October 27, 1994

Docket Nos. 50-369
and 50-370

Mr. T. C. McMeekin
Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, North Carolina 28078-8985

Distribution
Docket File
NRC/Local PDRs
PDII-3 Reading
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G.Hill(4) TWF 5-C-3
C.Grimes 11F23
ACRS(10) P-135
PA 17F2
OC/LFMB MNB4702
J.Johnson(A),RII

Dear Mr. McMeekin:

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M77359, M77360, M77429, AND M77430)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 150 to Facility Operating License NPF-9 and Amendment No. 132 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated November 21, 1991.

The amendments were submitted as a result of NRC recommendations pertaining to Generic Letter 90-06 for the power-operated relief valves and block valves and low-temperature overpressure protection systems.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

Victor Nerses, Project Manager
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 150 to NPF-9
2. Amendment No. 132 to NPF-17
3. Safety Evaluation

cc w/enclosures:
See next page

OFFICE	PDII-3/LA	PDII-3/PM	BC:SRXB	OGC	D:PDII-3
NAME	L. BERRY	V. NERSES	RCJONES ~~[illegible]~~		HBERKOW
DATE	10/24/94	11/4/94	10/13/94	10/12/94	10/27/94

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FILE NAME:G:\MCGUIRE\MCG77359.AMD

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

October 27, 1994

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and 50-370

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Vice President, McGuire Site
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See next page

Mr. T. C. McMeekin
Duke Power Company

McGuire Nuclear Station

cc:
A. V. Carr, Esquire
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242-0001

County Manager of Mecklenberg County
720 East Fourth Street
Charlotte, North Carolina 28202

Mr. R. O. Sharpe
Compliance
Duke Power Company
McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

J. Michael McGarry, III, Esquire
Winston and Strawn
1400 L Street, NW.
Washington, DC 20005

Senior Resident Inspector
c/o U. S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Mr. T. Richard Puryear
Nuclear Technical Services Manager
Westinghouse Electric Corporation
Carolinas District
2709 Water Ridge Parkway, Suite 430
Charlotte, North Carolina 28217

Dr. John M. Barry
Mecklenberg County
Department of Environmental Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Dayne H. Brown, Director
Department of Environmental, Health and Natural Resources
Division of Radiation Protection
P. O. Box 27687
Raleigh, North Carolina 27611-7687

Mr. Marvin Sinkule, Chief
Project Branch #3
U. S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Mr. G. A. Copp
Licensing - EC050
Duke Power Company
526 South Church Street
Charlotte, North Carolina 28242-0001

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, NW. Suite 2900
Atlanta, Georgia 30323

Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

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

DUKE POWER COMPANY

DOCKET NO. 50-369

McGUIRE NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 150
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 November 21, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 150 , 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses for

Herbert N. Berkow, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
  Changes

Date of Issuance: October 27, 1994

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 132
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 November 21, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-17 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 132 , 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Nerses for

Herbert N. Berkow, Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: October 27, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 150

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 132

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.

Remove Pages	Insert Pages
3/4 4-10	3/4 4-10 3/4 4-10a
3/4 4-37 3/4 4-38	3/4 4-37 3/4 4-38
B 3/4 4-3 B 3/4 4-3a	B 3/4 4-3 B 3/4 4-3a*

*no change, overflow page

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 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 because of excessive leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and maintain power to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With two PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore the PORVs to OPERABLE status or close the associated block valves and remove power from the block valves; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the block valves have been closed and power has been removed, restore at least once PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With three PORVs inoperable due to causes other than excessive leakage, within 1 hour either restore at least one PORV to OPERABLE status or close the associated block valves and remove power from the block valves and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. With one block valve inoperable, within 1 hour restore the block valve to OPERABLE status or place its associated PORV switch in the "close" position and remove power from its associated solenoid valve (do not enter action statement b for the resulting inoperable PORV); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

3/4.4.4 RELIEF VALVES (continued)

LIMITING CONDITION FOR OPERATION

f. With two block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement c for the resulting inoperable PORVs); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If the PORV switches have been placed in the "close" position, restore at least one block valve to OPERABLE status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

g. With three block valves inoperable, within 1 hour restore the block valves to OPERABLE status or place their associated PORV switches in the "close" position (do not enter action statement d for the resulting inoperable PORVs). Restore at least one block valve to OPERABLE status within the next hour; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

h. The provisions of Specification 3.0.4 are no applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 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 during MODE 3 or MODE 4 when the temperature of all RCS cold legs is greater than 300F with the block valve closed.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION a., b., c., or d. in Specification 3.4.4.

4.4.4.3 The emergency power supply for the PORVs shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive power from the normal (air) supply to the emergency (nitrogen) supply.

b. Isolating and venting the normal (air) supply, and

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

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 400 psig, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 4.5 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 300°F, MODE 5, and MODE 6 when the head is on the reactor vessel.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or complete depressurization and venting of the RCS through at least a 4.5 square inch vent(s) within the next 8 hours.

b. With one PORV inoperable in MODE 5, suspend all operations that could lead to water-solid RCS conditions. Restore the inoperable PORV to OPERABLE status within 72 hours or complete depressurization and venting of the RCS through at least a 4.5 square inch vent(s) within the next 24 hours.

c. With one PORV inoperable in MODE 6, restore the inoperable PORV to OPERABLE status within 24 hours or complete depressurization and venting of the RCS through at lest a 4.5 square inch vent(s) within the next 8 hours.

d. With both PORVs inoperable, complete depressurization and venting of the RCS through at lest a 4.5 square inch vent(s) within 8 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions: 1) Manual control of PORVs to control RCS pressure. This is a function that is used for the steam generator tube rupture accident coincident with a loss of all offsite power and for plant shutdown. 2) Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage. 3) Manual control of the block valve to unblock an isolated PORV to allow it to be used for manual control of RCS pressure and isolate a PORV with excessive leakage. 4) Automatic control of PORVs to control RCS pressure. This is a function that reduces challenges to the code safety valves for overpressurization events. 5) Manual control of a block valve to isolate a stuck-open PORV.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken. The B&W process (or method) equivalent to the inspection method described in Topical Report BAW-2045(P)-A will be used. Inservice inspection of steam generator sleeves is also required to ensure RCS integrity. Because the sleeves introduce changes in the wall thickness and diameter, they reduce the sensitivity of eddy current testing, therefore, special inspection methods must be used. A method is described in Topical Report BAW-2045(P)-A with supporting validation data that demonstrates the inspectability of the sleeve and underlying tube. As required by NRC for licensees authorized to use this repair process, McGuire commits to validate the adequacy of any system that is used for periodic inservice inspections of the sleeves, and will evaluate and, as deemed appropriate by Duke Power Company, implement testing methods as better methods are developed and validated for commercial use.

3/4.4.5 STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plantshutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Repair will be required for all tubes with imperfections exceeding the repair limit of 40% of the tube nominal wall thickness. Installed sleeves with imperfections exceeding 40% of the sleeve nominal wall thickness will be plugged. Defective steam generator tubes can be repaired by the installation of sleeves which span the area of degradation, and serve as a replacement pressure boundary for the degraded portion of the tube, allowing the tube to remain in service. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type degradation that has penetrated 20% of the original tube wall thickness. For tubes with degradation below the F* distance, and not degraded within the F* distance, repair is not required. If a tube is sleeved due to degradation in the F* distance, then any defects in the tube below the sleeve will remain in service without repair.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 150 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 132 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

1.0 INTRODUCTION

By letter dated November 21, 1991, Duke Power Company (the licensee) submitted a request for changes to the McGuire Nuclear Station, Units 1 and 2, Technical Specifications (TS). The requested changes would revise the TS for the power-operated relief valves (PORVs) and block valves, and low-temperature overpressure protection (LTOP) systems as a result of NRC recommendations pertaining to Generic Letter 90-06. The purpose of this safety evaluation is to address the licensee's response changing the TS for the PORVs and LTOP systems.

2.0 EVALUATION

The proposed TS changes were submitted in accordance with the guidance provided by the NRC in its resolution of Generic Issue 70, "PORV and Block Valve Reliability" and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors." The specific changes are discussed individually in the following sections.

Regarding Generic Issue 94, licensees were requested to verify whether certain administrative restrictions concerning the restart of inactive reactor coolant pumps and concerning the operability of high pressure safety injection pumps have been implemented. These restrictions were imposed as a result of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." The licensee stated that these administrative restrictions have been previously implemented at McGuire.

Limiting Condition for Operations (LCO) 3.4.4

- Action statement a. becomes applicable only in the case of PORV inoperability due to excessive leakage. While GL 90-06 only refers to seat leakage, the new action statement a. simply refers to leakage in general, so that both seat leakage and packing leakage would constitute PORV inoperability under this action statement. The phrase dealing with the removal of power from the block valve(s) is changed to require power to be maintained to the block valve(s). This ensures that the valve(s) remain operable and capable of being opened to allow the PORV(s) to be used to control primary system pressure. In addition, the action statement has been modified to terminate the forced shutdown requirement in hot shutdown rather than cold shutdown, since the LCO only applies to Modes 1, 2, and 3.

- Action statements b., c., and d. govern the cases of PORV inoperability due to causes other than excessive leakage for the cases of one, two, or three inoperable PORVs, respectively. Action statement b. allows continued plant operation with one inoperable PORV, provided its block valve is closed and power is removed from the block valve. The licensee's basis for this position is that following a steam generator tube rupture with loss of offsite power, only one PORV is required to depressurize the Reactor Coolant System. All three of McGuire's PORVs have nitrogen backup capability to cope with a loss of instrument air to the valves. Therefore, even with one PORV inoperable for an extended period of time, redundant capability exists to depressurize the Reactor Coolant System following an SGTR event.

- Action statements e., f., and g. govern block valve inoperability for the cases of one, two, or three inoperable block valves, respectively. These action statements establish remedial measures consistent with the function of the block valves. The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. If the block valve(s) cannot be restored to operable status within 1 hour, the remedial action is to place the associated PORV switch(es) in the "close" position to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. (The guidance contained in the generic letter states to place the PORV in manual control; however, McGuire's PORV control switches are labeled "open", "close", and "auto", so the proposed change is consistent with the McGuire design.) Since the proposed Technical Specifications (TS) do not require the plant to be shut down for the case of one inoperable PORV or one inoperable block valve, action statement e. includes the additional requirement to remove power from the associated PORV solenoid valve. This provides additional assurance that the PORV with the inoperable block valve will not be inadvertently opened at a later date, because if the block valve is not capable of being closed, an unisolable leak path would be created if the PORV were to fail open. The requirement to remove power from the PORV solenoid valves is not necessary for action statements f. and g. since these action statements possess time limitations for restoring the block valves to operability. The time allowed to restore the block valve(s) to operable status is based upon the remedial action time limits for inoperable PORVs per action statements b., c., and d. since the PORVs are not capable of mitigating an overpressure event when the PORV switches are in the "close" position. The same basis proposed for allowing continued plant operation with one inoperable PORV in action statement b. also applies to action statement e. Finally, action statements e., f., and g. include a provision in parentheses which precludes entering action statements b., c., and d. respectively, for inoperable PORVs once the PORV switch(es) have been placed in the "close" position. This is necessary because placing the PORV switch(es) in the "close" position renders the PORV(s) inoperable and the resulting requirements for inoperable PORV(s) in action statements b., c., and d. (i.e., close the block valve(s) and remove power) are inappropriate since the block valve(s) are already inoperable under action statements e., f., and g. These changes are acceptable.

- Old action statement c. becomes new action statement h.

Surveillance Requirement 4.4.4.1

- The generic letter recommends that PORVs be stroked during Mode 3 or Mode 4 in order to accurately simulate environmental effects on the valves. The basis for this recommendation is that testing during Mode 5 may not be a representative test for assessing PORV performance under normal plant operating conditions. Surveillance Requirement 4.4.4.1.b has therefore been modified to require that the PORVs be cycled during Modes 3 or 4. It clarifies that the PORVs should only be cycled during that part of Mode 4 which is not in the LTOP regime and further specifies that the block valve must be closed prior to stroking the PORV to preclude the possibility of a loss of coolant should the PORV fail open with the block valve open. (The block valve would be closed just prior to stroking the PORV so that environmental conditions on the PORV are maintained.)

Surveillance Requirement 4.4.4.2

- This surveillance requirement exempts the block valve from having to be demonstrated operable when it is closed to isolate an inoperable PORV. The exemption is maintained for the case when a block valve is closed with power maintained to isolate a PORV which is inoperable due to excessive leakage. This is to prevent having to open the block valve and discharging large quantities of effluent to the pressurizer relief tank in the case of a severely-leaking PORV. If the block valve is closed with power removed to isolate an otherwise inoperable PORV (the proposed TS allows McGuire to operate with one PORV/block valve inoperable for an extended period of time), the block valve would not need to be demonstrated operable because redundant accident mitigation capability would be provided by the two remaining PORVs/block valves.

Surveillance Requirement 4.4.4.3

- Surveillance Requirement 4.4.4.3 has been added which requires demonstrating the PORV emergency power supply operable. It requires the valve motive power to be transferred from its normal instrument air supply to its emergency nitrogen supply, the air supply to be isolated and vented, and the PORV to be operated through a completed cycle of full travel. The generic letter also recommends that the surveillance requirements require operating solenoid air control valves and check valves on air accumulators in PORV control systems through a complete cycle of full travel. Surveillance Requirement 4.4.4.3 requires the PORVs to be stroked while aligned to the emergency nitrogen supply, with the normal air supply vented; this cycles the necessary valves. It is therefore not necessary to expand the surveillance requirements any further in this regard. Also, the guidance contained in the generic letter indicates that motive and control power for the PORVs and block valves should be manually transferred from the normal to the emergency power bus. At McGuire, transferring control power does not apply because the PORVs and block valves are normally powered from an essential bus. Transferring motive power does not apply to the block valves; they are not pneumatic. Hence, the McGuire specification, as currently modified, complies with the guidance contained in the generic letter and is therefore acceptable.

Limiting Condition for Operation 3.4.9.3

- In LCO 3.4.9.3, the applicability is modified to change the phrase "with the reactor vessel head on" to "when the head is on the reactor vessel." It should be noted that in the guidance contained in the generic letter, it is suggested that the applicability of the LCO be clarified to exclude Mode 6 when the Reactor Coolant System is adequately vented and that the depressurizing and venting of the Reactor Coolant System should no longer be classified as an overpressure protection system. The generic letter recommends that an additional action statement be added to specify verifying the vent pathway when the Reactor Coolant System is depressurized and vented. This appears inappropriate, because once the Reactor Coolant System is vented, LCO 3.4.9.3 would no longer apply and the action statement requiring verification of the vent pathway would therefore not have to be entered. For this reason, the licensee proposed that the present structure of the McGuire TS be maintained in that the depressurizing and venting of the Reactor Coolant System will continue to be classified as an overpressure protection system and the requirement to verify the vent pathway when the system is depressurized and vented will continue to be governed by Surveillance Requirement 4.4.9.3.2. The staff agrees with the licensee's proposal.

- Action statement a. is modified to clarify that it is only applicable in Mode 4. Also, the phrase "depressurize and vent" has been changed to "complete depressurization and venting of" to avoid any possible questions as to when the required depressurization and venting action must be completed. It also provides for consistency in wording throughout the LCO and is therefore acceptable.

- A new action statement b. is added which prohibits entering a water-solid condition in the Reactor Coolant System when one PORV is inoperable while in Mode 5. The licensee proposed an allowable outage time of 72 hours for one PORV while in Mode 5. When the Reactor Coolant System is not water-solid, 72 hours is viewed as a reasonable compromise between the 7-day allowable outage time for Mode 4 and the 24-hour allowable outage time for Mode 6 (see below). Also, 72 hours provides sufficient time for completing crud burst activities. The licensee proposed a 24-hour time limit for completing depressurization and venting activities for this action statement in order to allow for a more controlled depressurization that could be accomplished in 8 hours. The staff agrees with this proposal.

- A new action statement c. is added to reduce the allowable outage time for an inoperable PORV to 24 hours while in Mode 6.
- Old action statement b. becomes new action statement d. and the phrase "depressurize and vent" has been changed to "complete depressurization and venting of" for the reason as stated before.
- Old action statement c. becomes new action statement e.
- Old action statement d. becomes new action statement f.

These changes are acceptable.

Surveillance Requirement 4.4.9.3.1

- Surveillance Requirement 4.4.9.3.1.a. is simplified by removing requirements that exist because of general requirements applicable to all surveillance requirements as specified in Section 4.0 of the TS. This is consistent with GL 90-06 and is acceptable.

Bases Section 3/4.4.4

- The Bases for the PORV and block valve TS have been expanded to identify the major functions of the PORVs and block valves. These major functions are as follows:

    1) Manual control of Reactor Coolant System pressure following accidents;

    2) Maintaining reactor coolant pressure boundary integrity by controlling leakage;

    3) Manual control of block valves to isolate and unblock PORVs (for manual pressure control and for controlling PORV leakage);

    4) Automatic control of Reactor Coolant System pressure to prevent code safety valve challenges; and

    5) Manual control of block valves to isolate a stuck-open PORV.

The expanded Bases more clearly delineate these functions.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 59748 dated November 10, 1993). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

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

Principal Contributor: V. Nerses

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