



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. 139
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 October 28, 1992, as supplemented December 14, 1993, 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.

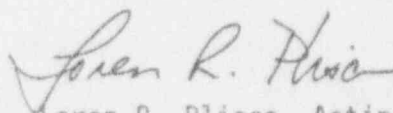
2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-9 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 139, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 31, 1994



UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

DUKE POWER COMPANY

DOCKET NO. 50-370

McGUIRE NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. NPF-17

- I. 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-17 filed by the Duke Power Company (licensee) dated October 28, 1992, as supplemented December 14, 1993, 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. 121, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION



Loren R. Plisco, Acting Director
Project Directorate II-3
Division of Reactor Projects - 1/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 31, 1994

ATTACHMENT TO LICENSE AMENDMENT NO.139

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO.121

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

Index Page X

3/4 4-30

3/4 4-35

B 3/4 4-8

Insert Pages

Index Page X

3/4 4-30

3/4 4-35

B 3/4 4-8

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
3/4.4.9	PRESSURE/TEMPERATURE LIMITS	
	Reactor Coolant System.....	3/4 4-30
FIGURE 3.4-2a	UNIT 1 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY.....	3/4 4-31
FIGURE 3.4-2b	UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS-APPLICABLE UP TO 10 EFPY.....	3/4 4-32
FIGURE 3.4-3a	UNIT 1 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY.....	3/4 4-33
FIGURE 3.4-3b	UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS-APPLICABLE UP TO 10 EFPY.....	3/4 4-34
TABLE 4.4-5	[DELETED].....	3/4 4-35
	Pressurizer.....	3/4 4-36
	Overpressure Protection Systems.....	3/4 4-37
3/4.4.10	STRUCTURAL INTEGRITY.....	3/4 4-39
3/4.4.11	REACTOR VESSEL HEAD VENT SYSTEM.....	3/4 4-40
<u>3/4.5 EMERGENCY CORE COOLING SYSTEMS</u>		
3/4.5.1 ACCUMULATORS		
	Cold Leg Injection.....	3/4 5-1
	[Deleted].....	3/4 5-3
3/4.5.2	ECCS SUBSYSTEMS - $T_{avg} \geq 350^{\circ}F$	3/4 5-5
3/4.5.3	ECCS SUBSYSTEMS - $T_{avg} \leq 350^{\circ}F$	3/4 5-9
3/4.5.4	[Deleted].....	3/4 5-11
3/4.5.5	REFUELING WATER STORAGE TANK.....	3/4 5-12

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3, 3.4-4, and 3.4-5 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

- a. Maximum heatup rates as specified in Figures 3.4-2 and 3.4-3
- b. Maximum cooldown rates as specified in Figures 3.4-4 and 3.4-5
- c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, 3.4-4, and 3.4-5.

This Page Intentionally Deleted

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, ASTM E185-73, and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT} , at the end of the effective full power years (EFPY) of service life identified on the applicable technical specification figure. The 10 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT} ; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the fluence, copper content, and phosphate content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} . For Unit 1, the adjusted reference temperature has been computed by Regulatory Guide 1.99, Revision 2. For Unit 2, the adjusted reference temperature has been computed as discussed in WCAP-11029. The heatup and cooldown limit curves of Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5 include predicted adjustments for this shift in RT_{NDT} at the end of the identified service life. Adjustments for possible errors in the pressure and temperature sensing instruments are included when stated on the applicable figure.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed in accordance with the requirements of ASTM E185-73 and 10 CFR 50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the pressure vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the pressure vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.