DISTRIBUTION FOR AMENDMENT 2 TO LICENSE NO. NPF-9
FOR MCGUIRE FUEL LOAD AND ZERO POWER TESTING LICENSE

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Docket No.: 50-369

APR 2 1981

Mr. William O. Parker, Jr.
Vice President - Steam Production
Duke Power Company
P. O. Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Parker:

Subject: Issuance of Amendment No. 2 to License NPF-9 For Fuel Loading and Zero Power Testing for McGuire Nuclear Station, Unit 1

The Nuclear Regulatory Commission has issued Amendment No. 2 to License No. NPF-9 in accordance with your letter, dated February 23, 1981. A copy of this Amendment is enclosed. This amendment revises certain pages of Appendix A to License NPF-9 issued on January 23, 1981 for Fuel Loading and Zero Power Testing for the McGuire Nuclear Station, Unit 1. Also enclosed is a copy of our Safety Evaluation supporting this change and the public notice concerning the issuance of this amendment. The notice has been forwarded to the Office of the Federal Register for publication.

Sincerely,

/s/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 2 to NPF-9 for Fuel Loading and Zero Power Testing
2. Safety Evaluation
3. Federal Register Notice

OFFICE	DL:LB#1	DL:LB#1	LGB/DST	OELD	DL:LB#1		
SURNAME	MRushbrook/	sRABirkel	JWhetmore	KETCHEN	BJYoungblood		
DATE	4/1/81	4/1/81	4/ /81	4/1/81	4/2/81		

NRC FORM 318 (10/80) NRCM 0240

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☆ USGPO: 1980—329-824

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DUKE POWER COMPANY

DOCKET NO. 50-369

MCGUIRE NUCLEAR STATION, UNIT 1

LICENSE FOR FUEL LOADING AND ZERO POWER TESTING

License No. NPF-9
Amendment No. 2

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

   A. The issuance of this amendment to the Duke Power Company for the McGuire Nuclear Station, Unit 1 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the license, 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 regulations;

   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.

8104090128

OFFICE							
SURNAME							
DATE							

2. Accordingly, the license is amended by changes to the Appendix A Technical Specifications as indicated in the attachment to this license amendment. License NPF-9 for Fuel Loading and Zero Power Testing is herey amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 2, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Date of Issuance: APR 2 1981

Enclosure:
Revised pages to Appendix A
Technical Specifications

OFFICE	DL/LB#1	DL/LB#1	DST/LGB	OELD	DL/LB#1		
SURNAME	MRushbrook	RBirkel	JWhetmore	KETCHEN	JYoungblood		
DATE	4/1/81	4/1/81	4/1/81	4/1/81	4/2/81		

ATTACHMENT TO LICENSE AMENDMENT NO. 2

LICENSE FOR FUEL LOADING AND ZERO POWER TESTING, NPF-9

DOCKET NO. 50-369

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 area of change.

Pages

3/4 4-18
3/4 4-19
3/4 4-20
3/4 5-1
3/4 7-60
3/4 8-9
3/4 6-1
2-7
2-9
3/4 1-7
3/4 1-8
3/4 3-50
3/4 3-51
3/4 4-8
3/4 4-9

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.7.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator.

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and

f. 1 GPM leakage at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.7.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.7.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months.

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months.

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	FUNCTION
MC-1562-2.0	
1NI60	Accumulator Discharge
1NI71	Accumulator Discharge
1NI59	Accumulator Discharge
1NI70	Accumulator Discharge
MC-1562-2.1	
1NI182	Accumulator Discharge
1NI94	Accumulator Discharge
1NI81	Accumulator Discharge
1NI93	Accumulator Discharge
MC-1562-3.0	
1NI159	Safety Injection (Hot Leg)
1NI156	Safety Injection (Hot Leg)
1NI128	Safety Injection (Hot Leg)
1NI124	Safety Injection (Hot Leg)
1NI160	Safety Injection (Hot Leg)
1NI157	Safety Injection (Hot Leg)
1NI126	Safety Injection (Hot Leg)
MC-1562-3.1	
1NI165	Safety Injection/Residual Heat Removal (Cold Leg)
1NI167	Safety Injection/Residual Heat Removal (Cold Leg)
1NI169	Safety Injection/Residual Heat Removal (Cold Leg)
1NI171	Safety Injection/Residual Heat Removal (Cold Leg)
1NI175	Safety Injection/Residual Heat Removal (Cold Leg)
1NI176	Safety Injection/Residual Heat Removal (Cold Leg)
1NI180	Safety Injection/Residual Heat Removal (Cold Leg)
1NI181	Safety Injection/Residual Heat Removal (Cold Leg)
MC-1562-4.0	
1NI250	Upper Head Injection
1NI251	Upper Head Injection
1NI252	Upper Head Injection
1NI253	Upper Head Injection
1NI249	Upper Head Injection
1NI248	Upper Head Injection
MC-1561-1.0	
1ND1B*	Residual Heat Removal
1ND2A*	Residual Heat Removal

*Testing per Section 4.4.7.2.2.d not applicable due to positive indication of valve position in Control Room.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1.1 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 8261 and 8496 gallons,

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 400 and 454 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one cold leg injection accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable cold leg injection accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one cold leg injection accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each cold leg injection accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

TABLE 3.7-6

FIRE HOSE STATIONS

LOCATION	ELEVATION
55-FF	695 ft.
52-CC	716 ft.
54-GG	716 ft.
55-MM	716 ft.
51-MM	716 ft.
40-AA	733 ft.
40-CC	733 ft.
43-DD	733 ft.
46-AA	733 ft.
52-DD	733 ft.
52-EE	733 ft.
54-GG	733 ft.
51-JJ	733 ft.
52-MM	733 ft.
55-NN	733 ft.
46-CC	750 ft.
51-CC	750 ft.
52-AA	750 ft.
54-BB	750 ft.
56-DD	750 ft.
58-BB	750 ft.
52-GG	750 ft.
54-LL	750 ft.
50-MM	750 ft.
56-QQ	750 ft.
51-BB	767 ft.
56-GG	767 ft.
54-JJ	767 ft.
51-MM	767 ft.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite essential auxiliary power system, and

b. One diesel generator with:

   1. A day tank containing a minimum volume of 120 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 28,000 gallons fo fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 4.8.1.1.2 and 4.8.1.1.4 except for Requirement 4.8.1.1.2.a.5.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-2 of Specification 3.6.3.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of a penetration subject to Type B testing, except the containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a (14.8 psig) and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

*Except valves, blind flanges, and deactivated automatic valves which are located inside the annulus or containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
20. Power Range Neutron Flux - (P-8) Low Reactor Coolant Loop Flow, and Reactor Coolant Pump Breaker Position	$\leq$ 48% of RATED THERMAL POWER	$\leq$ 49% of RATED THERMAL POWER
21. Power Range Neutron Flux - (P-10) - Enable block of Source, Intermediate, and Power Range (low setpoint) Reactor Trips	$\geq$ 10% of RATED THERMAL POWER	$\geq$ 9% of RATED THERMAL POWER
22. Reactor Trip P-4	Not Applicable	Not Applicable

NOTATION

NOTE 1: Overtemperature $\Delta T \leq \Delta T_o \left[K_1 - K_2 \left(\frac{1+\tau_1 S}{1+\tau_2 S}\right)(T-T') + K_3(P-P') - f_1(\Delta I)\right]$

where: ΔT_o = Indicated ΔT at RATED THERMAL POWER

T = Average temperature, °F

$T' \leq$ 588.2°F Reference T_{avg} at RATED THERMAL POWER

P = Pressurizer pressure, psig

P' = 2235 psig (Nominal RCS operating pressure)

$\frac{1+\tau_1 S}{1+\tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg} τ_1 = 33 secs, τ_2 = 4 secs.

S = Laplace transform operator, sec^{-1}

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (Continued)

Note 2: Overpower $\Delta T \leq \Delta T_o [K_4 - K_5 \left(\frac{\tau_3 S}{1+\tau_3 S}\right) T - K_6 (T - T'') - f_2(\Delta I)]$

where:

ΔT_o	=	Indicated ΔT at RATED THERMAL POWER
T	=	Average temperature, $^{\circ}F$
T''	=	$\leq 588.2^{\circ}F$ Reference T_{avg} at RATED THERMAL POWER
K_4	=	1.0908
K_5	=	$0.02/^{\circ}F$ for increasing average temperature and 0 for decreasing average temperature
K_6	=	$0.00126/^{\circ}F$ for $T > T''$; $K_6 = 0$ for $T \leq T''$
$\frac{\tau_3 S}{1+\tau_3 S}$	=	The function generated by the rate lag controller for T_{avg} dynamic compensation
τ_3	=	Time constant utilized in the rate lag controller for T_{avg} $\tau_3 = 5$ secs.
S	=	Laplace transform operator, sec^{-1}
$f_2(\Delta I)$	=	0 for all ΔI

Note 3: The channel's maximum trip point shall not exceed its computed trip point by more than 2 percent.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the boric acid tanks via a boric acid transfer pump and charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heated portion of the flow path is greater than or equal to 65°F when a flow path from the boric acid tanks is used.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System, and

b. Two flow paths from the refueling water storage tank via a charging pump to the Reactor Coolant System.

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 At least two of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the heated portion of the flow path from the boric acid tanks is greater than or equal to 65°F when it is a required water source.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months during shutdown by verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection (S_s) test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2.a delivers at least 30 gpm to the Reactor Coolant System.

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	2	1
3. Reactor Coolant Pressure - Wide Range	2	1
4. Pressurizer Water Level	2	1
5. Steam Line Pressure	2/steam generator	1/steam generator
6. Steam Generator Water Level - Narrow Range	2/steam generator	1/steam generator
7. Refueling Water Storage Tank Water Level	2	1
8. Auxiliary Feedwater Flow Rate	2/steam generator	1/steam generator
9. Reactor Coolant System Subcooling Margin Monitor	1	0
10. PORV Position Indicator	2/valve	1/valve
11. PORV Block Valve Position Indicator	1/valve	1/valve
12. Safety Valve Position Indicator	2/valve	1/valve
13. In-Core Thermocouples	4/core quadrant	2/core quadrant
14. Containment Water Level	2	1

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Temperature - T_{HOT} and T_{COLD} (Wide Range)	M	R
3. Reactor Coolant Pressure - Wide Range	M	R
4. Pressurizer Water Level	M	R
5. Steam Line Pressure	M	R
6. Steam Generator Water Level - Narrow Range	M	R
7. RWST Water Level	M	R
8. Auxiliary Feedwater Flow Rate	M	R
9. Reactor Coolant System Subcooling Margin Monitor	M	R
10. PORV Position Indicator	M	R
11. PORV Block Valve Position Indicator	M	R
12. Safety Valve Position Indicator	M	R
13. In-Core Thermocouples	M	R
14. Containment Water Level (Wide Range)	M	R

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a water volume of less than or equal to 1600 cubic feet (equivalent to an indicated level of less than or equal to 92% on narrow range instrument), and at least two groups of pressurizer heaters each having a capacity of at least 150 kw.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one group of pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 The pressurizer water volume shall be determined to be within its limit at least once per 12 hours.

4.4.4.2 The capacity of each of the above required groups of pressurizer heaters shall be verified at least once per 92 days.

4.4.4.3 The emergency power supply for the pressurizer heater shall be demonstrated OPERABLE at least once per 18 months by manually transferring power from the normal to the emergency power supply and energizing the heaters.

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.5 All power operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the valve through one complete cycle of full travel.

4.4.3.2.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.3.2.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power supply, and

b. Operating the valves through a complete cycle of full travel.

APR 2 1981

SAFETY EVALUATION BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 2
TO LICENSE NPF-9
DUKE POWER COMPANY

Introduction

By letter dated February 23, 1981, the licensee requested certain revisions to the McGuire Nuclear Station Unit 1 Technical Specifications Appendix A to License NPF-9. These revisions included the following items:

1. Correction of several typographical errors.
2. Revise reactor coolant system isolation valve identification and testing requirements.
3. Increase the allowable concentration of oxygen in the waste gas holdup system and eliminate the requirements for a hydrogen monitor in the waste gas holdup system.
4. Allow one safety injection pump to be operable whenever the temperature of one or more of the reactor coolant legs is less than or equal to 300 F.

The McGuire Unit 1 Technical Specifications were issued on January 23, 1981 as an integral part of the fuel-load, zero power operating license and are standard Westinghouse-PWR technical specifications reflecting plant specific design.

Evaluation

The McGuire Nuclear Station utilizes the standard Westinghouse-PWR technical specifications and as such have been developed to reflect the McGuire design features. During this development changes to the specifications are required to reflect editorial matters such as typographical errors and clarifications to improve technical intent. We have reviewed and evaluated the licensee proposed changes and find that items (1) and (2) better reflect the intent of the specifications. Item (2) has been changed to revise valve identification and allow leakage testing prior to entering Mode 2 consistent with the standard Westinghouse PWR technical specifications resulting in no significant safety considerations. Regarding item (3) we reject the proposed change since the staff had previously discussed this matter with the licensee in response to its January 12, 1981 letter and find no justification for the change. Further action on item (4) has been deferred pending additional justification by the licensee. The staff in further refining specific areas of the specifications have made revisions in addition to those proposed by the licensee. These matters were discussed with the licensee and revisions were mutually agreed upon.

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Environmental Consideration

We have determined that the amendment does 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 amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability of consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

DATE: APR 2 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-369

DUKE POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO LICENSE NO. NPF-9

FOR FUEL LOADING AND ZERO POWER TESTING

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 2 to License No. NPF-9 for Fuel Loading and Zero Power Testing for the McGuire Nuclear Station, Unit 1. This amendment was issued to Duke Power (licensee) to correct several typographical errors, revise reactor coolant system testing requirements and make minor changes to reflect revisions to the Westinghouse Standard Technical Specifications. The McGuire Nuclear Station, Unit 1 is located near Charlotte, North Carolina in Mecklenburg County. This amendment is effective as of its date of issuance.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's requirements. The Commission has made appropriate findings as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards considerations.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment, dated February 23, 1981; (2) Amendment No. 2 to License NPF-9 for Fuel Loading and Zero Power Testing; and (3) the Commission's related Safety Evaluation. All of these documents are available for public inspection at the Commission's Public Document Room, located at 1717 H Street, N. W., Washington, D. C. 20555 and at the Atkins Library, University of North Carolina Charlotte (UNCC Station) North Carolina 28223 or may be requested by writing to U. S. Nuclear Regulatory Commission, Washington, D. C. 20555 Attention: Director, Technical Information and Document Control.

Dated at Bethesda, Maryland, this 2nd day of April 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

No OELD Concurrence 4/2/81

See Note SFS to Birkel changes made RB 4/2/81

OFFICE	DL/LB#1	DL/LB#1	OELD	DL/LB#1			
SURNAME	MRushbrook	RBirkel	KETCHEN	BJYoungblood			
DATE	4/1/81	4/1/81	4/1/81	4/2/81			