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

SUBJECT: ISSUANCE OF AMENDMENTS - McGUIRE NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M91901 AND M91902)

Dear Mr. McMeekin:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. **162** to Facility Operating License NPF-9 and Amendment No. **144** 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 March 29, 1995, supplemented by letters dated September 18 and November 16, 1995.

The amendments revise TS requirements for the Low Temperature Overpressure Protection (LTOP) system and update the heatup and cooldown curves for both units.

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, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-369 and 50-370

Enclosures:
1. Amendment No. **162** to NPF-9
2. Amendment No. **144** to NPF-17
3. Safety Evaluation

cc w/encl: See next page

DOCUMENT NAME: G:\MCGUIRE\91901.AMD

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

January 11, 1996

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Vice President, McGuire Site
Duke Power Company
12700 Hagers Ferry Road
Huntersville, NC 28078-8985

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Mr. T. C. McMeekin
Duke Power Company

McGuire Nuclear Station

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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. 162
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 March 29, 1995, supplemented by letters dated September 18 and November 16, 1995, 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.

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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. 162 , 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

Robert E Martin
for Leonard A. Wiens, Acting Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 11, 1996

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. 144
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 March 29, 1995, supplemented by letters dated September 18 and November 16, 1995, 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. 144 , 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

Robert E Martin

for Leonard A. Wiens, Acting Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Technical Specification
Changes

Date of Issuance: January 11, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 162

FACILITY OPERATING LICENSE NO. NPF-9

DOCKET NO. 50-369

AND

TO LICENSE AMENDMENT NO. 144

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
Index Page X	Index Page X
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-37	3/4 4-37
3/4 4-38	3/4 4-38
3/4 4-39	3/4 4-39
3/4 4-40	3/4 4-40*
	3/4 4-41*
3/4 5-9	3/4 5-9
3/4 5-10	3/4 5-10
B 3/4 4-8	B 3/4 4-8
B 3/4 4-16	B 3/4 4-16
B 3/4 4-17	B 3/4 4-17

* only page number change

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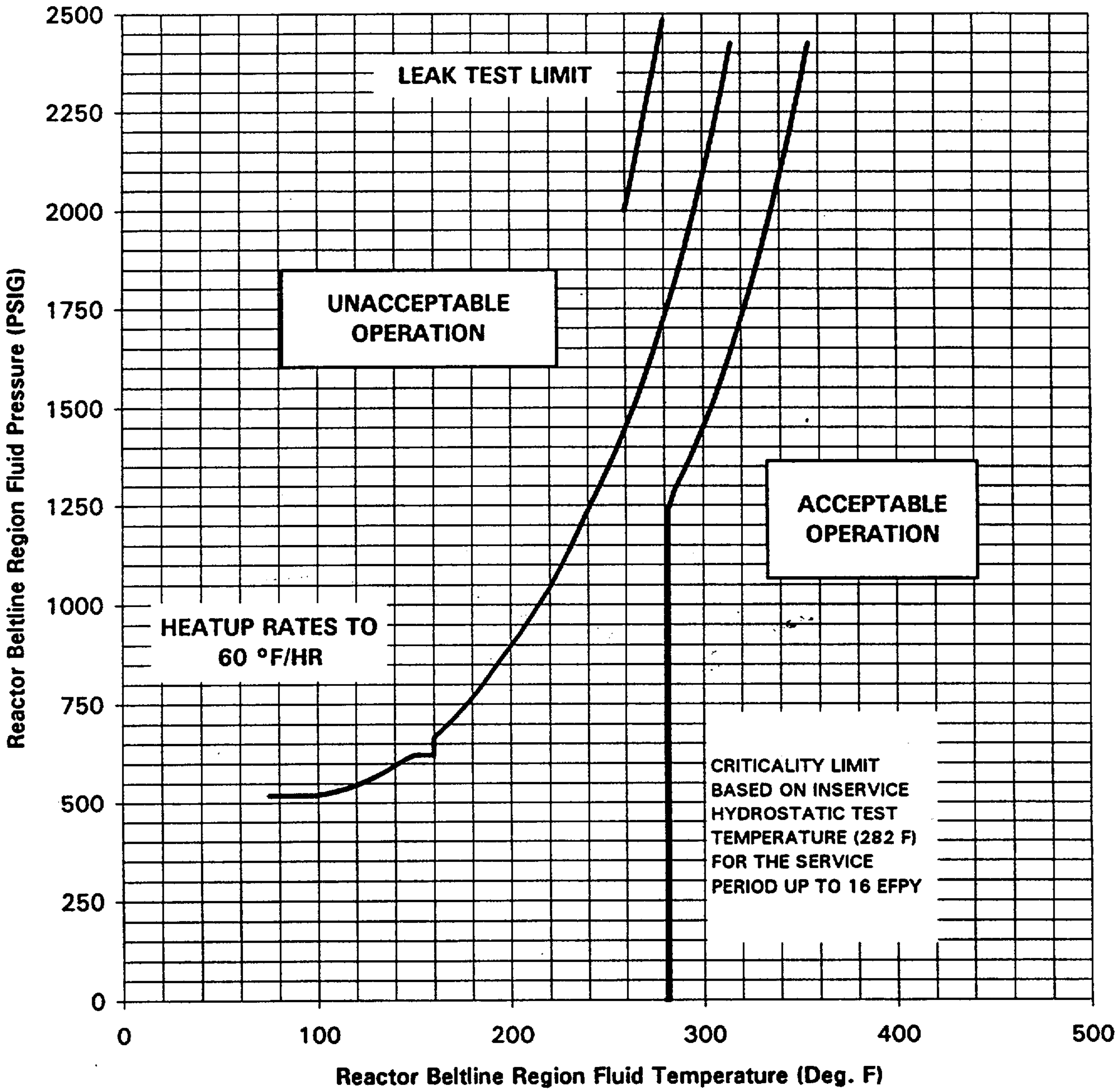
LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING MATERIALS: LOWER SHELL LONGITUDINAL WELDS 3-442A and LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY:

1/4-t, 149.5 deg. F
3/4-t, 102.0 deg. F

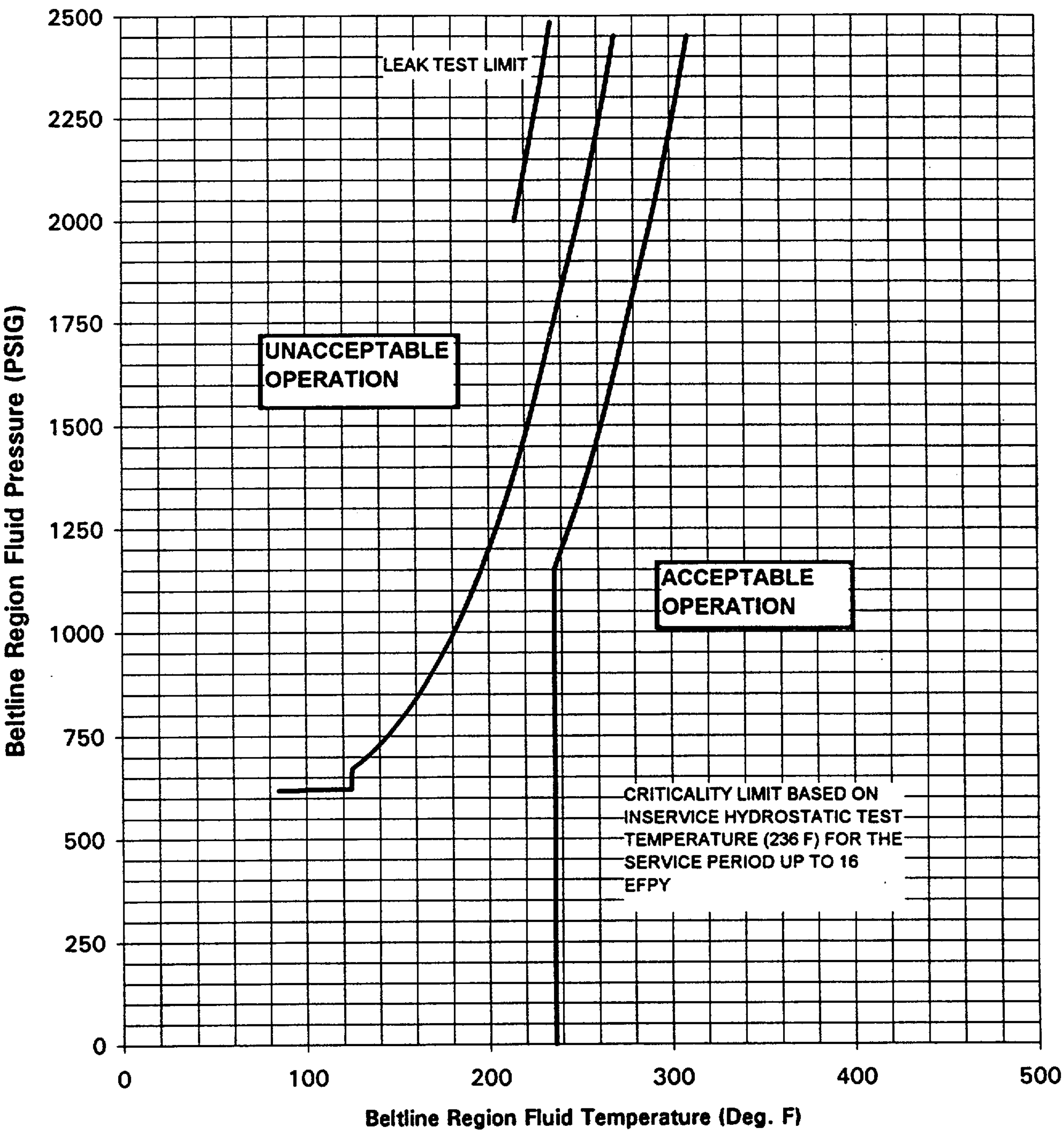


McGuire Unit 1 Reactor Coolant System Heatup Limitations
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the first 16 EFPY

Figure 3.4-2

LIMITING MATERIALS: LOWER SHELL FORGING 04
LIMITING ART AT 16 EFPY:

1/4-t, 104 deg F
3/4-t, 73 deg F



McGuire Unit 2 Reactor Coolant System Heatup Limitations
(Without Margins for Instrumentation Errors)
NRC REG GUIDE 1.99 Rev. 2
Applicable for the First 16 EFPY

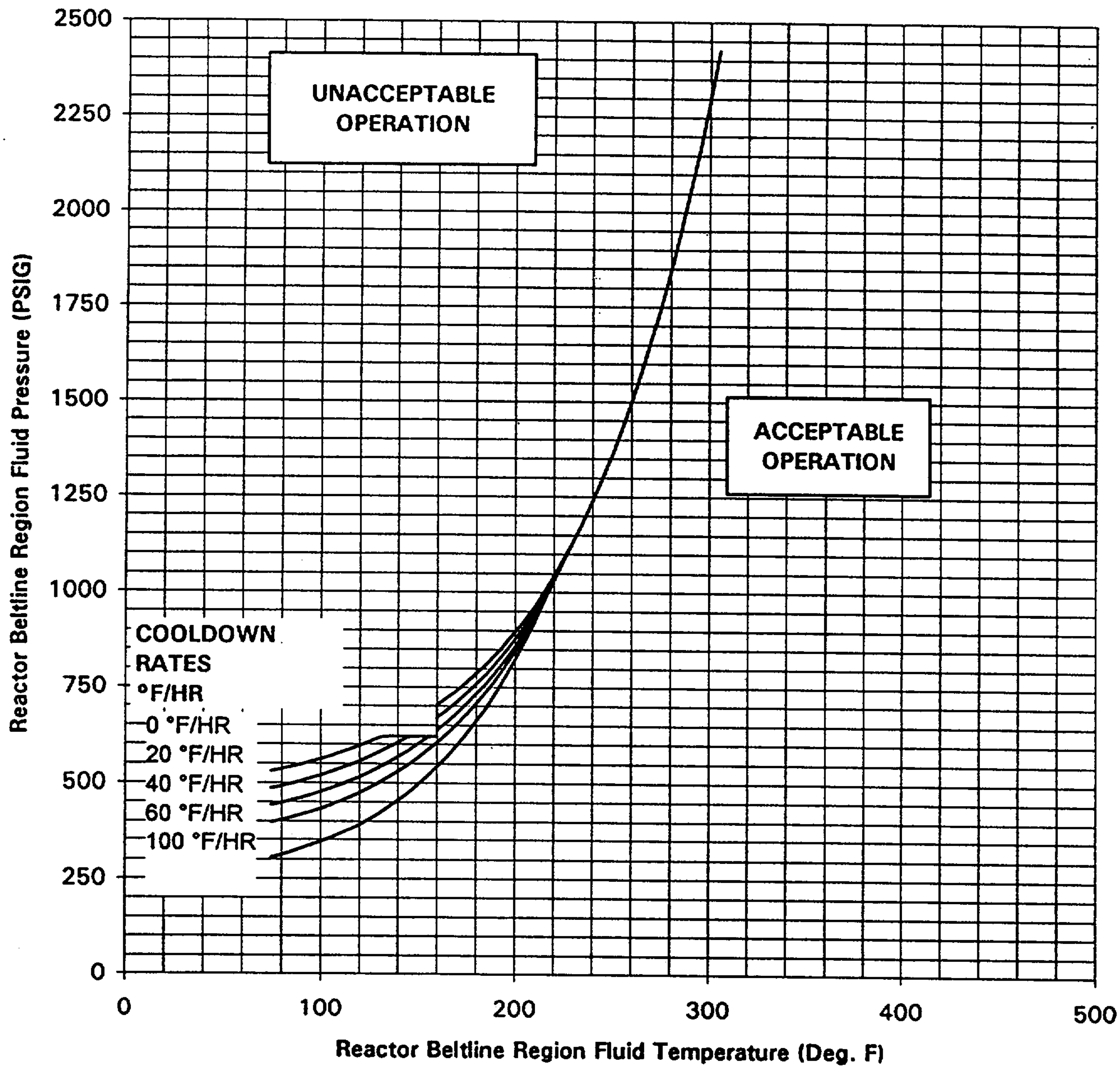
Figure 3.4-3

MATERIAL PROPERTY BASIS:

LIMITING MATERIALS:

LOWER SHELL LONGITUDINAL WELDS 3-442A and
LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY:
1/4-t, 149.5 deg. F
3/4-t, 102.0 deg. F

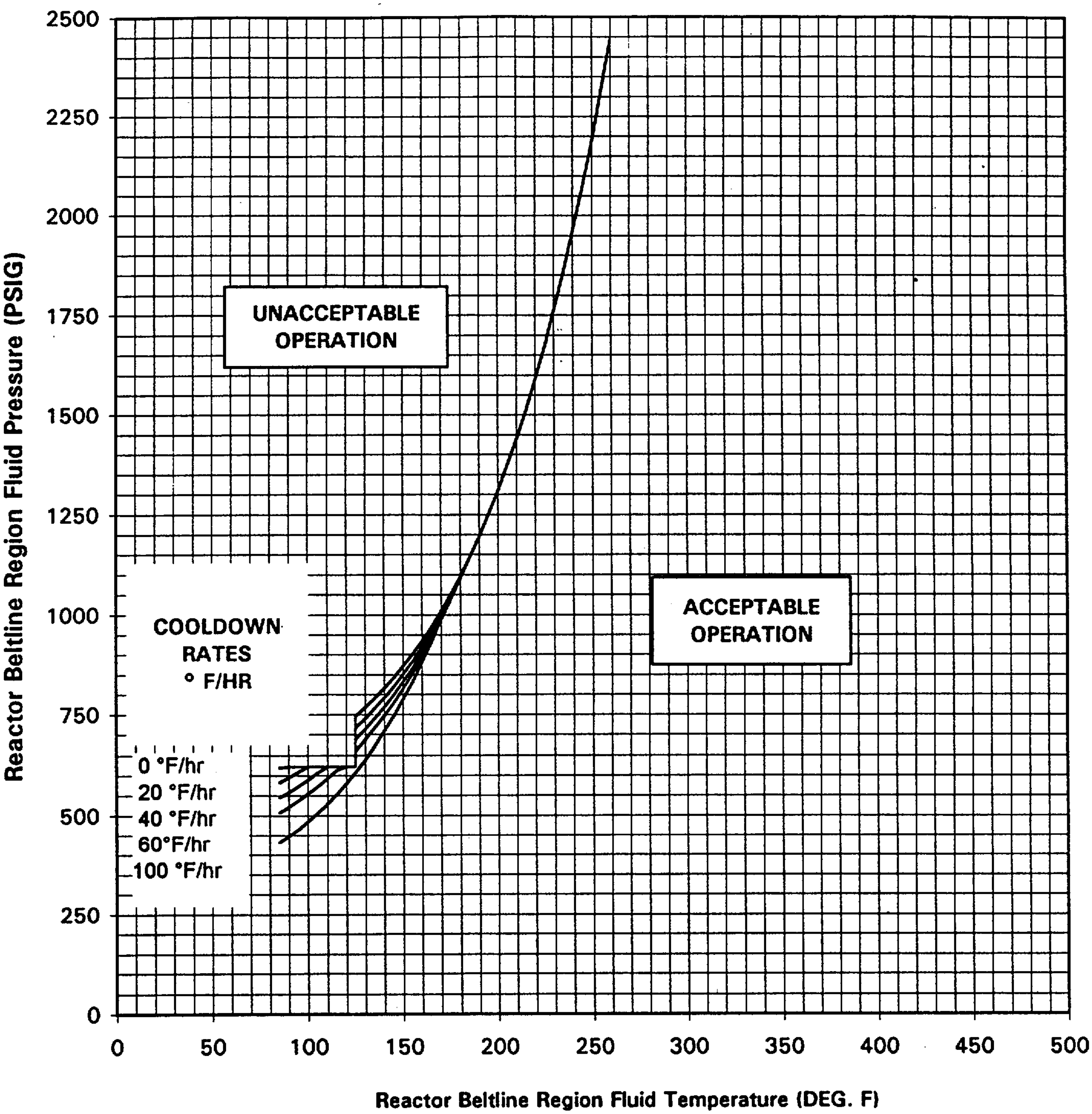


McGuire Unit 1 RCS Cooldown Limitations,
Cooldown Rates up to 100 deg. F/HR
(Without Margins for Instrumenation Errors)
NRC REG GUIDE 1.99, REv. 2
Applicable for the First 16 EFPY

Figure 3.4-4

LIMITING MATERIALS: LOWER SHELL FORGING 04

LIMITING ART AT 16 EFPY:
1/4-t, 104 deg F
3/4-t, 73 deg F



McGuire Unit 2 RCS Cooldown Limitations,
Cooldown Rates up to 100 deg F/HR
(without Margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the First 16 EFPY

LIMITING CONDITION FOR OPERATION

3.4.9.3 As a minimum, a Low Temperature Overpressure Protection (LTOP) System shall be OPERABLE as follows:

a. A maximum of one Centrifugal Charging (NV) pump or one Safety Injection (NI) pump capable of injecting into the Reactor Coolant System (RCS) with all remaining NV and NI pump motor circuit breakers open or the discharge of the remaining NV and NI pumps isolated from the RCS by at least 2 valves with power removed#

AND

b. All accumulators isolated

AND

c. One of the following conditions met:

1. Two PORVs with a lift setting of ≤ 385 psig

OR

2. The RCS depressurized with a vent of ≥ 2.75 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 two or more Charging (NV) or Safety Injection(NI) pumps capable of injecting into the RCS*, immediately initiate action to restore a maximum of one NI or one NV pump capable of injecting into the RCS.

Two Charging pumps (NV or NI) maybe capable of injecting into the RCS during pump swap operation for ≤ 15 Minutes.

* One Safety Injection pump and one Charging pump, or two Charging pumps may be operated concurrently provided:

1. RHR suction relief valve (ND-3) is OPERABLE, and the RHR suction isolation valves (ND-1 and ND-2) are open and one of the following conditions is met:

a. RCS cold leg temperature is greater than 167° F or

b. RCS cold leg temperature is greater than 107° F and cooldown rate is less than 20° F per hour.

OR

2. Two PORVs secured in the open position with their associated block valves open and power removed.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (continued)

b. With an accumulator not isolated, isolated the affected accumulator within 1 hour. If required action is not met, either:

   1. Depressurize the accumulator to less than the maximum RCS pressure for the existing cold leg per Specification 3/4.4.9 within 12 hours,

   OR

   2. Increase RCS cold leg temperature to greater than or equal to 300° F within 12 hours.

c. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days. If required action is not met, depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

d. With one PORV inoperable in MODES 5 or 6, suspend all operations which could lead to a water-solid pressurizer. Restore the inoperable PORV to OPERABLE status within 24 hours. If required action is not met, either:

   1. Ensure RCS temperature is greater than 167° F, and ND-3 is OPERABLE, and ND-1 and ND-2 are open within one hour.

   OR

   2. Depressurize the RCS and vent through at least a 2.75 square inch vent within 8 hours.

e. With the LTOP system inoperable for any reason other than a., b., c., or d. above, depressurize the RCS and vent through a least a 2.75 square inch vent within 8 hours.

f. In the event that either the PORVs or the RCS vent are used to mitigate and 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 circumstance initiating the transient, the effect of the PORVs or vent on the transient, and any corrective action necessary to prevent recurrence.

g. 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 Once every 12 hours*, verify that an RCS vent of ≥ 2.75 square inches is open when the vent is used for overpressure protection.

4.4.9.3.3 Once every 12 hours, verify that each accumulator is isolated and that only one NV or NI pump is capable of injecting into the RCS.

4.4.9.3.4 Once every 12 hours, verify that RHR suction isolation valves ND-1 and ND-2 are open when RHR suction relief valve ND-3 is being used for overpressure protection.

4.4.9.3.5 Once every 72 hours, verify that the PORV block valve is open for each required PORV.

* 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.

A PORV secured in the open position may be used to meet this vent requirement provided that its associated block valve is open and power is removed.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.11 Two reactor vessel head vent paths, each consisting of two valves in series powered from emergency buses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With one of the above reactor vessel head paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both of the above reactor vessel head vent paths inoperable; maintain the inoperable vent path closed with power removed from the valve actuators of all the valves in the inoperable vent paths, and restore at least two of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING.

2. Verifying flow through the reactor vessel head vent paths during venting during COLD SHUTDOWN or REFUELING.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} F 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the centrifugal charging pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the RHR heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

A maximum of one centrifugal charging pump and one Safety Injection pump shall be capable of injecting into the RCS whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F. Two charging pumps may be operable and operating for ≤ 15 minutes to allow swapping charging pumps. Additional requirements are provided by Specification 3.4.9.3.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 All centrifugal charging pumps and Safety Injection pumps, not capable of injecting into the RCS shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position or by verifying the discharge of each pump has been isolated from the RCS by at least two isolation valves with power removed from the valve operators at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

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 1989 Edition of Section XI 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 16 EFPY service life period continues to ensure that the limiting RT_{NDT} at the 1/4T location in the core region is a bounding value. 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. 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.

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 2.75 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

PRESSURE/TEMPERATURE LIMITS (Continued)

the RCS cold legs are less than or equal to 300°F. Either of the PORVs or the RCS vent opening has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a HPSI pump and its injection into a water-solid RCS. The Pressurizer PORV setpoints for low temperature overpressure protection are based on limiting the peak pressure during the limiting transient to 1.10 times the ASME Section XI, Appendix G limits, in accordance with ASME code case N-514.

Credit is taken for the RHR suction relief valve (ND-3) during conditions where relieving capacity at rated accumulation is sufficient to prevent exceeding the above allowable pressure limits.

Cooldown limits/minimum RCS temperature restrictions ensure the allowable pressure limits will not be exceeded.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition and Addenda through Winter 1972.

3/4.4.11 REACTOR VESSEL HEAD VENT SYSTEM

Reactor Vessel Head Vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one reactor coolant system vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function. (Operability of the pressurizer steam space vent path is provided by Specifications 3/4.4.4 and 3/4.4.9.3.)

The valve redundancy of the reactor coolant system vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply or control system does not prevent isolation of the vent path.

The surveillance to verify Reactor Vessel Head Vent flowpath is qualitative as no specific size or flow rate is required to exhaust noncondensible gases. The function, capabilities, and testing requirements of the reactor coolant system vent systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements", November 1980.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NPF-9

AND AMENDMENT NO. 144 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 March 29, 1995, as supplemented by letters dated September 18 and November 16, 1995, 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 TS requirements for the Low Temperature Overpressure Protection system and update the heatup and cooldown curves for both units. The TS changes include TS 3.4.9.3 "Overpressure Protection Systems," TS 3.5.3, "ECCS subsystems - T_{avg} < 350°F," and updating of the Pressure Temperature limit curves, Figures 3.4-2, 3.4-3, 3.4-4 and 3.4-5. These proposed changes are to revise the Low Temperature Overpressure Protection (LTOP) system maximum setpoint and the minimum vent requirements, and enhance system operation and reliability. The September 18 and November 16, 1995, letters provided clarifying information that did not change the scope of the initial Federal Register notice and the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

2.1 LTOP Setpoint

The purpose of the LTOP system is to control the reactor coolant system (RCS) pressure at low temperature so that the integrity of the reactor coolant pressure boundary is not compromised by violating the P-T limits of 10 CFR Part 50, Appendix G, which is based on Appendix G, Section XI of the ASME Code. Currently, the McGuire TS state that the power-operated relief valve (PORV) lift setting be less than or equal to 400 psig. The licensee is proposing to reduce the PORV lift setting to 385 psig, a more conservative setpoint that allows the PORV to open earlier during an LTOP event. ASME Code Case N-514, approved by the staff for use at McGuire by letter dated September 30, 1994, is utilized to establish the lift setpoint of the PORV for overpressure protection during low temperature conditions. As delineated in the Code Case, the LTOP system shall limit the maximum pressure in the vessel to 110% (1.1) of the pressure determined to satisfy Appendix G, Section XI of the ASME Code.

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The licensee indicated that they evaluated the proposed setpoint by NRC approved methodology with three possible transients: (1) a mass input from an operable safety injection pump, (2) a mass input from an operable centrifugal charging pump, and (3) a heat input from a 50°F temperature difference between the secondary side of the steam generators and the RCS (consistent with current TS 3.4.1.3 and 3.4.1.4.1).

The licensee concluded that the PORV setpoint of 385 psig is sufficient to ensure that the peak reactor vessel beltline pressure is less than 1.1 times the ASME Section III, Appendix G, limits during anticipated pressure transients, provided appropriate limits on the heatup and cooldown rates are established. The beltline pressure includes instrument uncertainties, pressure corrections for the difference between the indicated pressure and the actual reactor beltline pressure, and the pressure corrections for the differential pressure across the reactor core. The most limiting pressure transient is the mass input from the inadvertent start of a safety injection pump with a worst case peak pressure of 537 psig. Therefore, to ensure that the reactor vessel pressure/temperature limits will not be exceeded, the licensee established limits on heatup and cooldown rates with controlled procedures for both units. The staff reviewed the limits on heatup and cooldown rates and found them acceptable.

2.2 LTOP Enable Temperature

Paragraph B.2 of the Branch Technical Position RSB 5-2, Revision 1, indicates that the LTOP System is required to be operable at a water temperature corresponding to a metal temperature of RT_{NDT} + 90°F at the beltline location that is controlling in the Appendix G limit calculations. Based on the limiting RT_{NDT} of 149.45°F (obtained from WCAP-13949) in Unit 1, the metal temperature would be 239.45°F. The LTOP enable temperature of 300°F provides a 60°F margin between the metal temperature of the vessel and the corresponding temperature of the RCS during an LTOP event. The staff agrees with the licensee that the current TS LTOP enable temperature of 300°F is conservative and therefore acceptable.

2.3 LTOP TS Changes

The licensee is proposing changes to the LTOP TS 3.4.9.3 to enhance overpressure protection during low temperature operation. The changes define additional conditions for invoking the LTOP system and the associated actions when these conditions are not met. By making these changes the licensee assures that operation and configuration of the units during low temperature operation are consistent with the LTOP event analysis.

The proposed TS changes, although consistent with NUREG-1431, have been justified for plant-specific use by the licensee. The changes, which establish the operability of the LTOP system, include:

(1) limiting the number of pumps capable of injecting into the RCS to one,

(2) isolating all the accumulators,

(3) a reduction in the RCS vent size,

(4) the action that if the PORV is inoperable, terminate any activities that could lead to a water-solid pressurizer within 24 hours, or use the RHR suction relief valve if the RCS temperature is >167°F, or depressurize and vent the RCS within 8 hours, and

(5) the action that if the LTOP is inoperable, depressurize and vent the RCS within 8 hours.

As noted above, the basis for all these changes is to enhance the overpressure protection during low temperature operation. Because these changes will lead to an acceptable depressurization process, the changes will ensure safer reactor operation.

The reduction in RCS vent size and the use of residual heat removal (RHR) suction relief valve for LTOP are discussed in greater detail in Sections 2.4 and 2.5 below. For the remaining proposed changes listed above, the licensee stated that these changes are more restrictive than the current LTOP TS.

The licensee also proposed changes to the TS surveillances that are additional to the current LTOP TS surveillances. The additional TS surveillances verify the accumulators have been isolated, the RHR suction isolation valves are open when the suction relief valves are being used for overpressure protection, and the block valve associated with the PORV providing LTOP protection is open. The staff agrees with the licensee's assessment that these changes are more restrictive and add more assurance that the depressurization process will take place when needed; therefore, the staff finds the changes acceptable.

2.4 Reduction of RCS Vent Size

In the event that a PORV is not available, the licensee is directed by its TS to make a vent available. Currently, that vent is specified as 4.5 square inches. This vent size appeared as overly conservative to the licensee. The licensee proposed to reduce the required vent size from 4.5 to 2.75 square inches. The licensee indicated that they completed an analysis, using NRC-approved methods, that verifies that the 2.75 square-inch vent is more than adequate for overpressure protection during an LTOP event. Also, by reducing the vent size in the TS, the licensee indicated that they are providing consistency in establishing the required vent based on the required relief flowrate. The licensee calculation of the vent size by NRC-approved methodology is acceptable.

2.5 RHR Suction Relief Valve

The main purpose of the RHR system is to remove decay heat during low temperature conditions. After the reactor coolant temperature and pressure have been reduced to approximately 350°F and 385 psig, the RHR system is placed into operation. While the RHR system is in service, the RHR suction relief valve is exposed to the RCS and is able to relieve RCS overpressure transients.

The licensee indicated that the current TS does not define how to use the RHR suction relief valve for overpressure protection during low temperature conditions. The proposed change to the TS defines the specific conditions under which the RHR suction relief valve can be used: (1) to enable a second emergency core cooling system (ECCS) pump to inject into the RCS; or (2) when one PORV is inoperable while in Modes 5 or 6, if the RCS temperature is greater than 167°F, and the RHR suction isolation valves are open. The first option, to use the RHR suction relief valve to enable a second ECCS pump to inject, is permitted at RCS temperatures below 167°F with additional restrictions: the RCS temperature must be greater than 107°F, the unit is being shut down, and the rate of the cooldown is less than 20°F per hour.

The licensee indicated that the capacity of the RHR suction relief valve is 902 gpm, which is adequate to relieve the full flow of either pump (565 gpm or 660 gpm) not both. Therefore, in situations where two pumps are capable of injecting into the RCS, both PORVs and the RHR suction relief valve are required to be operable.

When the RCS temperature is below 167°F and an LTOP event is mitigated by the RHR suction relief valve, the resultant peak pressure could exceed the allowable pressure for a cooldown rate of 100°F/hour. To avoid exceeding the 570 psig pressure limit on the 100°F/hour cooldown rate, the licensee proposes to restrict the use of the RHR suction relief valve below an RCS temperature of 167°F. The RHR suction relief valve can be used between RCS temperatures 107°F and 167°F provided the cooldown rate is limited to 20°F/hour or less. The associated allowable pressure is approximately 562 psig at 107°F. Since the resultant peak pressure of an LTOP event mitigated by the RHR suction relief valve, when the RCS temperature is below 107°F, could exceed the allowable pressure for a cooldown rate of 20°F/hour or less, the licensee proposes to prohibit the use of the RHR suction relief valve, as the means of LTOP, below an RCS temperature of 107°F. In this case, proposed TS 3.4.9.3 requires that two PORVs are secured in the open position for LTOP.

The staff has reviewed the licensee's intended use of the RHR suction relief valve and agrees that by defining how the RHR suction relief valve is to be used for LTOP, the licensee has ensured the integrity of the cooldown limits. The proposed TS changes also impose increased restrictions on the licensee and therefore the staff finds these changes acceptable.

2.6 Relocation of Instrument Error

Currently, the instrument uncertainty is included in the heatup and cooldown curves as 10°F and 60 psig margins. The proposed TS change is to move the instrument uncertainties to controlling procedures for unit operations and into the LTOP system setpoint selection calculations. The staff endorses the relocation of the heatup and cooldown curves from the TS, in their entirety, to controlled document, "Pressure Temperature Limit Report" (PTLR). The staff's endorsement is reflected in the new Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The licensee is proposing to relocate only the instrument uncertainties, while maintaining the curves in the TS. The uncertainties were developed by the NRC-approved Westinghouse Reactor Protection and Engineered Safeguards Setpoint Methodology and ISA SP67.15, Draft 10. The licensee has committed to incorporating the uncertainties into the Operating Procedures. Based on the licensee's conformance to the standards set forth in NUREG-1431 and their commitment to place the uncertainties in the Operating Procedures, the staff finds the proposed change to relocate the uncertainties to the PTLR acceptable.

The margins associated with the relocation of the uncertainties are 12°F and 30 psig. The licensee indicated that the increase in the temperature margins is for added assurance with no modification to the existing instrumentation. However, the reduction in the pressure margin from 60 psig to 30 psig, reflects the replacement of the wide range RCS transmitters with a narrow range pressure transmitter. The licensee used approved methods to verify the total instrument loop uncertainty for the RCS narrow range pressure instrumentation and its associated LTOP function. The worst case total loop uncertainty was calculated as ± 21 psig. The staff reviewed the calculation and found it acceptable; therefore, the use of 30 psig margin is acceptable.

2.7 Pressure-Temperature Limit Curves

The staff evaluates Pressure-Temperature (P-T) limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GL) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section III of the ASME Code. GL 88-11 informs licensees to use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation by calculating adjusted reference temperature (ART) of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{ndt}) of the material, the increase in RT_{ndt} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{ndt} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of RG 1.99, Rev. 2. If credible surveillance data is not available, the chemistry factor is calculated depending upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Rev. 2. GL 92-01 indicated that licensees should submit reactor vessel materials data, which the staff used in the review of the P-T limits submittals.

For the McGuire Unit 1 reactor vessel, the licensee determined two different limiting materials at the 1/4T and 3/4T locations. The licensee determined that the lower shell longitudinal weld material, 3-442 A&C, is the limiting material for the 1/4T location. Using integrated surveillance data, the licensee calculated an ART of 149.45°F at the 1/4T location at 16 effective full-power years (EFPY). The integrated surveillance data used for this material is based on the Diablo Canyon Unit 2 surveillance data. The staff approved use of Diablo Canyon Unit 2 surveillance data for McGuire Unit 1 in a letter to the licensee dated July 17, 1995. The licensee determined that the lower shell plate material, B5013-2, is the limiting material for the 3/4T location. Using the chemistry data of plate B5013-2 (the material was not included in the integrated surveillance program), the licensee calculated an ART of 102.03°F at the 3/4T location at 16 EFPY. The neutron fluence used in the ART calculation was 4.348 x 10^{18} n/cm^2 at the 1/4T location and 2.134 x 10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} values for weld 3-442 A&C and plate B5013-2 were -50°F and 30°F, respectively. The margin terms used in calculating the ART for weld 3-442 A&C and plate B5013-2 were 28°F and 34°F, respectively.

For the McGuire Unit 2 reactor vessel, the licensee determined that the lower shell forging 04 is the limiting material for both the 1/4T and 3/4T locations. Using the chemistry data of the forging (the material was not included in the surveillance program), the licensee calculated an ART of 104°F at the 1/4T location and 73°F at the 3/4T location at 16 EFPY. The neutron fluence used in the ART calculation was 6.138 x 10^{18} n/cm^2 at the 1/4T location and 2.222 x 10^{18} n/cm^2 at the 3/4T location. The initial RT_{NDT} and margin term were -30°F and 34°F, respectively.

For the McGuire Unit 2 reactor vessel, the licensee also calculated the ART values using surveillance data for the intermediate shell 05 and the intermediate/lower shell weld. The ART values calculated for both materials were less than the ART value of shell forging 04; therefore, the licensee concluded the shell forging 04 is limiting.

The staff verified for McGuire Units 1 and 2 that the copper and nickel content and initial RT_{NDT} agreed with the NRC reactor vessel material database as reported by the licensee in response to GL 92-01. The staff used the material properties to perform an independent calculation of the ART values for the limiting materials using RG 1.99, Revision 2. In addition, the staff used the surveillance data, as submitted in previous reports to the NRC, to perform an independent calculation of the ART values for the surveillance materials using Position 2 of RG 1.99, Revision 2. Based on the staff's calculation, the staff verified that the licensee's limiting material for McGuire Unit 1 is the lower shell longitudinal weld material, 3-442 A&C, at the 1/4T location and the lower shell plate material, B5013-2, at the 3/4T location. The staff also verified that the limiting material for McGuire Unit 2 is shell forging 04 at both the 1/4T and 3/4T locations. The staff's calculated ART values for the limiting materials agreed with the licensee's calculated ART values.

Substituting the ART values for McGuire Units 1 and 2 into equations in SRP 5.3.2, the staff verified that the proposed P-T limits for heatup, cooldown, criticality, and inservice hydrostatic test satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{ndt} of 40°F for Unit 1 and 1°F for Unit 2, provided by the licensee, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test, and leak test.

3.0 STAFF CONCLUSION

The staff's review of the licensee's proposed changes to the LTOP TS 3.4.9.3, the LTOP setpoint, the LTOP enable temperature, the reduction in vent size, the use of the RHR suction relief valve for LTOP mitigation, and the relocation of the instrument error finds these changes acceptable because they (1) have been analyzed by approved methods, (2) are more restrictive than the current TS, and (3) conform to NUREG-1431, Westinghouse Standard Technical Specifications.

The staff has performed an independent analysis to verify the licensee's proposed P-T limits. The staff concludes that the proposed P-T limits for heatup, cooldown, inservice hydrostatic test, and criticality are valid for 16 EFPY because: 1) the limits meet the requirements of Appendix G of 10 CFR Part 50 and conform to GL 88-11; 2) the material properties and chemistry used in calculating the P-T limits are consistent with data submitted under GL 92-01; and 3) the surveillance data used in calculating the P-T limits are consistent with data in surveillance reports submitted to the staff. Therefore, the proposed P-T limits may be incorporated in the McGuire Units 1 and 2 TS. In addition, the proposed editorial changes in the Bases section of the TS are consistent with the P-T limits changes; therefore, they are acceptable.

4.0 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.

5.0 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 effluents 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 (60 FR 49933 dated September 27, 1995). 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.

6.0 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.

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Date: January 11, 1996