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.

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" Tecnnical 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

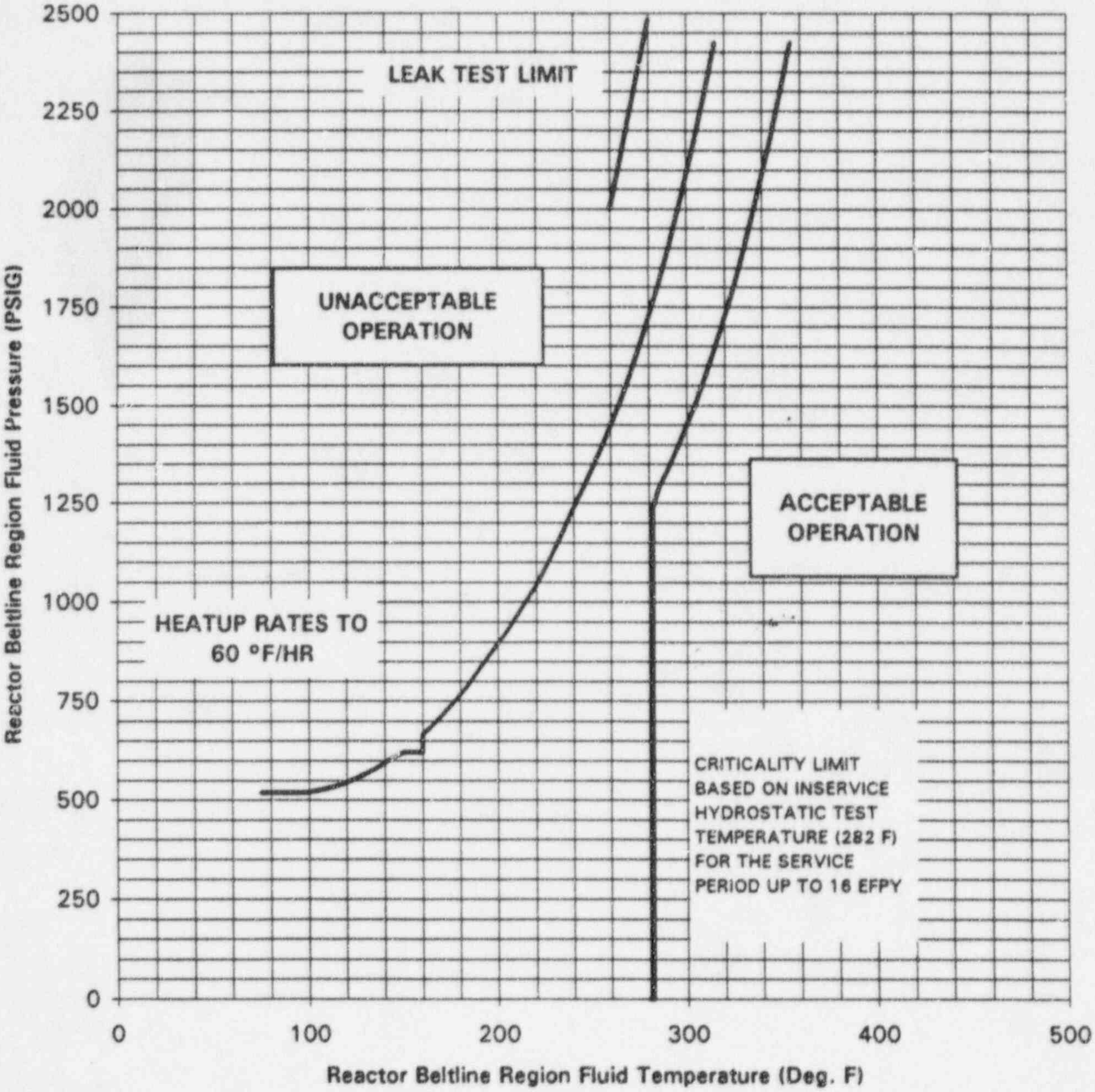
LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

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

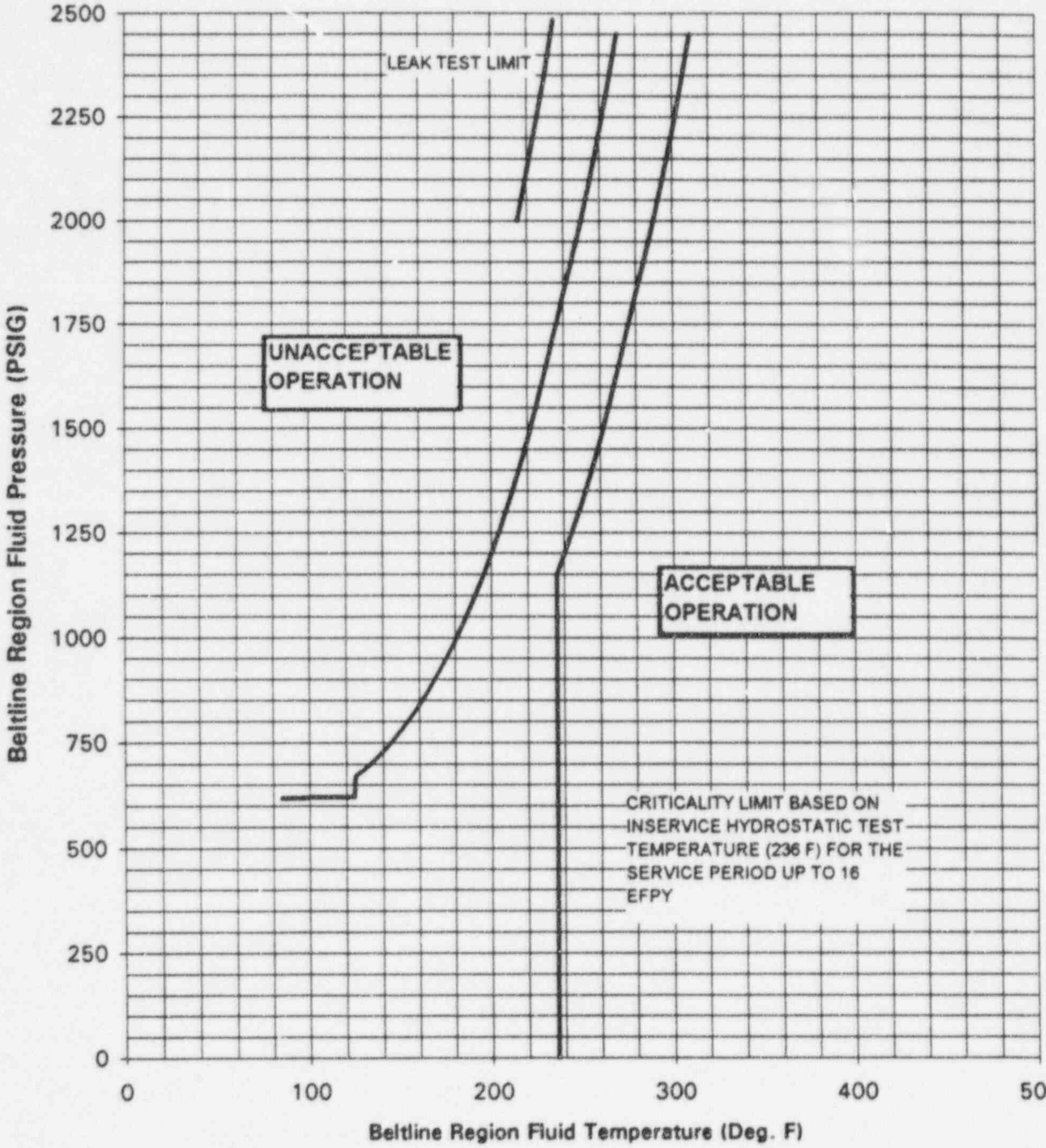


McGuire Unit 1 Reactor Coolant System Heatup Limitations
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the first 16 EFPY

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

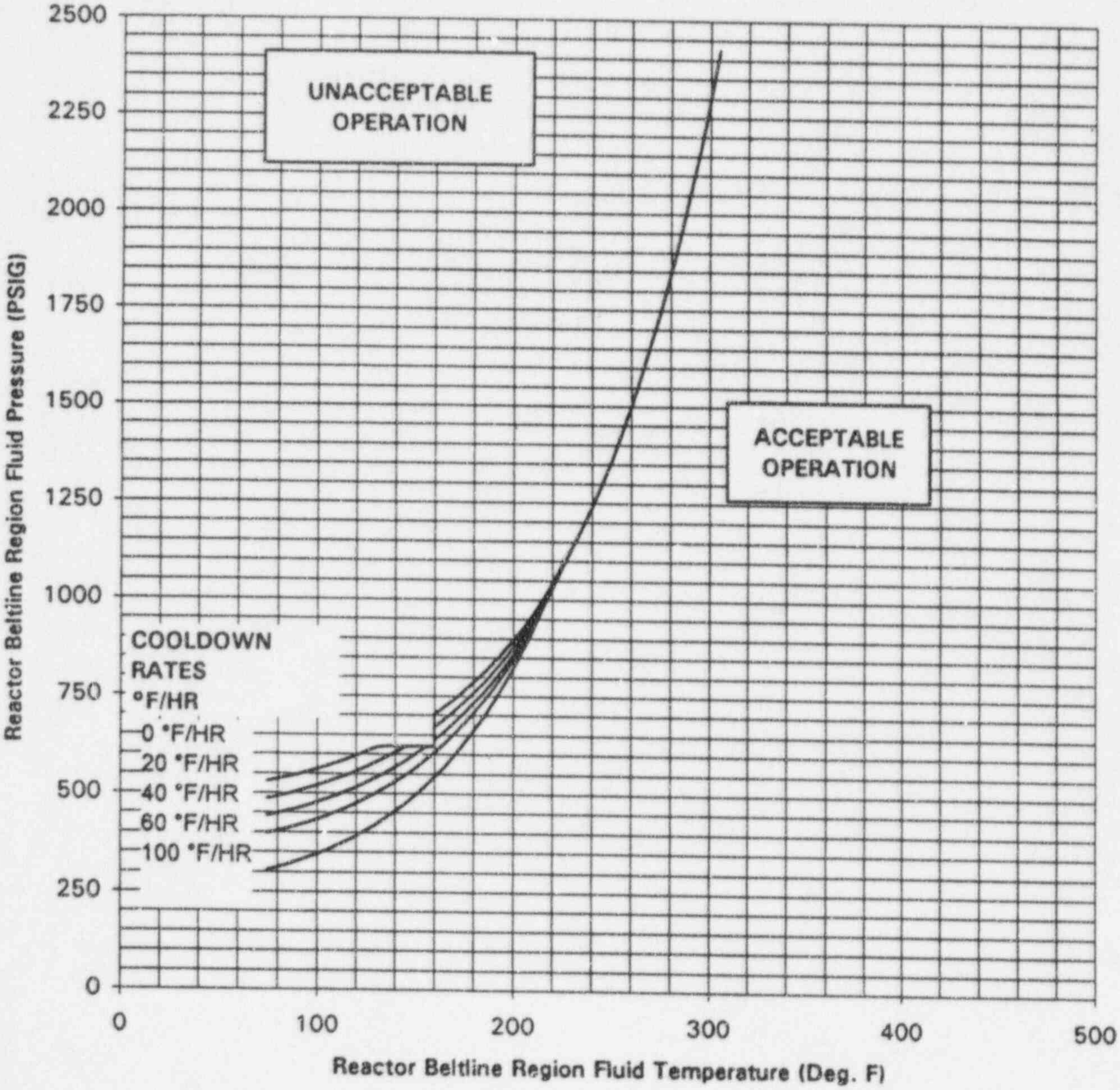
Figure 3.4-3

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

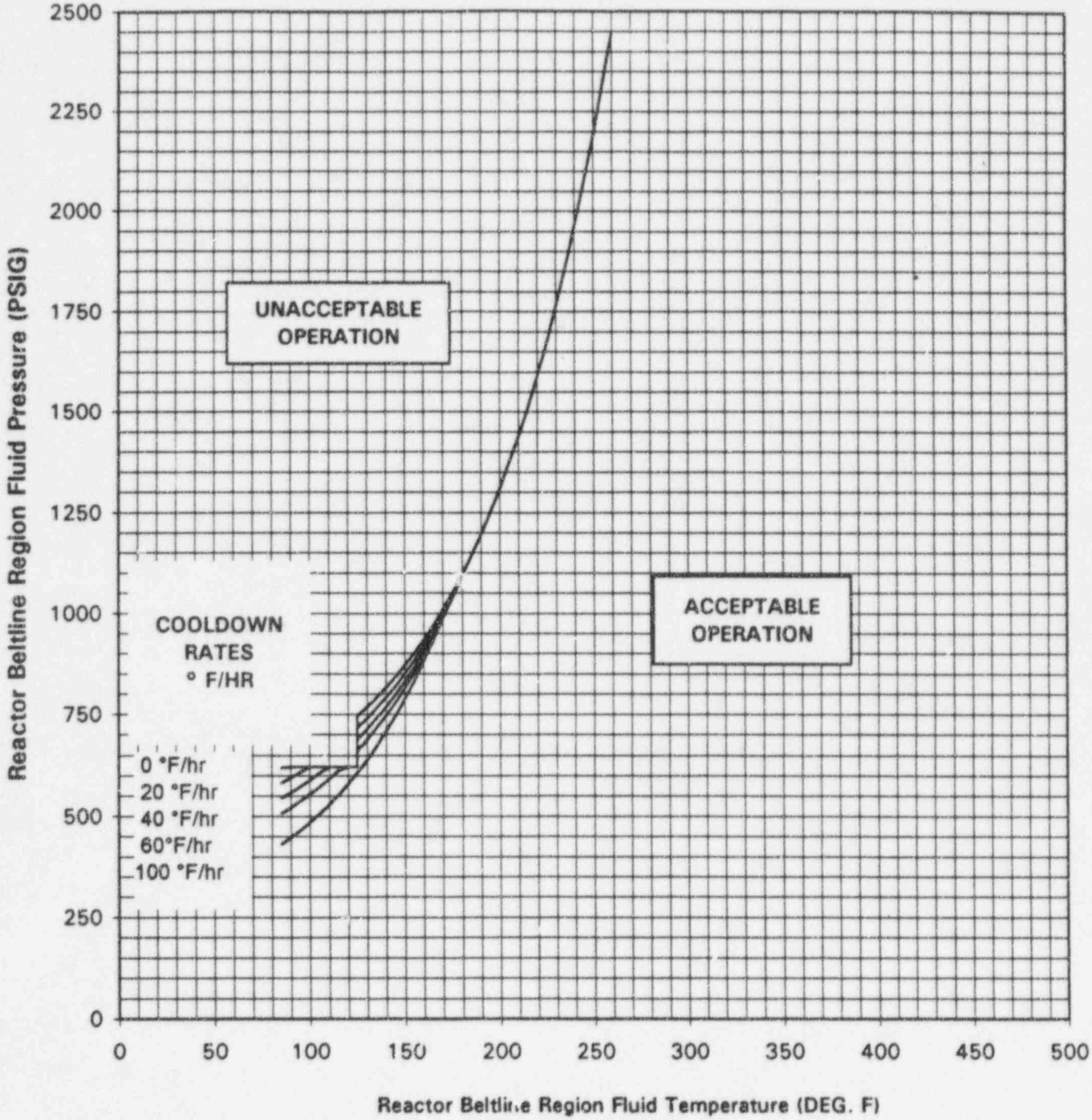


McGuire Unit 1 RCS Cooldown Limitations,
Cooldown Rates up to 100 deg. F/HR
(Without Margins for Instrumenation Errors)
NRC REG GUIDE 1.99, REv. 2
Applicable for the First 16 EFPY

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.