Docket No. 50-369
and 50-370

May 13, 1986

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.57 to Facility Operating License NPF-9 and Amendment No.38 to Facility Operating License NPF-17 - McGuire Nuclear Station, Units 1 and 2

The Nuclear Regulatory Commission has issued the enclosed Amendment No.57 to Facility Operating License NPF-9 and Amendment No.38 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 May 9, 1985 and supplemented October 2 and 14, December 17 and 23, 1985, January 14, March 17, and April 8, 1986.

The amendments change the Technical Specifications to permit operation up to full power with the Upper Head Injection Accumulator System (UHIS) functionally disabled or physically removed. The amendments are effective as of their date of issuance.

A copy of the related safety evaluation supporting Amendment No.57 to Facility Operating License NPF-9 and Amendment No.38 to Facility Operating License NPF-17 is enclosed.

Also enclosed is a copy of the related Federal Register notice.

Sincerely,

/S/

Darl Hood, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Enclosures:
1. Amendment No.57 to NPF-9
2. Amendment No.38 to NPF-17
3. Safety Evaluation
4. F. R. Notice

cc w/enclosures: See next page

KWJ for
PWR#4/DPWR-A
DHood/mac
05/ 2/86
5/13/86 DSH

PWR#4/DPWR-A
MDuncan
05/ 2/86

PWOL
PWR#4/DPWR-A
for BJYoungblood
05/02/86

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PDR ADOCK 05000369
P PDR

Mr. H. B. Tucker
Duke Power Company

McGuire Nuclear Station

cc:

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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. 57
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 (the licensee) dated May 9, 1985, and supplemented October 2 and 14, December 17 and 23, 1985, January 14, March 17, and April 8, 1986, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the 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.57, are hereby incorporated into the 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

/S/

B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Technical Specification
Changes

Date of Issuance: May 13, 1986

PWR#4/DPWR-A	PWR#4/DPWR-A	OELD	PWR#4/DPWR-A
MDuncan:mac	DHood	J. GRAY	for BJYoungblood
04/24/86	05/2/86	04/8/86	05/02/86
	5/13/86		
	DSH		

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. 38
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 (the licensee) dated May 9, 1985, and supplemented October 2 and 14, December 17 and 23, 1985, January 14, March 17, and April 8, 1986, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the 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. 38 are hereby incorporated into the 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

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B. J. Youngblood, Director
PWR Project Directorate #4
Division of PWR Licensing-A

Attachment:
Technical Specification
Changes

Date of Issuance: May 13, 1986

PWR#4/DPWR-A
MDuncan:mac
04/24/86

KNJ
PWR#4/DPWR-A
DHood
05/2/86
5/12/86

OELD
J. GRAY
05/6/86

PWR
PWR#4/DPWR-A
for BJYoungblood
05/02/86

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 38

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 vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Amended Page	Overleaf Page
3/4 4-21	3/4 4-22
3/4 5-1	
3/4 5-2a	
3/4 5-2b	
3/4 5-3	
3/4 5-3a	
3/4 5-3b	
3/4 5-4	
3/4 6-3	
3/4 6-5	3/4 6-6
3/4 6-26	3/4 6-25
3/4 8-21	
3/4 8-22	
3/4 8-60	
3/4 8-61	3/4 8-62
B 3/4 5-1	
B 3/4 5-1a	
B 3/4 5-2	

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER		FUNCTION
MC-1562-2.0		
NI60		Accumulator Discharge
NI71		Accumulator Discharge
NI59		Accumulator Discharge
NI70		Accumulator Discharge
MC-1562-2.1		
NI82		Accumulator Discharge
NI94		Accumulator Discharge
NI81		Accumulator Discharge
NI93		Accumulator Discharge
MC-1562-3.0		
NI134		Safety Injection (Hot Leg)
NI159		Safety Injection (Hot Leg)
NI156		Safety Injection (Hot Leg)
NI128		Safety Injection (Hot Leg)
NI124		Safety Injection (Hot Leg)
NI160		Safety Injection (Hot Leg)
NI157		Safety Injection (Hot Leg)
NI126		Safety Injection (Hot Leg)
NI129		Safety Injection (Hot Leg)
NI125		Safety Injection (Hot Leg)
MC-1562-3.1		
NI165		Safety Injection/Residual Heat Removal (Cold Leg)
NI167		Safety Injection/Residual Heat Removal (Cold Leg)
NI169		Safety Injection/Residual Heat Removal (Cold Leg)
NI171		Safety Injection/Residual Heat Removal (Cold Leg)
NI175		Safety Injection/Residual Heat Removal (Cold Leg)
NI176		Safety Injection/Residual Heat Removal (Cold Leg)
NI180		Safety Injection/Residual Heat Removal (Cold Leg)
NI181		Safety Injection/Residual Heat Removal (Cold Leg)
MC-1562-4.0		
NI250	Note 1	Upper Head Injection
NI251	Note 1	Upper Head Injection
NI252	Note 1	Upper Head Injection
NI253	Note 1	Upper Head Injection
NI249	Note 1	Upper Head Injection
NI248	Note 1	Upper Head Injection
MC-1561-1.0		
NDIB*		Residual Heat Removal
ND2A*		Residual Heat Removal

*Testing per Specification 4.4.6.2.2d not applicable due to positive indication of valve position in Control Room.

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

a. With any one or more chemistry parameter in excess of its Steady State Limit but within its Transient Limit, restore the parameter to within its Steady-State Limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With any one or more chemistry parameter in excess of its Transient Limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady-State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 psig, if applicable, and 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 prior to increasing the pressurizer pressure above 500 psig or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

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 6870 and 7342 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 585 and 639 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. (UHI physically disconnected or isolation valves closed. Cold Leg Accumulators and discharge paths suitably modified.)

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 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 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.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.2 Each cold leg injection accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 8022 and 8256 gallons,

c. A boron concentration of between 1900 and 2100 ppm,

d. A nitrogen cover-pressure of between 430 and 484 psig, and

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*. (UHI operability required).

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.2.1 Each cold leg injection accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying 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.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution;

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that power to the isolation valve operator is disconnected by removal of the breaker from the circuit.

4.5.1.2.2 Each cold leg injection accumulator water level and pressure channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of an ANALOG CHANNEL OPERATIONAL TEST, and

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

3/4.5.1.3.4 UPPER HEAD INJECTION

[Deleted upon physical removal of the system]

LIMITING CONDITION FOR OPERATION

3.5.1.3 Each Upper Head Injection Accumulator shall be OPERABLE with:

a. The isolation valves open,

b. The water-filled accumulator containing a minimum of 1850 cubic feet of borated water having a concentration of between 1900 and 2100 ppm of boron, and

c. The nitrogen bearing accumulator pressurized to between 1206 and 1264 psig.

APPLICABILITY: MODES 1, 2, and 3*. (UHI operability required. Cold leg accumulators and discharge paths not suitably modified.)

ACTION:

a. With the Upper Head Injection Accumulator System inoperable, except as a result of a closed isolation valve(s), restore the Upper Head Injection Accumulator System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1900 psig within the following 6 hours.

b. With the Upper Head Injection Accumulator System inoperable due to the isolation valve(s) being closed, either immediately open the isolation valve(s) or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1900 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.3 Each Upper Head Injection Accumulator System shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the contained borated water volume and nitrogen pressure in the accumulators, and

2) Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1900 psig.

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator;

c. At least once per 18 months by:

1) Verifying that each accumulator isolation valve closes automatically when an actual or simulated water level signal of 76.25 ± 3.3 inches above the bottom inside edge of the water-filled accumulator exists. If actual water level is used, then the accumulator shall be at atmospheric pressure.

2) Verifying that the total dissolved nitrogen and air in the water-filled accumulator is less than 80 scf per 1800 cubic feet of water (equivalent to 5×10^{-5} pounds nitrogen per pounds water).

d. At least once per 5 years by replacing the membrane installed between the water-filled and nitrogen bearing accumulators and verifying that the removed membrane burst at a differential pressure of 40 ± 10 psi.

UPPER HEAD INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1.4 Each Upper Head Injection Accumulator System shall be isolated with the isolation valves closed.

APPLICABILITY: MODES 1, 2, and 3. (With Cold Leg Accumulators and discharge paths suitably modified.)

ACTION:

a. With the Upper Head Injection Accumulator System isolation valve(s) open, either immediately close the isolation valve(s) or be in HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1900 psig within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.4 Each Upper Head Injection Accumulator System shall be demonstrated isolated at least once per 12 hours by:

1) Verifying that each accumulator isolation valve is closed.

TABLE 3.6-1

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M216	Pressurizer Relief Tank Makeup	Auxiliary Building	Type C
M212	Nitrogen to Pressurizer Relief Tank	Auxiliary Building	Type C
M259	Reactor Makeup Water Tank to NV System	Auxiliary Building	Type C
M373	Ice Condenser Glycol In	Auxiliary Building	Type C
M372	Ice Condenser Glycol Out	Auxiliary Building	Type C
M330	Nitrogen to Accumulators	Auxiliary Building	Type C
M321	Safety Injection Test Line	Auxiliary Building	Type C
M348	Upper Head Injection Test Line	Auxiliary Building	Type C Note 1
M374	Containment Floor Sump Incore Instrument Sump Discharge	Auxiliary Building	Type C
M360	Reactor Coolant Drain Tank Gas Space to Waste Gas System	Auxiliary Building	Type C
M375	Reactor Coolant Drain Tank Heat Exchanger Discharge	Auxiliary Building	Type C
M356	Equipment Decontamination	Auxiliary Building	Type C
M235	Pressurizer Sample	Auxiliary Building	Type C
M309	Reactor Coolant Hot Leg Sample	Auxiliary Building	Type C
M322	Component Cooling to Component Cooling Drain Tank	Auxiliary Building	Type C

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

TABLE 3.6-1 (Continued)

SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS

PENETRATION NUMBER	SERVICE	RELEASE LOCATION	TEST TYPE
M307	Nuclear Service Water to Reactor Coolant Pump	Auxiliary Building	Type C
M315	Nuclear Service Water From Reactor Coolant Pump	Auxiliary Building	Type C
M213	Incore Instrumentation Room Purge In	Auxiliary Building	Type C
M138	Incore Instrumentation Room Purge Out	Auxiliary Building	Type C
M367	Upper Compartment Purge Inlet	Auxiliary Building	Type C
M454	Upper Compartment Purge Inlet	Auxiliary Building	Type C
M357	Lower Compartment Purge Inlet	Auxiliary Building	Type C
M456	Lower Compartment Purge Inlet	Auxiliary Building	Type C
M368	Containment Purge Exhaust	Auxiliary Building	Type C
M455	Containment Purge Exhaust	Auxiliary Building	Type C
M119	Containment Purge Exhaust	Auxiliary Building	Type C
M337	Demineralized Water	Auxiliary Building	Type C
M240	Containment Ventilation Cooling Water In	Turbine Building	Type C
M385	Containment Ventilation Cooling Water In	Turbine Building	Type C

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
1. Phase "A" Isolation (continued)		
CF-153A	Auxiliary Nozzle Temper SG B	≤10
CF-155B	Auxiliary Nozzle Temper SG C	≤10
CF-157B	Auxiliary Nozzle Temper SG D	≤10
KC-305B#	Excess Letdown Hx Supply Pent. Isolation Outside	≤30
KC-315B#	Excess Letdown Hx Ret. Hdr. Pent. Isolation Outside	≤30
KC-320A	NCDT Hx Supply Hdr. Pent. Isolation Outside	≤15
KC-332B	NCDT Hx Supply Hdr. Pent. Isolation Inside	≤15
KC-333A	NCDT Hx Return Hdr. Pent. Isolation Outside	≤15
KC-429B	RB Drain Header Inside Containment Isolation	≤15
KC-430A	RB Drain Header Outside Containment Isolation	≤15
NB-260B	Reactor Makeup Water Tank to NV System	≤15
NC-53B	Nitrogen to Pressurizer Relief Tank Containment Isolation Outside	≤10
NC-54A	Nitrogen to Pressurizer Relief Tank Containment Isolation Inside	≤10
NC-56B	PRT Makeup	≤10
NC-195B	NC Pump Motor Oil Containment Isolation Outside	≤15
NC-196A	NC Pump Motor Oil Containment Isolation Inside	≤15
NF-228A	Air Handling Units Glycol Supply Containment Isolation Outside	≤15
NF-233A	Air Handling Units Glycol Supply Containment Isolation Inside	≤15
NF-234A	Air Handling Units Glycol Supply Containment Isolation Outside	≤15
NI-47A	Accumulator Nitrogen Supply Outside Containment Isolation	≤15
NI-95A	Test HDR Inside Containment Isolation	≤10

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SEC)
1. Phase "A" Isolation (continued)			
NI-96B		Test HDR Outside Containment Isolation	≤10
NI-120B		Safety Injection Pump to Accumulator Fill Line Isolation	≤10
NI-122B		Hot Leg Injection Check NI124, NI128 Test Isolation	≤10
NI-255B	Note 1	UHI Check Valve Test Line Isolation	≤10
NI-258A		UHI Check Valve Test Line Isolation	≤10
NI-264B		UHI Check Valve Test Line Isolation	≤10
NI-266A		UHI Check Valve Test Line Isolation	≤10
NI-267A		UHI Check Valve Test Line Isolation	≤10
NM-3A		Pressurizer Liquid Sample Line Inside Containment Isolation	≤15
NM-6A		Pressurizer Steam Sample Line Inside Containment Isolation	≤15
NM-7B		Pressurizer Sample Header Outside Containment Isolation	≤15
NM-22A		NC Hot Leg #1 Sample Line Inside Containment Isolation	≤15
NM-25A		NC Hot Leg #4 Sample Line Inside Containment Isolation	≤15
NM-26B		NC Hot Legs Sample Hdr. Outside Containment Isolation	≤15
NM-72B		NI Accumulator A Sample Line Inside Containment Isolation	≤15
NM-75B		NI Accumulator B Sample Line Inside Containment Isolation	≤15
NM-78B		NI Accumulator C Sample Line Inside Containment Isolation	≤15
NM-81B		NI Accumulator D Sample Line Inside Containment Isolation	≤15
NM-82A		NI Accumulator Sample Hdr. Outside Containment Isolation	≤15
NM-187A#		SG A Upper Shell Sample Containment Isolation Inside	≤15
NM-190A#		SG A Blowdown Line Sample Containment Isolation Inside	≤15
NM-191B#		SG A Sample Hdr. Containment Isolation Outside	≤15
NM-197B#		SG B Upper Shell Sample Containment Isolation Inside	≤15
NM-200B#		SG B Blowdown Line Sample Containment Isolation Inside	≤15
NM-201A#		SG B Sample Hdr. Containment Isolation Inside	≤15
NM-207A#		SG C Upper Shell Sample Containment Isolation Inside	≤15
NM-210A#		SG C Blowdown Line Sample Containment Isolation Inside	≤15
NM-211B#		SG C Sample Hdr. Containment Isolation Outside	≤15
NM-217B#		SG D Upper Shell Sample Containment Isolation Inside	≤15
NM-220B#		SG D Blowdown Line Sample Containment Isolation Inside	≤15

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

Amendment No. 57 (Unit 1)
Amendment No. 38 (Unit 2)

TABLE 3.8-1a (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2. 600 VAC-MCC (Continued)			
1EMXA-2 2B			
Primary Bkr	20	45 @ 60A	N2 to Prt Cont Isol Inside
Backup Fuse	20	N.A.	Vlv 1NC54A
1EMXA-2 2C			
Primary Bkr	20	45 @ 60A	RCP Mtg Brg Oil Fill Isol
Backup Fuse	20	N.A.	Vlv 1NC196A
1EMXA-2 3A			
Primary Bkr	30	45 @ 90A	Accumulator 1A Disch Isol
Backup Fuse	30	N.A.	Vlv 1NI54A
1EMXA-2 3B			
Primary Bkr	30	45 @ 90A	Accumulator 1C Disch Isol
Backup Fuse	30	N.A.	Vlv 1NI76A
1EMXA-2 3C			
Primary Bkr	20	45 @ 60A	Test Hdr Inside Cont Isol
Backup Fuse	20	N.A.	Vlv 1NI95A
1EMXA-2 4A			
Primary Bkr } Note 1	20	45 @ 60A	UHI Check Vlv Test Line Isol
Backup Fuse } Note 1	20	N.A.	Vlv 1NI266A

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

TABLE 3.8-1a (Continued)

UNIT 1 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION		TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2. 600 VAC-MCC (Continued)				
1EMXA-2 4B				
Primary Bkr	Note 1	20	45 @ 60A	UHI Check Vlv Test Line Isol
Backup Fuse		20	N.A.	Vlv 1NI267A
1EMXA-2 4C				
Primary Bkr		20	45 @ 60A	Accum 1A Vent to 1NC34 for
Backup Fuse		20	N.A.	Blkout Vlv 1NI430A
1EMXA-5 1B				
Primary Bkr		20	45 @ 60A	Pzr Steam Sample Line Inside
Backup Fuse		20	N.A.	Cont Isol Vlv 1NM3A
1EMXA-5 2B				
Primary Bkr		20	45 @ 60A	Pzr Steam Sample Line Inside
Backup Fuse		20	N.A.	Cont Isol Vlv 1NM6A
1EMXA-5 3B				
Primary Bkr		20	45 @ 60A	NC Hotleg 1A Sample Line Cont
Backup Fuse		20	N.A.	Isol Vlv 1NM22A
1EMXA-5 2D				
Primary Bkr		20	45 @ 60A	NC Hotleg 1D Sample Line Cont
Backup Fuse		20	N.A.	Isol Vlv 1NM25A
1EMXA-2 7A				
Primary Bkr		20	45 @ 60A	SG 1A Upper Shell Sample Cont
Backup Fuse		20	N.A.	Isol Vlv 1NM187A

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

TABLE 3.8-1b (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2. 600 VAC-MCC (Continued)			
2EMXA-2 2B			
Primary Bkr	20	45 @ 60A	N2 to Prt Cont Isol Inside
Backup Fuse	20	N.A.	Vlv 2NC54A
2EMXA-2 2C			
Primary Bkr	20	45 @ 60A	RCP Mtg Brg Oil Fill Isol
Backup Fuse	20	N.A.	Vlv 2NC196A
2EMXA-2 3A			
Primary Bkr	30	45 @ 90A	Accumulator 2A Disch Isol
Backup Fuse	30	N.A.	Vlv 2NI54A
2EMXA-2 3B			
Primary Bkr	30	45 @ 90A	Accumulator 2C Disch Isol
Backup Fuse	30	N.A.	Vlv 2NI76A
2EMXA-2 3C			
Primary Bkr	20	45 @ 60A	Test Hdr Inside Cont Isol
Backup Fuse	20	N.A.	Vlv 2NI95A
2EMXA-2 4A			
Primary Bkr } Note 1	20	45 @ 60A	UHI Check Vlv Test Line Isol
Backup Fuse } Note 1	20	N.A.	Vlv 2NI266A

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

TABLE 3.8-1b (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION		TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEM POWERED
2. 600 VAC-MCC (Continued)				
2EMXA-2 4B				
Primary Bkr	Note 1	20	45 @ 60A	UHI Check Vlv Test Line Isol
Backup Fuse		20	N.A.	Vlv 2NI267A
2EMXA-2 4C				
Primary Bkr		20	45 @ 60A	Accum 2A Vent to 2NC34 for
Backup Fuse		20	N.A.	Blkout Vlv 2NI430A
2EMXA-5 1B				
Primary Bkr		20	45 @ 60A	Pzr Steam Sample Line Inside
Backup Fuse		20	N.A.	Cont Isol Vlv 2NM3A
2EMXA-5 2C				
Primary Bkr		20	45 @ 60A	Pzr Steam Sample Line Inside
Backup Fuse		20	N.A.	Cont Isol Vlv 2NM6A
2EMXA-5 3B				
Primary Bkr		20	45 @ 60A	NC Hotleg 2A Sample Line Cont
Backup Fuse		20	N.A.	Isol Vlv 2NM22A
2EMXA-5 2D				
Primary Bkr		20	45 @ 60A	NC Hotleg 2D Sample Line Cont
Backup Fuse		20	N.A.	Isol Vlv 2NM25A
2EMXA-2 7A				
Primary Bkr		20	45 @ 60A	SG 2A Upper Shell Sample Cont
Backup Fuse		20	N.A.	Isol Valve 2NM187A

Note 1: Upon the deactivation of the UHI System by removal of related components and piping and modifications to the Cold Leg Accumulators, this specification is no longer applicable.

TABLE 3.8-1b (Continued)

UNIT 2 CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER & LOCATION	TRIP SETPOINT OR CONT. RATING (AMPERES)	RESPONSE TIME (SECONDS)	SYSTEMS POWERED
2. 600 VAC-MCC (Continued)			
2EMXA-2 7B			
Primary Bkr	20	45 @ 60A	SG 2A Blowdown Line Sample
Backup Fuse	20	N.A.	Cont Isol Vlv 2NM190A
2EMXA-2 7C			
Primary Bkr	20	45 @ 60A	SG 2C Upper Shell Sample
Backup Fuse	20	N.A.	Cont Isol Vlv 2NM207A
2EMXA-2 8A			
Primary Bkr	20	45 @ 60A	SG 2C Blowdown Line Sample
Backup Fuse	20	N.A.	Cont Isol Vlv 2NM210A
2EMXA-4 1B			
Primary Bkr	20	45 @ 60A	NC Pump Seal Return Cont
Backup Fuse	20	N.A.	Isol Vlv 2NV94AC
2EMXA-3 3A			
Primary Bkr	20	45 @ 60A	H2 Purge Exhaust Cont
Backup Fuse	20	N.A.	Vessel Isol Vlv 2VE5A
2EMXA-3 3B			
Primary Bkr	20	45 @ 60A	Cont H2 Purge Blower
Backup Fuse	20	N.A.	Inlet Valve 2VE8A
2EMXA-3 3C			
Primary Bkr	20	45 @ 60A	H2 Purge Inlet Cont
Backup Fuse	20	N.A.	Vessel Isol Vlv 2VE10A
2EMXA-4 2C			
Primary Bkr	20	45 @ 60A	Standby Makeup Pump Inlet
Backup Fuse	20	N.A.	Isol Valve 2NV842AC

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) Cold Leg Accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The original licensing bases of McGuire assumes both the UHI system and the Cold Leg Accumulators function to mitigate postulated accidents. Subsequent analyses, documented in "McGuire Nuclear Station, Safety Analysis for UHI Elimination" dated September 1985, and docketed by Duke letter dated October 2, 1985, support the determination that UHI is no longer required provided the Cold Leg Accumulator volume is adjusted to be consistent with that assumed in the Safety Analysis.

Accordingly, Specification 3/4.5.1.1 on pages 3/4 5-1, -2 is provided as the planned final condition of the cold leg injection system with water volume and nitrogen cover pressure as assumed in the Safety Analysis. It is applicable for the conditions of UHI being either physically removed or isolated. It has a prerequisite that the cold leg accumulators and discharge flow paths are consistent with assumptions used in the Safety Analysis. The actions required to align the plant configuration with the safety analyses assumptions are:

1. Replace flow restricting orifices in the Cold Leg Accumulators discharge piping to ensure actual accumulator discharge line resistances are within tolerance bands of ECCS analysis.

ACCUMULATORS (Continued)

2. Reduce the nominal water level in the Cold Leg Accumulators and modify instrumentation, alarm functions, and procedures accordingly.

3. Increase the nominal gas cover pressure in the Cold Leg Accumulators and modify instrumentation, alarm functions, and procedures accordingly.

4. Ensure UHI isolation valves remain closed during operation.

Items 1, 2, 3 constitute modifications required by the phrase "Cold Leg Accumulator and discharge paths suitably modified" contained in the technical specifications. Item 4 is a requirement that will be implemented by administrative controls and is reflected in Technical Specification 3/4 5.1.4.

Specification 3/4.5.1.2 on page 3/4 5-2a, -2b is the Technical Specification requirements for the Cold Leg Injection with UHI system operable. As this is expected to be an interim condition, the pages have been renumbered with suffixes a, b. With respect to Technical Specifications for the UHI system, page 3/4 5-3 is provided as the planned final condition of the plant, that being with the UHI system deleted. New page 3/4 5-3c is provided to cover the condition following modification of the cold leg accumulators and discharge paths. This new requirement is that the isolation valves be closed and remain closed. Surveillance once per shift (12 hours) is provided. New page 3/4 5-3a duplicates the present Technical Specification for operability of the UHI system and is now Specification 3.5.1.3. As this is expected to be an interim condition the page has been renumbered to 3/4 5-3a. Page 3/4 5-3b provides the continuation of the surveillance requirements of Specification 4.5.1.2 when UHI system operability is required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump and one Safety Injection pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4

[Deleted]

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 10.5 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 57TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO.38 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

McGUIRE NUCLEAR STATION, UNITS 1 AND 2

INTRODUCTION

By letter dated May 9, 1985 and revised or supplemented October 2 and 14, December 17 and 23, 1985, January 14, March 17 and April 8, 1986, Duke Power Company (the licensee) requested changes to the Technical Specifications to provide for operation up to full power with the Upper Head Injection (UHI) system in one of three conditions: (1) fully operable in accordance with existing Technical Specifications, (2) physically installed but functionally disabled by the closing of both isolation valves in each of the two injection lines, and (3) deleted by physical removal of major portions of the system such as injection piping, valves, support/restraints and instrumentation. Other changes associated with UHI system isolation or removal were also requested. These include deletion of technical specifications requiring UHI system maintenance, surveillance, and leakage verification and modifications of Technical Specifications to reflect deletion of UHI related containment penetrations and associated conductor overcurrent protection devices, containment isolation valves, and system piping snubbers.

The licensee's letter also requested Technical Specification changes for the ECCS cold leg injection accumulator which would be implemented prior to and during the startup with the UHI system isolated or removed. These changes increase the operable range of the nitrogen gas cover-pressure and decrease the operable range of the water volume from 8022 and 8256 gallons to 6870 and 7342 gallons. (The changes to the ECCS cold leg injection accumulator will also be accompanied by appropriate modifications to instrumentation, alarm functions and procedures, and by resizing the flow restricting orifices in the discharge piping; however, these accompanying changes do not involve a change to the Technical Specifications).

EVALUATION

The present plant design for McGuire, Units 1 and 2 incorporates an ice condenser containment and Upper Head Injection System. The ice condenser containment was introduced as a less costly alternative to the large dry containment. The ice beds were designed to efficiently condense the steam from the postulated design basis large break loss-of-coolant accident (LOCA), limiting the pressure excursion in the containment. The low containment pressure contributed to increased steam binding in the primary system, thereby delaying reflood of the core and leading to higher calculated peak clad temperatures.

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Upper Head Injection (UHI) was introduced in some Westinghouse plants to enhance core cooling during blowdown and to generally compensate for core cooling problems associated with low containment pressure. Five plants containing UHI have been licensed and three more are nearing operation. The UHI system may be visualized as a pressurized accumulator tank containing at least 1850 cubic feet of borated water which is connected through a system of shut-off and check valves to the upper head of the reactor vessel. A nitrogen gas system keeps the UHI system pressurized to over 1200 psi. When the primary system is above 1900 psi, the UHI is isolated only by check valves. UHI plant upper head internals are designed to assure good mixing of the injected flow with the steam from the hot legs during a cold leg LOCA.

Experience has demonstrated that the UHI system adds to the complexity of plant operation, requires additional maintenance and generally reduces plant availability. Licensee event reports have identified numerous problems with the UHI systems. Duke Power Company began discussions with the NRC early in 1985 while evaluating the possibility of eliminating the UHI systems from the McGuire Station, Units 1 and 2. After a series of interchanges with the staff and the ACRS, it was generally agreed that there were adequate incentives to consider removal.

As identified above, the licensee submitted a number of documents to support the proposed removal of the UHI system during successive refueling outages.

This SER presents the results of the staff review of the licensee proposal to remove the UHI system and modify the associated Technical Specifications. The staff review included the revised Chapter 15 FSAR analyses and supporting discussion of plant characteristics as they would be affected by the removal of the UHI system.

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. There are three distinct transients analyzed in each phase: (1) the thermal-hydraulic transient in the reactor coolant system (RCS), (2) the pressure and temperature transient within the containment, (3) and the fuel and cladding temperature transient of the hottest fuel rod in the core. Based on these considerations, a system of interrelated computer codes has been developed for the analysis of the LOCA.

The SATAN-IV computer code analyzes the thermal-hydraulic transient in the RCS during blowdown and refill. The WREFLOOD and BART computer codes are used to calculate the thermal-hydraulic transient during the reflood phase of the accident. The BART computer code is used to calculate the fluid and heat transfer conditions in the core during reflood. The LOTIC computer code is used to calculate the containment pressure transient during all three phases of the LOCA analysis. Similarly, the LOCTA-IV computer code is used to compute the thermal transient of the hottest fuel rod during the three phases. References 1, 2, 3 and 4 describe the major phenomena modeled, the interfaces among the computer codes, and the features of the codes which assure compliance with NRC Acceptance Criteria. The staff has confirmed that acceptable evaluation models were used for the large break LOCA analysis.

The large break LOCA (double ended cold leg break at reactor coolant pump discharge) analysis results are listed in Table 1 along with a column describing the 10 CFR 50.46 limits. The break spectrum was developed by varying the break discharge coefficient (C_D) assuming the worst single failure. The limiting case (C_D = 0.6) resulted in a peak clad temperature (PCT) of 1896°F. When the limiting break analysis was repeated assuming full safety injection (SI) capability, including full containment spray, a PCT of 2132°F resulted. The assumption of full safety injection resulted in a lower containment pressure, leading to steam binding. This delayed reflood and resulted in the higher PCT. The full containment spray plus the large quantity of safety injection flow issuing from the vessel side of the break contributed to the condensation of additional steam and the lower calculated containment pressure. The full SI case with a C_D = 0.6 thus became the limiting case for the non-UHI configuration.

The small break LOCA reanalysis submitted by the licensee was performed with the NOTRUMP code as described in Reference 5 and approved by the NRC in August 1985. Using the WFLASH code, the previous (1984 update of the McGuire FSAR) small break LOCA analysis identified the limiting small break as a break equivalent to a 6 inch diameter hole and reported a PCT of 1499°F. The reanalysis using the more realistic NOTRUMP code, identified the limiting small break as a break equivalent to a 3 inch diameter hole and PCT of 1488°F. The differences between these analyses are attributed to modeling changes and the use of non-equilibrium thermodynamics in NOTRUMP.

An acceptable version of the LOFTRAN code (reference 6) was used for transient analysis. Only two plant transients were reanalyzed because these were the only transients which were predicted to depressurize the primary system sufficiently to initiate the upper head injection. One was the "Inadvertent Opening of a Steam Generator Relief or Safety Valve" and the other was "Steam System Piping Failure". The THINC code was used to determine if departure from nucleate boiling (DNB) occurred. The steam system rupture was the more limiting transient. The DNB analysis confirmed that DNB ratios were always greater than the fuel design limit value of 1.17.

In summary, we have confirmed that the safety analyses were done with approved codes. Plant and core characteristics without UHI were demonstrated to be within regulatory limits.

The NRC staff was aided by consultants from the Sandia National Laboratory (SNL) in the review of the proposed removal of UHI from the McGuire Station. SNL was asked to use the TRAC-PFI/MOD1 code to study the characteristics of a generic Westinghouse 4-loop reactor with ice condenser, with and without UHI.

The UHI internals were modeled for both cases. The water in the upper head provides some cooling benefit whether or not it is augmented by flow from the UHI system. For the non-UHI case the cold leg accumulators were modeled similarly to other four loop Westinghouse plants (accumulator pressure up, water volume decreased slightly, and accumulator line resistance decreased) so that cold leg injection would be more effective.

In our two cases the first part of blowdown was identical. The blowdown PCT occurred at about 2½ seconds. As the UHI didn't come on until around 3 seconds, UHI had no effect on the blowdown PCT which was slightly less than 1400°F. At the end of blowdown the cladding temperature was about 100°F cooler with UHI than without. This is of interest primarily to show when some cooling benefit from UHI occurs.

In our UHI case there is no reheat of cladding after blowdown. It stays at roughly 1000°F until the hot rod is quenched. In our non-UHI case, only the hottest rods reheated to around 1400°F before they turned around. In our TRAC analysis, the cladding temperatures reached during blowdown and during reheat were well below the 2200°F limit of 10 CFR 50.46.

The TRAC analyses have been characterized as less conservative than an Appendix K analysis but somewhat more conservative than a "Best Estimate" or "most likely case". It has been pointed out by the ACRS and others that if a more realistic F_Q (lower than 2.17) had been selected for the TRAC input there likely would have been little or no clad reheat after blowdown in the non-UHI case. We believe that the analysis as done confirms that even if a few of the hottest rods do experience some reheat, there is ample margin relative to the 10 CFR 50.46 limits.

SNL was also requested to investigate the effects of nitrogen cover gas entering the primary system if the UHI isolation valves did not shut off at the proper tank low level mark. SNL concluded that a postulated failure to shut off the UHI at the low tank level was indicated to be slightly beneficial to core cooling. Continued UHI water flow into the core (nearly twice as much) provided additional cooling before the nitrogen cover gas would enter the upper head. When the gas did enter the upper head, there was significant venting to the cold legs and out the break and therefore very little nitrogen actually passed through the core.

We have reviewed the licensee's request to remove the UHI sytem and change the associated Technical Specifications. As stated previously, we have concluded that the McGuire Nuclear Station, Units 1 and 2, meets the requirements of 10 CFR 50.46 for large and small break LOCAs without UHI. In addition, potentially affected transients were evaluated without UHI and found not to exceed fuel design limits. Therefore, the licensee's request to eliminate the UHI system for the McGuire Nuclear Station, Units 1 and 2 is acceptable. Similarly, licensee's request to operate with the UHI functionally disabled by isolation is also acceptable.

ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32, the Commission has determined that issuance of the amendments will have no significant impact on the environment (51 FR 13574).

CONCLUSION

Notices of opportunity for a prior hearing were published in the Federal Register on July 26, 1985 (50 FR 30548) and February 4, 1986 (51 FR 4449). No requests

for a hearing were received. The licensee's letters of March 17 and April 8, 1986, do not alter the requested changes and are addressed in the Commission's Environmental Assessment and Finding of No Significant Impact (51 FR 13574). 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.

REFERENCES

1. Bordelon, F. M., Massie, H. W. and Borden, T. A., "Westinghouse ECCS Evaluation Model-Summary", WCAP-8339, (Non-Proprietary), July 1974.

2. Westinghouse ECCS Evaluation Model, 1981 Version", WCAP-9220-P-A, Rev. 1 (Proprietary), WCAP-9221-A, Rev. 1 (Non-Proprietary), February, 1982.

3. Young, M., et al., "BART-1A: A Computer Code for the Best Estimate Analyzed Reflood Transients", WCAP-9561-P-A, 1984 (Westinghouse Proprietary).

4. Chiou, J. S., et al., "Models for PWR Reflood Calculations Using the BART Code", WCAP-10062.

5. Lee, H., Tauche, W. D., Schwarz, W. R., "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code", WCAP-10081-A, August 1985.

6. Burnett, T. W. T., et al., "LOFTRAN Code Description", WCAP-7907-P-A, April 1984.

Principal Contributors: Darl S. Hood, PWR#4
J. Watt

Dated: May 13, 1986

TABLE 1

LARGE BREAK LOCA RESULTS

WITHOUT UHI

	$C_{D_{DECLG}} = 0.8$	$C_{D_{DECLG}} = 0.6$	$C_{D_{DECLG}} = 0.4$	$C_{D_{DECLG}} = 0.6$ (MAX SI)	LIMITS 10 CFR 50.46
FUEL CLADDING DATA					
Peak Clad Temperature (°F)	1865	1895	1863	2132	2200
Peak Clad Temperature Location (ft)	6.75	6.75	6.75	6.50	
Local Zr/H_2O Reaction (max), (%)	2.53	2.12	2.16	5.05	17.00
Local Zr/H_2O Location (ft)	5.50	6.00	5.50	6.50	
Total Zr/H_2O Reaction (%)	0.3	0.3	0.3	0.3	1.0
Hot Rod Burst Time, (sec)	61.4	62.2	88.8	63.0	Coolable
Hot Rod Burst Location, (ft)	5.50	6.00	5.50	6.00	

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U. S. NUCLEAR REGULATORY COMMISSION

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 57 to Facility Operating License No. NPF-9, and Amendment No. 38 to Facility Operating License No. NPF-17, issued to Duke Power Company (the licensee), which revised the Technical Specifications for operation of the McGuire Nuclear Station, Units 1 and 2 (the facility) located in Mecklenburg County, North Carolina. The amendments are effective as of the dates of their issuance.

The amendments change the Technical Specifications to permit operation up to full power with the Upper Head Injection System functionally disabled or physically removed.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notices of Consideration of Issuance of Amendment and Opportunity for Prior Hearing in connection with this action were published in the FEDERAL REGISTER on July 26, 1985 (50 FR 30548) and on February 4, 1986 (51 FR 4449). No request for a hearing or petition for leave to intervene was filed following these notices.

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The Commission has prepared an Environmental Assessment and Finding of No Significant Impact (51 FR 13574) related to the action and has concluded that an environmental impact statement is not warranted because there will be no environmental impact attributable to the action beyond that which has been predicted and described in the Commission's Final Environmental Statement for the facility dated April 1976 and its addendum dated January 1981.

For further details with respect to the action see (1) the application for amendment dated May 9, 1985, as supplemented October 2 and 14, December 17 and 23, 1985, January 14, March 17, and April 8, 1986, (2) Amendment No.57 to License No. NPF-9 and Amendment No.38 to License No. NPF-17, (3) the Commission's related Safety Evaluation and Environmental Assessment, and (4) the Advisory Committee on Reactor Safeguards letter dated April 16, 1986. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., and at the Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28242. A copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of PWR Licensing-A.

Dated at Bethesda, Maryland this day of 1986.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Paul W. O'Connor, Acting Director
PWR Project Directorate #4
Division of PWR Licensing-A

*SEE PREVIOUS CONCURRENCE

PWR#4/DPWR-A	PWR#4/DPWR-A	OELD	PWR#4/DPWR-A
*MDuncan/mac	*DHood	*	*BJYoungblood
04/ /86	04/ /86	04/ /86	04/ /86

DATED: May 13, 1986

AMENDMENT NO. 57 TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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