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

July 18, 1989

Docket Nos.: 50-369
and 50-370

Mr. H. B. Tucker, Vice President
Nuclear Production Department
Duke Power Company
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: ISSUANCE OF AMENDMENT NO.100 TO FACILITY OPERATING LICENSE NPF-9 AND AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NPF-17 - MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TACS 71963/71964)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 100to Facility Operating License NPF-9 and Amendment No. 82 to Facility Operating License NPF-17 for the McGuire Nuclear Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TS) in response to your application dated January 22, 1989 as supplemented May 17 and June 19, 1989.

The amendments update pressure and temperature (P-T) limits in TS 3/4.4.9 for heatup and cooldown of the reactor coolant system and associated Table 4.4-5 on the withdrawal and examination schedule for reactor vessel material irradiation surveillance specimens. The amendments are effective as of their date of issuance.

In accordance with Generic Letter 88-11, NRC review of P-T limits for McGuire Unit 2 has been based upon Revision 2 to Regulatory Guide 1.99. We find that the proposed limits contain sufficient margin to account for neutron irradiation damage throughout 5 effective full power years (EFPY). Currently, the earliest projected date for reaching 5 EFPY occurs at the end of fuel cycle 6. Therefore, proposed TS Figures 3.4-3 and 3.4-5 have been annotated to limit staff approval to completion of the refueling outage at the end of Unit 2 fuel cycle 6. This annotation was discussed with Mr. Bob Gill of your company on June 26, 1989. To provide for NRC staff review of the next set of P-T curves and satisfy the scheduling requirements of Generic Letter 88-11, you are requested to provide the next set of McGuire Unit 2 P-T limits four months prior to restart following completion of the refueling outage at the end of fuel cycle 6.

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A copy of the related safety evaluation supporting Amendment No. 100 to Facility Operating License NPF-9 and Amendment No. 82 to Facility Operating License NPF-17 is enclosed.

Notice of issuance of amendments will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

Original signed by:

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

Enclosures:
1. Amendment No. 100 to NPF-9
2. Amendment No. 82 to NPF-17
3. Safety Evaluation

cc w/enclosures:
See next page

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LA:PDII-3	PM:PDII-3	D:PDII-3
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Duke Power Company

McGuire Nuclear Station

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DATED: July 18, 1989

AMENDMENT NO. 100TO FACILITY OPERATING LICENSE NPF-9 - McGuire Nuclear Station, Unit 1
AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NPF-17 - McGuire Nuclear Station, Unit 2

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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. 100
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 January 22, 1989, as supplemented May 17 and June 19, 1989, 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, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 100, 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 and shall be implemented within 14 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: July 18, 1989

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*See previous concurrence

*LA:PDII-3	*PM:PDII-3	*DEST:EMTB	*OGC	D:PDII-3
MRood	DHood:	CCheng		DMatthews
06/19/89	06/19/89	06 /27/89	07/03/89	7/17/89

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. , 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

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance:

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DSH for	DSH	BJC		
LA:PDII-3	PM:PDII-3	DE:SRXB:EMTB	OGC	D:PDII-3
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6 /19 /89	6 /19 /89	6/20 /89	7/3 /89	/ /89

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. 82
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 January 22, 1989, as supplemented May 17 and June 19, 1989, 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, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations 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. 82 , 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 and shall be implemented within 14 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: July 18, 1989

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*See previous concurrence

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06/19/89	06/19/89	06/27/89	07/03/89	7/18/89

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. , 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

David B. Matthews, Director
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance:

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DSH/m	DSH	BGE		
LA:PDII-3	PM:PDII-3	DEST/EMTB	OGC	D:PDII-3
MRood	DHood:	CCheng		DMatthews
6/19/89	6/19/89	6/27/89	7/3/89	/ /89

ATTACHMENT TO LICENSE AMENDMENT NO. 100

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 82

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 4-30	
3/4 4-31	
3/4 4-32	
3/4 4-33	
3/4 4-34	
3/4 4-35	3/4 4-36
B 3/4 4-7	
B 3/4 4-8	
B 3/4 4-13	B 3/4 4-14
B 3/4 4-16	B 3/4 4-15

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.

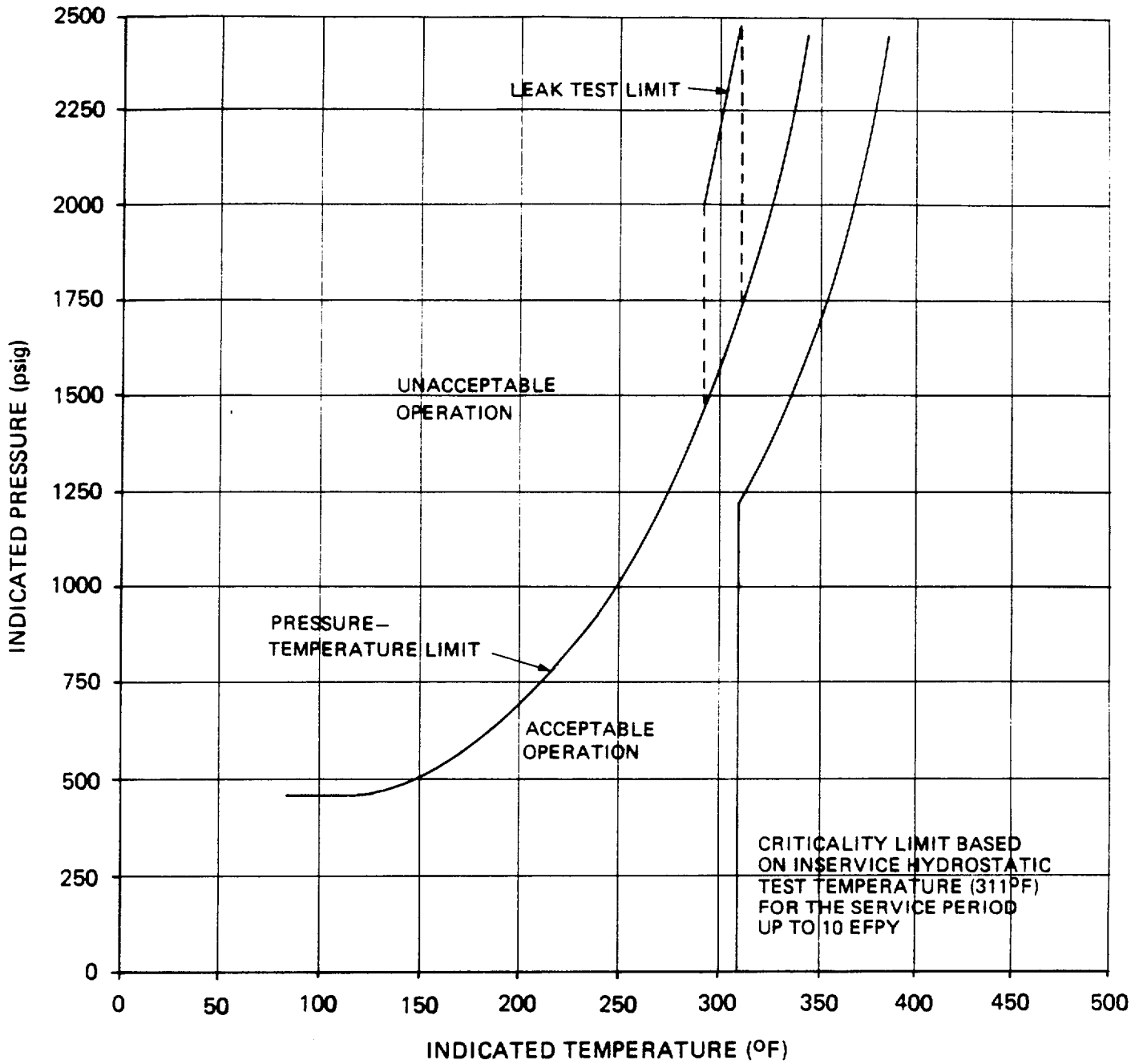
ACTION:

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_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

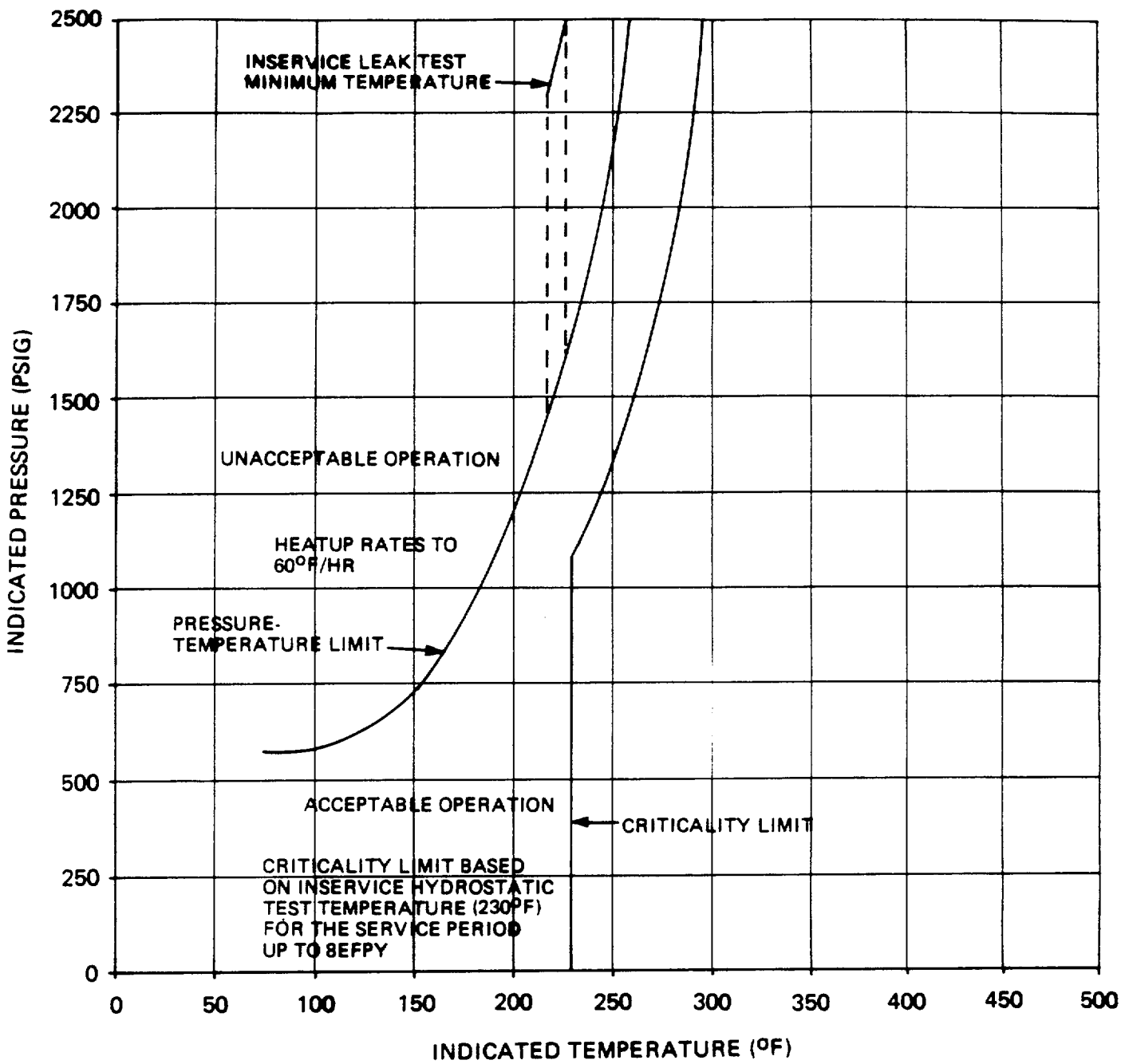
4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50, Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, 3.4-4, and 3.4-5.



CURVE APPLICABLE FOR HEATUP RATES UP TO 60°/HR FOR THE SERVICE PERIOD UP TO 10 EFPY CONTAINS MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL–WELD METAL
COPPER CONTENT–0.30wt%
PHOSPHORUS CONTENT–0.013wt%
RT_{NDT} INITIAL–0°F 1/4T, 165.5°F
RT_{NDT} AFTER 10 EFPY 3/4T, 113°F

FIGURE 3.4–2 McGUIRE UNIT 1, REACTOR COOLANT SYSTEM, HEATUP LIMITATIONS NRC RG 1.99 REV 2 APPLICABLE FOR THE FIRST 10 EFPY

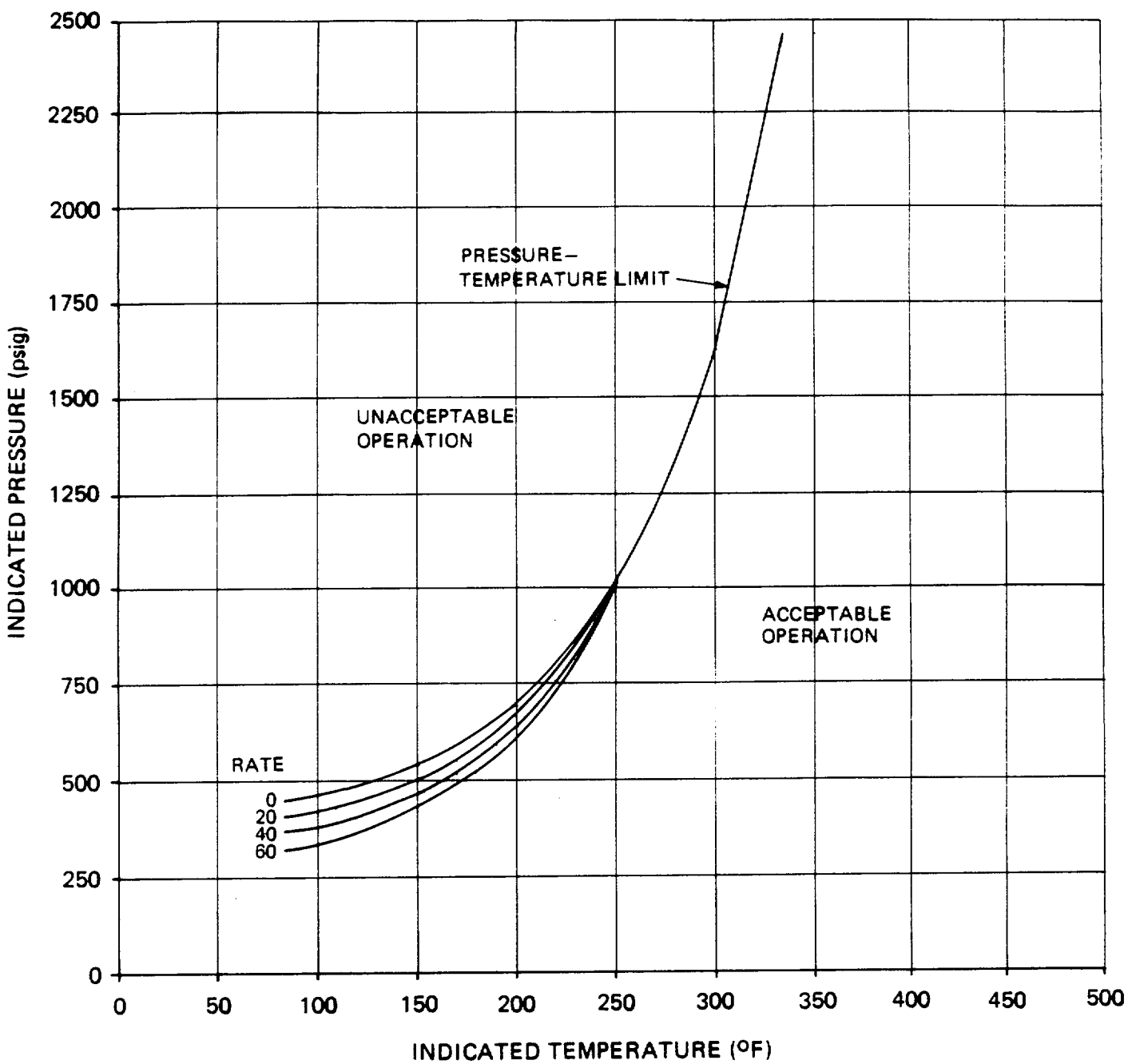


CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 8 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL—REACTOR VESSEL INTERMEDIATE SHELL 05
COPPER CONTENT 0.16
RT_{NDT} INITIAL −4°F
RT_{NDT} AFTER 8 EFPY
1/4T, 86°F
3/4T, 36°F

*Approved by NRC for first 5 EFPY or completion of the refueling outage at the end of fuel cycle 6, based on Generic Letter 88-11.

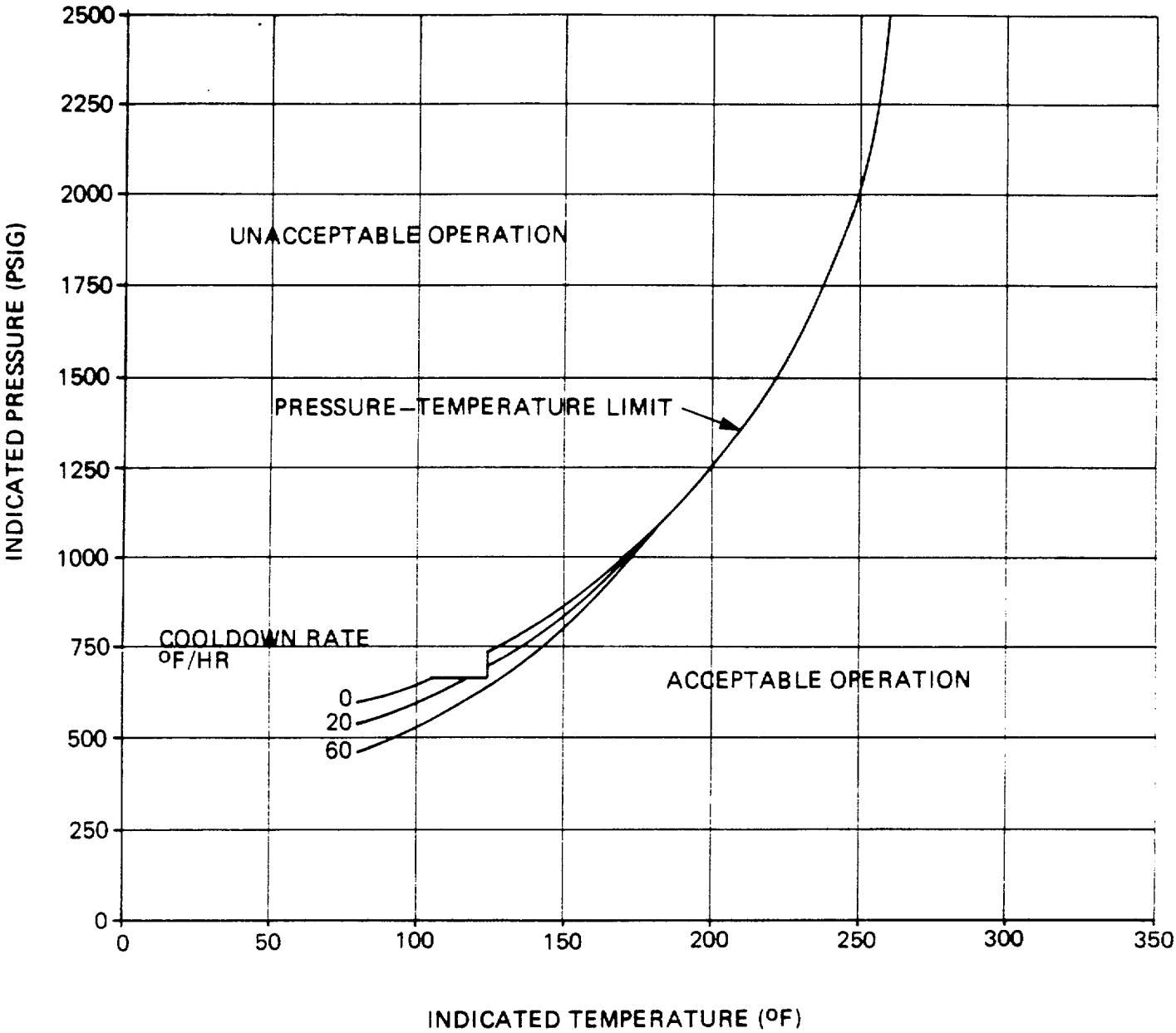
FIGURE 3.4-3 McGUIRE UNIT 2, REACTOR COOLANT SYSTEM, HEATUP LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY



CURVES APPLICABLE FOR COOLDOWN RATES UP TO 50°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY CONTAINS MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL–WELD METAL
COPPER CONTENT–0.30wt%
PHOSPHORUS CONTENT–0.013wt%
RT_{NDT} INITIAL–0°F 1/4T, 165.5°F
RT_{NDT} AFTER 10 EFPY 3/4T, 113°F

FIGURE 3.4–4 McGUIRE UNIT 1, REACTOR COOLANT SYSTEM, COOLDOWN LIMITATIONS NRC RG 1.99 REV 2 APPLICABLE FOR THE FIRST 10 EFPY



CURVES APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 8 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.

MATERIAL BASIS
CONTROLLING MATERIAL–REACTOR VESSEL INTERMEDIATE SHELL 05
COPPER CONTENT–0.16wt%
RT_{NDT} INITIAL–4°f F
RT_{NDT} AFTER 8 EFPY 1/4T,85°F
3/4T,36°F

*Approved by NRC for first 5 EFPY or completion of the refueling outage at the end of fuel cycle 6, based on Generic Letter 88-11.

FIGURE 3.4-5 McGUIRE UNIT 2, REACTOR COOLANT SYSTEM, COOLDOWN LIMITATIONS APPLICABLE FOR THE FIRST 8 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)* UNIT 1	UNIT 2
1. U	56°	4.76	Removed	7
2. V	58.5°	4.06	8	Removed
3. W	124°	4.76	Standby	Standby
4. X	236°	4.76	Removed	4
5. Y	238.5°	4.06	15	15
6. Z	304°	4.76	Standby	Standby

*Withdrawal time may be modified to coincide with those refueling outages or plant shutdowns most closely approaching the withdrawal schedule.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

SPECIFIC ACTIVITY (Continued)

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective ACTION. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G:

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

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 termperature 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 surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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.

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PRESSURE/TEMPERATURE LIMITS (Continued)

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \qquad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \qquad (2)$$

Where: K_{IM} is the stress intensity factor caused by membrane (pressure) stress,

K_{It} is the stress intensity factor caused by the thermal gradients,

K_{IR} is provided by the code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heautp or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses

PRESSURE/TEMPERATURE LIMITS (Continued)

resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factor, K_{IT}, for the reference flaw is computed. From Equation (2) the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

COOLDOWN

For the calculation of the allowable pressure versus coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that at any given reactor coolant temperature, the delta T developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the

PRESSURE/TEMPERATURE LIMITS (Continued)

end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses, of course, are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses, at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves in technical specifications for the heatup rate data and the cooldown rate data may be adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves. Where technical specification curves have not been adjusted, such adjustments are made by plant procedures.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of at least 4.5 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 100TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 82 TO FACILITY OPERATING LICENSE NPF-17

DUKE POWER COMPANY

DOCKET NOS. 50-369 AND 50-370

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

1.0 INTRODUCTION

By letter dated January 22, 1989, as supplemented May 17, 1989, Duke Power Company (the licensee) proposed amendments to the operating licenses for McGuire Nuclear Station, Units 1 and 2 to change the Technical Specification (TS). The proposed changes would update pressure and temperature (P-T) limits in TS 3/4.4.9 for heatup and cooldown of the reactor coolant system, including associated TS Table 4.4-5 on the withdrawal and examination schedule for reactor vessel material irradiation surveillance specimens. TS Bases 3/4.4.9 would be similarly updated to reference revised heatup and cooldown curves and information associated with their derivation and use.

Because the May 17 and June 19, 1989, submittals clarified or corrected certain aspects of the original submittal, the substance of the changes noticed in the Federal Register and the proposed no significant hazards determination were not affected.

2.0 EVALUATION

a. McGuire Unit 1 Heatup and Cooldown Curves

For McGuire Unit 1, these amendments replace the existing reactor coolant system (RCS) heatup and cooldown curves, referenced by TS 3/4.4.9.1, with new curves shown on TS Figures 3.4.2 and 3.4-4, respectively. As before, the new curves contain margins of 10° F and 60 psig for possible instrument errors, and are applicable for the service period up to ten effective full power years (EFPY). The new curves are based upon a Westinghouse Report, "McGuire Unit 1 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation" dated November 1988 and forwarded as Attachment 5 of the licensee's January 22, 1989 submittal. The method for predicting radiation embrittlement (i.e., determination of adjusted reference temperature for vessel beltline material) in this Westinghouse report is based upon Regulatory Guide (RG) 1.99, Revision 2. The associated maximum heatup or cooldown rate during normal operations as specified by TS

3.4.9.1a and TS 3.4.9.1b, respectively, is decreased from 100° F per hour to 60° F per hour. Duke's own associated administrative cooldown limit is not affected by these amendments and continues to be 50° F per hour.

The new RCS heatup and cooldown curves for McGuire Unit 1 are needed because the existing TS limits, based on analysis of surveillance Capsule U as documented in the licensee's letter of April 5, 1985 and Westinghouse Report WCAP-10786, are valid up to 4.86 EFPY. At the end of fuel cycle 5 (October 1988), McGuire Unit 1 had reached about 4.3 EFPY and was projected to reach the existing limit by about June 1989. Thus, absent this amendment, the existing Unit 1 heatup and cooldown P-T limits would become non-conservative about mid-1989.

The licensee also notes that the new Unit 1 operating limits are intended to apply for a limited period of time. During the end of fuel cycle 5 refueling outage (October - December 1988), Capsule X was removed from the Unit 1 vessel for analysis and for development of new P-T limit curves using RG 1.99, Revision 2. Pursuant to 10 CFR 50, Appendix H, results of analysis of this capsule will be provided to the NRC in late 1989 (i.e., within one year of removal of the capsule). The licensee will propose amendments to incorporate the resulting operating limits into the TS shortly thereafter.

The staff has reviewed the P-T limits and curves for McGuire Unit 1, including the November 1988 Westinghouse report. We find that the fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary have been determined in accordance with Standard Review Plan Chapter 5.3.2 and that the approach defined in Appendix G to the ASME Code Section III was followed to calculate the allowable limit curves for heatup and cooldown rates. Accordingly, since the new curves are based on results of capsule analyses performed with NRC approved methods, and since the new curves are conservative with respect to the existing P-T operating limits, we find that they appropriately reflect the change in material toughness of the reactor vessel due to irradiation effects and are acceptable.

b. <u>McGuire Unit 2 Heatup and Cooldown Curves</u>

For McGuire Unit 2, these amendments replace the existing RCS heatup and cooldown curves with new curves shown on TS Figures 3.4-3 and 3.4-5 respectively. The new curves are needed to reflect adjustments to existing limits based upon analysis of the last surveillance capsule removed from the Unit 2 vessel. Results of the analysis of this last capsule, Capsule V, were provided by the licensee's letter of April 2, 1986 and by WCAP-11029. As before, the new curves contain margins of 10° F and 60 psig for possible instrument errors.

The new curves are proposed to be applicable only for the first 8 EFPY. The curves provided in WCAP-11029 were developed using Revision 1 to RG 1.99. As with Unit 1 curves, the associated maximum heatup or cooldown rate during normal operations as specified by TS 3.4.9.1a and TS 3.4.9.1b, respectively, is decreased from 100° F per hour to 60° F per hour, which is more in line with Duke's own administrative cooldown limit of 50° F per hour.

The licensee notes that the new Unit 2 operating limits are intended to apply for a limited period of time. During the end of fuel cycle 5 refueling outage (July - September 1989), Capsule X will be removed from the Unit 2 vessel for analysis and for development of new P-T limit curves using RG 1.99, Revision 2. Pursuant to Appendix H of 10 CFR 50, results of analyses of this capsule will be provided to the NRC during the third quarter of 1990. The licensee will propose amendments to incorporate the resulting operating limits into the TS shortly thereafter.

On July 12, 1988 the Commission issued Generic Letter (GL) 88-11 "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," forwarding Revision 2 to RG 1.99 and noting that it would be used to review P-T limits and embrittlement analyses. The Commission stated that all actions (hardware, procedures, and/or staff modifications) resulting from the use of Revision 2 should be completed (fully implemented and operational) within two refueling outages (approximately 3 years) after the effective date of Revision 2 to RG 1.99. Using Revision 2 and the surveillance data reported in WCAP-11029, we find that the proposed P-T limits contain sufficient margin to account for neutron irradiation damage through 5 EFPY. McGuire Unit 2 has presently achieved 4.1 EFPY (June 1989) and is conservatively projected to reach 5 EFPY no sooner than the end of fuel cycle 6 (August - November 1990). On this basis, we find use of the Unit 2 P-T curves acceptable until completion of the refueling outage at the end of Unit 2 fuel cycle 6. Moreover, no waiver of the implementation requirement for Revision 2 of RG 1.99 is implied or intended by these amendments.

c. <u>Revised Capsule Withdrawal Schedule</u>

These amendments update TS Table 4.4-5, "Reactor Vessel Material Surveillance Program - Withdrawal Schedule," reflecting separate withdrawal schedules for Unit 1 and Unit 2 capsules, and consistent with the above discussions, denoting the previous removal of Unit 1

Capsules U and X and Unit 2 Capsule V. The associated lead factors in the table are also updated. The lead factors represent the relationship between the fast neutron flux density at the capsule location and the inner wall of the pressure vessel and are used along with the capsule withdrawal time to predict future radiation damage to the pressure vessel material (The heatup and cooldown curves are recalculated when the change in nil-ductility reference temperature (ΔRT_{NDT}) exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure). These revisions to the table are based upon information provided by Westinghouse in Section 7 of WCAP-10786 for McGuire Unit 1 and in WCAP-11029 for McGuire Unit 2.

The NRC staff has reviewed the revisions to TS Table 4.4-5 and finds that they are in accordance with the requirements of ASTM E 185-82 and 10 CFR 50, Appendix H, and are, therefore, acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational exposure. The NRC staff has made a determination that the amendments involve no significant hazards consideration, and there has been no public comment on such finding. 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 these amendments.

4.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register (54 FR 13763) on April 5, 1989. The Commission consulted with the state of North Carolina. No public comments were received, and the state of North Carolina did not have any comments.

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

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Dated: July 18, 1989