generator Water Level--High-High	
a. Turbine Trip	N.A.
b. Feedwater Isolation	≤ 13

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7. Steam Generator Water Level - Low-Low	
a. Motor-driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Turbine-driven Auxiliary Feedwater Pumps	$\leq$ 60
8. Negative Steam Line Pressure Rate - High	
Steam Line Isolation	$\leq$ 7
9. Start Permissive	
Containment Pressure Control System	N.A.
10. Termination	
Containment Pressure Control System	N.A.
11. Auxiliary Feedwater Suction Pressure - Low	
Auxiliary Feedwater Pumps (Suction Supply Automatic Realignment)	$\leq$ 13
12. RWST Level	
Automatic Switchover to Recirculation	$\leq$ 60
13. Station Blackout	
a. Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
b. Start Turbine-Driven Auxiliary Feedwater Pump	$\leq$ 60
14. Trip of Main Feedwater Pumps	
Start Motor-Driven Auxiliary Feedwater Pumps	$\leq$ 60
15. Loss of Power	
4 kV Emergency Bus Undervoltage-Grid Degraded Voltage	$\leq$ 11

TABLE 4.3-9 (Continued)

TABLE NOTATION

*At all times except when the isolation valve is closed and locked.

**During WASTE GAS HOLDUP SYSTEM operation.

(1) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm/Trip Setpoint,

b. Circuit failure (alarm only), and

c. Instrument indicates a downscale failure (alarm only).

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

a. Instrument indicates measured levels above the Alarm Setpoint,

b. Circuit failure, and

c. Instrument indicates a downscale failure.

(3) The initial CHANNEL CALIBRATION shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration shall be used.

(4) The CHANNEL CALIBRATION shall include the use of standard gas samples corresponding to alarm setpoints in accordance with the manufacturer's recommendations.

(5) The CHANNEL CALIBRATION shall include the use of standard gas samples in accordance with the manufacturer's recommendations. In addition, a standard gas sample of nominal 4 volume percent oxygen, balance nitrogen, shall be used in the calibration to check linearity of the oxygen analyzer.

LOOSE-PART DETECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.3.3.10 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

a. With one or more Loose-Part Detection System channels inoperable for more than 30 days, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 10 days outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.10 Each channel of the Loose-Part Detection Systems shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 24 hours, and

b. An ANALOG CHANNEL OPERATIONAL TEST except for verification of Setpoint at least once per 31 days, and

c. A CHANNEL CALIBRATION at least one per 18 months.

TABLE 4.4-4 (Continued)

TABLE NOTATION

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F for Unit 1 and 60°F for Unit 2 in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

TABLE 3.7-4b (Continued)

SAFETY-RELATED MECHANICAL SNUBBERS*

PACIFIC SCIENTIFIC

SYSTEM**	SMALL SIZE (350 Lbs. or Less TO 600 Lbs)	MEDIUM SIZE (1,487 Lbs. to 15,000 Lbs)	LARGE SIZE (50,000 Lbs. to 120,000 Lbs)
SV	0	3	0
VE	1	3	0
VI	30	1	0
VQ	0	2	0
WG	2	0	0
WL	8	0	0
WS	2	0	0
YC	1	1	0
Subtotal (Unit 1)	326	282	23
UNIT 2			
BB	67	55	0
CA	9	59	0
CF	13	80	8
FW	0	4	1
KC	66	81	0
KD	3	1	0
KF	2	5	0
LD	0	2	0
NB	5	1	0
NC	100	95	2
ND	29	41	0

*Snubbers may be added or deleted without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request. In lieu of any other report required by Specification 6.9.1, at least 15 days prior to the deletion of any listed snubber, a Special Report shall be prepared and submitted to the Commission in accordance with Specification 6.9.2 evaluating the safety significance of the proposed snubber removal.

**A listing of individual snubbers and more detailed information shall be available for NRC review at the McGuire Nuclear Station.

TABLE 3.7-4b (Continued)

SAFETY-RELATED MECHANICAL SNUBBERS*

PACIFIC SCIENTIFIC

SYSTEM**	SMALL SIZE (350 Lbs. or Less TO 600 Lbs)	MEDIUM SIZE (1,487 Lbs. to 15,000 Lbs)	LARGE SIZE (50,000 Lbs. to 120,000 Lbs)
NF	2	2	0
NI	56	55	3
NM	42	11	0
NR	10	8	0
NV	170	69	0
RF	1	3	0
RN	28	34	1
RV	9	8	0
SA	9	12	0
SM	2	24	30
SV	0	1	0
TE	1	3	0
VE	3	2	0
VG	1	2	0
VI	26	1	0
VN	0	4	0
VQ	3	1	0
VS	1	0	0
VX	2	1	0
WL	15	7	0
WN	0	2	0
Subtotal (Unit 2)	675	674	45
TOTAL for UNITS 1 and 2	1,001	956	65

*Snubbers may be added or deleted without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request. In lieu of any other report required by Specification 6.9.1, at least 15 days prior to the deletion of any listed snubber, a Special Report shall be prepared and submitted to the Commission in accordance with Specification 6.9.2 evaluating the safety significance of the proposed snubber removal.

**A listing of individual snubbers and more detailed information shall be available for NRC review at the McGuire Nuclear Station.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish a hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path which is accessible during plant operation is in its correct position,

b. At least once per 12 months, by cycling each testable valve in the flow path through at least one complete cycle of full travel,

c. At least once per 18 months:

   1) By performing a system functional test which includes simulated automatic actuation of the system, and:

      a) Verifying that the automatic valves in the flow path actuate to their correct position on a Fire Detection test signal, and

      b) Cycling each valve in the flow path that is not testable during plant operation through at least one complete cycle of full travel.

   2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

   3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

   4) By verifying that each valve (manual, power-operated, or automatic) in the flow path which is inaccessible during plant operation is in its correct position.

d. At least once per 3 years, by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.10.3 The following Halon Systems shall be OPERABLE:

a. Elevation 716 ft. - Auxiliary Building

Room No.	Equipment
600 B	Turbine Driven Aux. FW Pump - Unit 1
601B	Turbine Driven Aux. FW Pump - Unit 2

b. Elevation 733 ft. - Auxiliary Building

Room No.	Equipment
703-704	Diesel Generators - Unit 1
714-715	Diesel Generators - Unit 2

APPLICABILITY: Whenever equipment protected by the Halon System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Halon Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.3 Each of the above required Halon Systems shall be demonstrated OPERABLE:

a. At least once per 31 days, by verifying that each valve (manual, power-operated, or automatic) in the flow path is in its correct position,

b. At least once per 6 months, by verifying Halon storage tank weight to be at least 95% of full charge weight and pressure to be at least 90% of full charge pressure, and

c. At least once per 18 months, by:

1) Verifying the system, including associated ventilation dampers and fire door release mechanisms, actuates manually and automatically, upon receipt of a simulated actuation signal, and

2) Performance of a flow test through headers and nozzles to assure no blockage.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.10.4 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-5 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the OPERABLE hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above action shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.10.4 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station,

b. At least once per 18 months, by:

   1) Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station,

   2) Removing the hose for inspection and re-racking, and

   3) Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years, by:

   1) Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage, and

   2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

Table 3.7-5

FIRE HOSE STATIONS

NO.	LOCATION	ELEVATION (FT)
157	55-FF	695
180	52-CC	716
181	54-GG	716
176	52-CC	716
175	52-CC	716
893	40-AA	733
891	40-CC	733
892	43/44-DD	733
895	46 BB	733
889	51/52-DD	733
888	52-EE	733
171	54-HH	733
167	51-JJ	733
168	52-MM	733
169	55-NN	733
962	46-CC	750
964	51-CC	750
965	54-BB	750
966	56-DD	750
971	58-BB	750
303	52-GG	750
162	54-LL	750
161	50-MM	750
164	56-QQ	750
974	51-BB	767
191	56-GG	767
184	54-KK	767
186	51-MM	767
158	57-FF	695
183	59-CC/DD	716
182	58-GG	716
178	58/59-MM	716
179	61-LL	733
900	68-AA/BB	733
901	72-DD	733
902	68/69-DD	733
903	72-DD	733
914	66-BB	733
898	61-BB	733
899	66-BB	733
897	60-DD	733
896	60-EE	733
904	58-CC/DD	733
172	58-HH	733
174	61-JJ/KK	733
170	57-LL	733

SURVEILLANCE REQUIREMENTS (Continued)

4.7.11.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. That each unlocked fire door without electrical supervision is closed at least once per 24 hours,

b. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours and performing a functional test at least once per 18 months,

c. That each locked closed fire door is closed at least once per 7 days, and

d. The OPERABILITY of the Fire Door Supervision System for each electrically supervised fire door by performing a TRIP ACTUATING DEVICE OPERATIONAL TEST at least once per 31 days.

3/4.7.12 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.12 The temperature of each area shown in Table 3.7-6 shall be maintained within the limits indicated in Table 3.7-6.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-6:

a. For more than 8 hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to within its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.12 The temperature in each of the areas shown in Table 3.7-6 shall be determined to be within its limit at least once per 12 hours.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full-length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.2 and 4.2.2.3, and

b. Specification 4.2.3.2.

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

SAFETY EVALUATION REPORT

RELATED TO AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NPF-9

AND TO AMENDMENT NO.10 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

INTRODUCTION

In a letter dated August 2, 1983, the licensee submitted a number of proposed changes to the Technical Specifications for the McGuire Nuclear Station. The submittal included changes relating to (1) response time requirements for steamline isolation valves and reactor trip system steam generator water low-low level trip function, (2) fire protection sprinkler system valve surveillance requirements, (3) snubber surveillance requirements and (4) correction to administrative and typographical changes.

EVALUATION

Steamline Isolation Valves/Steam Generator Level Trip

Technical Specification Table 3.3-5, items 4(h), 5(c) and 8, specify the steamline isolation response time. The licensee has proposed to change the response time limit for steamline isolation from <9 seconds to <7 seconds. This change will make the Technical Specification consistent with the McGuire Final Safety Analysis Report (FSAR) Chapter 6 steamline break containment analysis and the Chapter 15 steamline break core analysis assumptions. Because the proposed change involves operation under a more restrictive requirement than was previously specified, and is consistent with the FSAR analyses, the staff finds the proposed change acceptable.

Item 13 of Table 3.3-1 of the McGuire Technical Specifications includes requirements for operation with an inoperable channel of steam generator level instrumentation. The licensee has proposed to change the action statement from 7 to 6. Action statement 7 requires an inoperable channel to be placed in the tripped condition within 1 hour and only permits continued operation until the next required surveillance test for these channels. With an inoperable channel tripped, surveillance testing of the remaining channels cannot be performed since testing places a channel in trip. Action Statement 6 allows an inoperable channel to be placed in the bypassed condition up to 2 hours for surveillance testing of the other channels. The staff finds this change to be acceptable since it is consistent with the action statement in Table 3.3-3 on Engineered Safety Features Actuation System for the same instrument channels and for the similar channels of the Reactor Trip System for which this action statement applies.

The licensee has also proposed to add a phrase "and specification 4.3.2.1" at the end of action statement number 6 on page 3/4 3-7 and action statement number 19 on page 3/4 3-24, which would clarify the fact that some of the instrumentation performs dual functions for reactor trip and engineered safety features actuation. The staff finds this clarification to be acceptable.

Fire Protection Sprinkler System

The fire protection sprinkler system valve surveillance requirement, section 4.7.10.2.a of the Technical Specification currently requires verification at least once per 31 days that each valve in the flow path of a spray or sprinkler system is in its correct position. The licensee proposes to verify the position of nine of these valves which are located in areas inaccessible during plant operation at least once every 18 months.

Justification for the proposed change is based on the following:

a. Each valve is either locked in the current position or electrically supervised in accordance with NUREG-0800. Therefore, inadvertent misposition of these valves is unlikely.

b. Containment entry is controlled and limited. Therefore, the opportunity to misposition these valves is minimized.

Because each valve is supervised in the correct position and containment entry is controlled and limited, which minimizes the opportunity to misposition these valves, we find that reasonable assurances exist that the valves will not be inadvertently mispositioned during an 18 month surveillance cycle. Based on our evaluation, we conclude that the proposed change is acceptable.

Snubber Surveillance

Safety-related mechanical snubbers are indicated in Technical Specification Table 3.7-4b. The Table allows the licensee to add or delete snubbers without prior License Amendment provided that a revision to the Table is included with the next license amendment request. The licensee proposes the deletion of one small-size mechanical snubber on the Unit 2 diesel generator lube oil system and one medium size snubber on the Unit 2 safety injection system. The piping mathematical model for the small-size snubber was reanalyzed by the licensee without this snubber in place according to ASME code requirement. The results of the analysis showed that allowable stresses can be met without the snubber. The licensee also proposes to replace the medium size snubber with a rigid support. The rigid support will be designed to withstand seismic loading as necessary to compensate for the deleted snubber. The licensee has analyzed stresses caused by thermal movements and determined the stresses to be within acceptable limits. This is an acceptable and approved procedure for snubber deletion/replacement and based on our evaluation we conclude that the proposed mechanical snubber changes in Table 3.7-4b are acceptable.

Administrative and Typographical Changes

The licensee has proposed correcting several Technical Specification administrative and typographical errors. Briefly these changes are described below.

Specification 3.4.9.1.a (page 3/4 4-30) is proposed to be changed due to an inconsistency with Figure 3.4-2b. This figure is based upon a maximum heatup rate for Unit 2 of 60°F per hour.

Table 4.3-9, Note (2) (page 3/4 3-77) is proposed to be changed due to an administrative error. The current note (2) contains the word "and" after part c which implies that part d is missing. In fact, part d in the Standardized Technical Specifications was intentionally deleted when the McGuire document was developed because it did not apply.

Two typographical errors in Table 3.7-5 (page 3/4 7-38) are proposed to be corrected as shown.

The ACTION section of Specification 3.7.12 (page 3/4 7-42) is proposed to be revised to properly reference the temperature limits shown in Table 3.7-6 and to be consistent with the LCO.

The proposed change to Specification 3.10.2 (page 3/4 10-2) involves deleting a reference to Specification 3.1.3.7 which does not exist.

Based on our review of these matters, we conclude that the proposed changes are acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (48 FR 49717) on October 27, 1983, and consulted with the state of North Carolina. No public comments were received, and the state of North 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Dated: February 29, 1984

February 29, 1984

AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NPF-9 - McGUIRE NUCLEAR STATION, UNIT 1
AMENDMENT NO. 10 TO FACILITY OPERATING LICENSE NPF-17 - McGUIRE NUCLEAR STATION, UNIT 2

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