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A Duke Energy Company

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December 12, 2002

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: McGuire Nuclear Station, Units 1 and 2
Docket Nos. 50-369 and 50-370
License Amendment Request (LAR)
Technical Specifications (TS) Changes
TS 3.3.2 - Engineered Safety Feature Actuation System (ESFAS) Instrumentation
TS 3.4.3 - Reactor Coolant System (RCS) Pressure and Temperature (P-T) Limits
TS 3.4.12 - Low Temperature Overpressure Protection (LTOP) System

By letter dated February 14, 2001, as supplemented by letters dated April 4, 2001 and May 30, 2001, Duke submitted a LAR for changes to the subject TSs. By letter dated August 1, 2001, the NRC transmitted a Request for Additional Information (RAI) to Duke. By letter dated January 28, 2002, Duke withdrew that LAR due to pending NRC review of WCAP-15315, Revision 1. This WCAP proposed the elimination of Reactor Pressure Vessel (RPV) flange requirements of 10 CFR 50 Appendix G.

Pursuant to 10 CFR 50.90, Duke Energy (Duke) hereby resubmits the LAR to revise the following: TS Table 3.3.2-1 Footnote (c) is revised to correct an editorial error in the description of Steam Line Isolation on Steam Line Pressure Negative Rate - High blocking. TS 3.4.3 is revised to update the RCS P-T limits for use up to 34 Effective Full Power Years (EFPY). TS 3.4.12 is revised to update the LTOP limits for use up to 34 EFPY.

Pursuant to 10 CFR 50.60, Duke hereby requests an exemption from the requirements of 10 CFR 50 Appendix G to allow application of American Society of Mechanical Engineers (ASME) Code Case N-641 in the development of P-T and LTOP limits.

The difference between the February 14, 2001 LAR and this LAR includes: the application of the full 10 CFR 50 Appendix G RPV flange temperature requirement in the P-T limits, use of the fourth Diablo Canyon Unit 2 RPV capsule surveillance data for McGuire Unit 1, use of calculated neutron fluence projection instead of best estimate fluence projection, change in the initial RT_{NDT} of the RPV and closure head flanges (for Unit 1 only), and application of ASME Code Case N-641 in the

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determination of P-T limits and LTOP setpoints. The enclosed technical analysis includes the information requested by the RAI. Duke concludes that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c).

Duke requests approval of the LAR no later than May 30, 2003 to allow for implementation before Unit 1 P-T limits expire in Mid-August of 2003 and before Unit 2 end of cycle 15 refueling outage starts in mid-September of 2003. A significant number of procedure changes is required to support implementation; therefore, Duke requests that once approved, the LAR shall be implemented within 90 days.

Upon NRC approval of this LAR, Duke commits to implement procedure changes to incorporate the maximum allowable heatup and cooldown rates for RCS pump operating conditions, as specified in Enclosure 2, Tables 3 and 6. Revision to the Updated Final Safety Analysis Report (UFSAR) as a result of this LAR will be made in accordance with the requirements of 10 CFR 50.71(e).

The McGuire Plant Operations Review Committee and the Corporate Nuclear Safety Review Board have reviewed and approved this LAR. Pursuant to 10 CFR 50.91, a copy of this LAR is being forwarded to the North Carolina Division of Radiation Protection.

If there are any questions regarding this LAR, please contact P. T. Vu at (704) 875-4302.

Very truly yours,

D. M. Jamil

Enclosures:

1. Notarized Affidavit
2. Licensee's Evaluation of the Proposed Changes
3. Exemption Request for use of ASME Code Case N-641
4. Westinghouse WCAP-15192, Revision 2
5. Westinghouse WCAP-15201, Revision 2
6. ASME Code Case N-641

U.S. Nuclear Regulatory Commission
December 12, 2002

Attachments:

1. Proposed Technical Specification Changes (mark-up)
2. Changes to TS Bases Pages (mark-up)
3. Proposed Technical Specification Pages (retyped)
4. TS Bases Pages (retyped)
5. NUREG-1431 mark-up page in May 27, 1997 LAR

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ENCLOSURE 1

NOTARIZED AFFIDAVIT

U.S. Nuclear Regulatory Commission
December 12, 2002

Enclosure 1

AFFIDAVIT

D. M. Jamil, being duly sworn, states that he is Vice President of Duke Energy Corporation; that he is authorized on the part of said corporation to sign and file with the Nuclear Regulatory Commission this amendment to the McGuire Nuclear Station Facility Operating Licenses Numbers NPF-9 and NPF-17 and Technical Specifications; and that all statements and matters set forth herein are true and correct to the best of his knowledge.

D. M. Jamil, Vice President

Subscribed and sworn to me: December 12, 2002
Date

Notary Public: Deborah S. Rome
Deborah S. Rome

My Commission Expires: December 19, 2004
Date

SEAL

ENCLOSURE 2

LICENSEE'S EVALUATION OF THE PROPOSED CHANGES

LICENSEE'S EVALUATION

SUBJECT: Proposed Changes to Technical Specifications (TS). TS 3.4.3, Reactor Coolant System (RCS) Pressure and Temperature (P-T) Limits, TS 3.4.12, Low Temperature Overpressure Protection (LTOP) System, and TS 3.3.2, Engineered Safety Features Actuation System (ESFAS) Instrumentation.

1. DESCRIPTION

2. PROPOSED CHANGE

3. BACKGROUND

4. TECHNICAL ANALYSIS

5. REGULATORY SAFETY ANALYSIS

   5.1 No Significant Hazards Consideration

   5.2 Applicable Regulatory Requirements/Criteria

6. ENVIRONMENTAL CONSIDERATION

7. REFERENCES

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses Numbers NPF-9 and NPF-17 for McGuire Nuclear Station Units 1 and 2.

The proposed changes would revise TS 3.4.3 and TS 3.4.12 to update the RCS P-T and LTOP limits for use up to 34 EFPY, and to correct an editorial error in TS 3.3.2. The proposed changes are necessary because the current P-T and LTOP limits will expire in Mid-August of 2003 for Unit 1 and January of 2004 for Unit 2.

2.0 PROPOSED CHANGES

TS Table 3.3.2-1 Footnote (c) is revised to specify that the Steam Line Isolation on Steam Line Pressure Negative Rate - High function may be blocked below P-11 when Steam Line Isolation on Steam Line Pressure - Low is not blocked. This footnote currently incorrectly specifies that this function may be blocked below P-11 when Safety Injection Steam Line Pressure - Low is not blocked.

TS 3.4.3 is revised to update the P-T limits for use up to a maximum of 34 EFPY for Units 1 and 2. The current P-T limits are effective only to 16 EFPY. Figure 3.4.3-1 currently has two heatup figures, one for each unit. Figure 3.4.3-2 currently has two cooldown figures, one for each unit. Figure 3.4.3-1 will be replaced with four heatup figures (Figures 3.4.3-1 through 3.4.3-4), two for each unit. Figure 3.4.3-2 will be replaced with two cooldown figures (Figures 3.4.3-5 and 3.4.3-6), one for each unit. Limiting condition for operation (LCO) 3.4.3 is revised to refer to these new figures.

TS 3.4.12, Required Actions A.2.2.1 and A.2.2.2 are revised to reflect the temperature and/or cooldown rate limits associated with the use of the residual heat removal (RHR) suction relief valve when any combination of two centrifugal charging and safety injection pumps are capable of injecting into the RCS. TS 3.4.12, Required Action F.1 is revised to reflect the temperature limit associated with the use of the RHR suction relief valve when the required action and associated completion time for one inoperable power operated relief valve (PORV) (Condition E) are not met.

Two new Required Actions (A.5.1 and A.5.2) are proposed for TS 3.4.12. The combination of these two new required actions, a RCS vent of greater than or equal to 2.75 square inches and two operable PORV, is proposed for the condition when any combination of two centrifugal charging and safety injection pumps are capable of injecting into the RCS.

In summary, the proposed amendments revise TS 3.4.3 and TS 3.4.12 to update P-T and LTOP limits for use up to a maximum of 34 EFPY, and to correct an editorial error in TS 3.3.2.

TS 3.3.2 Bases is revised to reflect the change for TS 3.3.2 described above. TS 3.4.3 Bases is revised to include a summary description of the new excore cavity dosimetry program at McGuire. The new program employs excore cavity dosimetry to determine the reactor vessel neutron fluence through calculation-based fluence determination. The new program meets the requirements of 10 CFR 50 Appendix H and guidance of Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence, March 2001."

TS 3.4.12 Bases is revised to reflect the changes for TS 3.4.12 described above, and to clarify that a secured open PORV means that it is physically secured or locked open to prevent it from being subject to active failure. This clarification is consistent with that for a RCS vent already described in the same Bases.

3.0 BACKGROUND

RCS P-T Limits

RCS P-T limits are established to provide a margin to brittle failure of the reactor vessel and piping of the RCS pressure boundary. The vessel is the component most subject to brittle failure, and the P-T limits apply mainly to the vessel. 10 CFR 50 Appendix G requires the establishment of P-T limits for specific material fracture toughness requirements of the RCS pressure boundary materials. This rule requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of ASME Section XI Appendix G.

The neutron embrittlement effect on the material toughness is reflected by an increase in the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases. The actual shift in the RT_{NDT} of the vessel material is established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E-185 and Appendix H of 10 CFR 50. The operating P-T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 Revision 2. The normal use of P-T limits is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable P-T curve to determine that operation is within the allowable region.

McGuire Units 1 and 2 current P-T limits are effective only up to 16 EFPY. The development of the proposed P-T limits relies, in part, on an alternate methodology allowed by ASME Code Case N-641. Westinghouse performed reactor vessel integrity assessments and generated new P-T limits for McGuire Units 1 and 2. The new P-T limits are effective up to 34 EFPY for both units. Westinghouse Topical Reports WCAP-15192, Revision 2 and WCAP-

15201, Revision 2 provide the details regarding the determination of the new P-T limits.

10 CFR 50 Appendix G Table 1 specifies that the metal temperature of the flange region must exceed the material unirradiated RT_{NDT} by at least 120 °F for normal operation when the pressure exceeds 20 percent of the pre-service hydrostatic test pressure, which is 621 psig for a typical PWR. This requirement was originally based on concerns about the fracture margin in the flange region. During the boltup process, outside surface stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120 °F and the pressure limitation of 20 percent of pre-service hydrostatic test pressure were developed using the K_{IA} fracture toughness curve, in the mid 1970s, to ensure that appropriate margins would be maintained.

The requirements for P-T limits incorporate nine numbers of safety margins, one of which is the lower bound fracture toughness curve. There are two lower bound fracture toughness curves available in ASME Section XI, K_{IA}, which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC}, which is a lower bound on static fracture toughness only. ASME Code Case N-641 permits the use of K_{IC} fracture toughness curve instead of K_{IA} fracture toughness curve in development of P-T limits. The other margins involved with the process remain unchanged.

The NRC has approved the use of ASME Code Case N-641 in LARs for several nuclear power stations. Below are the licensee application dates and NRC approval dates for these LARs:

Plant Name	Licensee Application Date	NRC Approval Date
Point Beach 1/2	7/14/00	10/6/00
Turkey Point 3/4	7/7/00	10/24/00
North Anna 1/2	6/22/00	5/2/01
Arkansas Nuclear One 2	10/30/01	4/15/02

LTOP System

The LTOP system controls RCS pressure at low temperatures so that integrity of the RCS pressure boundary is not compromised by violating the P-T limits of 10 CFR 50 Appendix G. The reactor vessel is the most limiting component within the RCS pressure boundary for demonstrating such protection. Technical Specifications for LTOP provide the maximum allowable actuation logic setpoints for the PORV. Technical Specifications for P-T limits provide the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the requirements of 10 CFR 50 Appendix G during the LTOP mode. The

reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased. The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown. Exceeding the RCS P-T limits by a significant amount could cause brittle cracking of the reactor vessel. Once the LTOP system is enabled, no operator action is involved for the system to perform its intended pressure mitigation function. Each time the P-T limits are revised, the LTOP system must be re-evaluated to ensure its functional requirement can still be met. LTOP calculation provides LTOP setpoints and heatup and cooldown restrictions. LTOP setpoint determination is consistent with 10 CFR 50 Appendix G and ASME Code Case N-641.

Reference to Updated Final Safety Analysis Report

UFSAR Section 5.2.2 describes the overpressurization protection for the RCS. Section 5.2.4 describes the requirements for RCS pressure boundary fracture toughness. Section 5.4.3.7 describes the requirements for RPV material surveillance program.

Prior Correspondence

By letter dated February 14, 2001, as supplemented by letters dated April 4, 2001 and May 30, 2001, Duke submitted a LAR for changes to the subject TSs. By letter dated August 1, 2001, the NRC transmitted a Request for Additional Information (RAI) to Duke. By letter dated January 28, 2002, Duke withdrew that LAR due to pending NRC review of WCAP-15315, Revision 1. This WCAP proposed the elimination of Reactor Pressure Vessel (RPV) flange requirements of 10 CFR 50 Appendix G.

The difference between that LAR and this LAR includes: the application of the full 10 CFR 50 Appendix G RPV flange temperature requirement in the P-T limits, use of the fourth Diablo Canyon Unit 2 RPV capsule surveillance data for McGuire Unit 1, use of calculated neutron fluence projection instead of best estimate fluence projection, change in the initial RT_{NDT} of the RPV and closure head flanges (for Unit 1 only), and application of ASME Code Case N-641 in the determination of P-T limits and LTOP setpoints. The technical analysis section below includes the information requested by the RAI.

Effect on Current P-T limits

The effect of ASME Code Case N-641 and Diablo Canyon's RPV capsule surveillance data on McGuire's current P-T limits has been evaluated. The current P-T limits bound the proposed P-T limits for all analyzed heatup and cooldown conditions;

therefore, they are valid for continued operation of the plant to 16 EFPY. Duke's justification for this conclusion is documented in a site specific calculation (Reference 7).

4.0 TECHNICAL ANALYSIS

Determination of Adjusted RT_{NDT} (ART)

The 34 EFPY ART values at the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations for the beltline regions of the McGuire reactor vessels were calculated by Westinghouse. These calculations were in accordance with Regulatory Guide 1.99, Revision 2. Regulatory Guide 1.99, Revision 2 credibility criteria are applied by Westinghouse to determine the appropriate margin term. The calculations determined the ART for the various reactor vessel (RV) materials using Regulatory Guide 1.99, Revision 2, Regulatory Positions 1.1 and 2.1. The selected controlling values are those RV locations with the highest ART for 1/4T and 3/4T whether determined using Regulatory Position 1.1 or 2.1 methodology. Surveillance capsule weld data obtained from Diablo Canyon Unit 2 are used in the determination of the limiting ART value for McGuire Unit 1. The first of four weld data points from Diablo Canyon falls outside the scatter band of +/- 28 °F. Regulatory Guide 1.99, Revision 2 treats this data as non-credible; however, the data is very predictable when analyzed using more recent embrittlement correlations, such as the recently approved ASTM E-900 correlation. Thus, the Diablo Canyon Unit 2 surveillance weld data described above are utilized in the determination of the limiting ART for McGuire Unit 1, with conservatism added by including a full margin of 56 °F.

The calculation of the ART values for the 1/4T and 3/4T locations at 34 EFPY is presented in Tables 8 and 9 of WCAP-15192, Revision 2 and WCAP-15201, Revision 2. The limiting ARTs used in the generation of the P-T curves are summarized below:

Summary of the Limiting ART Values Used in the Generation of the McGuire Unit 1 Heatup/Cooldown Curves

Limiting Material	RG 1.99 R2 Method	¼ T Limiting ART	¾ T Limiting ART
Lower Shell Longitudinal Weld Seams 3-442A & C	Position 2.1 but with full margin added	202	146

Summary of the Limiting ART Values Used in the Generation of the McGuire Unit 2 Heatup/Cooldown Curves

Limiting Material	RG 1.99 R2 Method	¼ T Limiting ART	¾ T Limiting ART
Lower Shell Forging 04	Position 1.1	123	91

Determination of P-T Limits

The proposed P-T limits are developed using the methods and criteria described in NRC-approved Topical Report WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", with exception of the following:

1) The fluence values are calculated fluence values (i.e., the values are derived to comply with Regulatory Guide 1.190), and are not the best estimate fluence values.

2) The K_{IC} fracture toughness curve is used in place of the K_{IA} fracture toughness curve (ASME Code Case N-641).

3) The 1996 version of Appendix G to ASME Section XI is used rather than the 1989 version.

The P-T limits and supporting technical analysis are included in WCAP-15192, Revision 2 and WCAP-15201, Revision 2.

The limiting unirradiated RT_{NDT} of 10 °F for the Unit 1 RPV and closure head flanges results in the minimum allowable temperature of 130 °F for this region at pressures greater than 621 psig. It should be noted that the Charpy V-notch tests for Unit 1 RPV and closure head flanges were re-evaluated (Appendix A of Reference 1) using the Material Certifications from Bethlehem Steel Corporation. This re-evaluation concluded that the initial RT_{NDT} values for the RPV and closure head flanges should be 10 °F. The limiting unirradiated RT_{NDT} of 1 °F for Unit 2 flange region results in the minimum allowable temperature of 121 °F for this region at pressures greater than 621 psig. These limits are included in the P-T curves proposed in this LAR.

Determination of LTOP Limits

In general, the methodology presented in Westinghouse technical report, "Pressure Mitigation Systems Transient Analysis Results," dated July 1977 and its supplement dated September 1977 is used since this is the current NRC approved licensing basis for McGuire's LTOP system. During Catawba Nuclear Station's most recent LAR for LTOP, the NRC required more advanced methodology be applied to the analysis of the heat input transient due to the recent change in steam generators. A similar analysis is performed for McGuire. The RETRAN computer code is used to model the response of the RCS to the LTOP heat input transient. The RCS model is described in Duke's NRC-approved Topical Report DPC-NE-3000-PA, "Thermal-Hydraulic Transient Analysis Methodology." This topical report is listed in TS 5.6.5.

For operation at low temperatures, three bounding transients are analyzed which will cause an increase in internal vessel pressure. These are a mass input from a safety injection pump, a

mass input from a centrifugal charging pump, and a heat input from reactor coolant pump (RCP) startup with temperature asymmetry between the RCS and steam generators. These three transients are evaluated for 2 second PORV opening times to determine a peak pressure equal to setpoint plus overshoot. The pressure overshoot for a mass input transient is a function of the mass input rate, RCS volume, PORV stroke time and an additional factor that accounts for relief capacity as a function of set pressure.

The peak transient pressures are adjusted for static pressure effect, dynamic pressure effect and instrument uncertainty. Duke Energy has an engineering directive that outlines the requirements for performing instrument uncertainty and setpoint calculations. The primary purpose of this directive is to provide a consistent methodology based on standard industry practices for performing instrument uncertainty and setpoint calculations. The calculation methodology is consistent with the intent of ISA-67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The methodology conservatively accounts for the typical uncertainty terms such as reference accuracy, drift, temperature effect, and calibration effects (which include measurement and test equipment uncertainty, calibration tolerance and resolution). The random-independent uncertainty terms are combined via the "square-root-sum-of-the-square" (SRSS) technique, whereas the random-dependent and bias uncertainty terms are combined via a combination of SRSS and/or algebraic techniques. The uncertainty calculation performed for the RCS resulted in +/- 12 °F (computer point), +/- 14 °F (chart recorder) and +/- 30 psid (LTOP pressure).

The peak transient pressures are compared to the Appendix G limits, and the heatup and cooldown restrictions are developed to keep the peak pressure for each transient below the Appendix G P-T limits. The PORV nominal setpoint and tolerance are included in the calculation of the peak transient pressures. The PORV actuation setpoint for LTOP is high enough to allow RCP startup and normal operations but low enough to assure adequate protection against system pressurization above the Appendix G limits.

The temperature at which the LTOP protection system must be placed in service is determined using the methodology of ASME Code Case N-641.

Additional requirements must be established for conditions where two centrifugal charging pumps, two safety injection pumps, or one centrifugal charging pump and one safety injection pump are capable of mass addition. These conditions are evaluated and all specific requirements are developed.

LTOP Limits - Unit 1

Tables 1 and 2 below summarize the Appendix G limits, and peak vessel pressures for the mass input and heat input transients, respectively. The peak transient pressure is the sum of the nominal PORV setpoint, pressure overshoot, dynamic pressure effect, static pressure effect, and instrument error. A 380 psig PORV nominal setpoint and 30 psid instrument error are used in the calculation of the peak transient pressures. The 385 psig PORV lift setpoint in TS 3.4.12 is the maximum allowable value. The difference between this TS value and the nominal value is bounded by the 30 psid instrument error. This allowable value is verified to be adequate for LTOP protection by comparing the peak transient pressures to the Appendix G limits and establishing the maximum heatup and cooldown rates to ensure the peak transient pressures do not exceed the Appendix G limits. The 385 psig allowable value is adequate for LTOP protection provided the maximum heatup and cooldown rates in Table 3 are followed.

Table 1: Unit 1 Appendix G Limits

RV Beltline Region Temperature (°F)		RV Beltline Region Pressure Limit per WCAP -15192, Rev. 2 (psig)					
Actual	Indicated (w / 12°F Margin)	Steady state	Cooldown @ 20 F/Hr	Cooldown @ 40 F/hr	Cooldown @ 60F/hr	Cooldown @ 100F/hr	Heatup @ 100F/hr
60	72	0	0	0	0	0	0
60	72	621	579	528	476	367	579
65	77	621	582	530	478	370	579
70	82	621	584	533	481	373	579
75	87	621	587	536	484	376	579
80	92	621	590	539	488	380	579
85	97	621	594	543	492	385	579
90	102	621	598	547	496	390	579
95	107	621	602	552	501	396	579
100	112	621	607	557	507	403	579
105	117	621	613	563	513	410	579
110	122	621	619	570	520	419	579
115	127	621	621	577	528	428	579
120	132	621	621	585	537	439	580
125	137	621	621	594	547	450	583
130	142	621	621	604	557	464	587
135	147	706	661	615	570	478	593
140	152	717	672	628	583	495	600
145	157	728	685	641	598	513	609
150	162	741	699	657	615	534	620
155	167	755	714	674	634	557	633
160	172	771	731	692	654	582	647
165	177	788	750	713	677	611	664
170	182	807	771	736	703	642	683
175	187	828	794	762	731	677	704

Table 2: Unit 1 Peak Transient Pressures

Transient Description	Nominal PORV Setpoint (psig)	Pressure Overshoot (psi)	Dynamic Pressure Effects (psi)	Static Pressure Effects (psi)	Instrument Error (psi)	Peak Pressure (psig)
Safety Injection with....						
0 RCP's in Operation	380	***51.3***	***0.1***	4.7	30	**466.1**
1-2 RCP's in Operation	380	***51.3***	***15.9***	4.7	30	**481.9**
3 RCP's in Operation	380	***51.3***	***34.0***	4.7	30	**500.0**
4 RCP's in Operation	380	***51.3***	***58.9***	4.7	30	**524.9**
Charging/Letdown Mismatch with....						
0 RCP's in Operation	380	***45.6***	***0.1***	4.7	30	***460.4***
1-2 RCP's in Operation	380	***45.6***	***15.9***	4.7	30	***476.2***
3 RCP's in Operation	380	***45.6***	***34.0***	4.7	30	***494.3***
4 RCP's in Operation	380	***45.6***	***58.9***	4.7	30	***519.2***
Heat Input (Primary - Secondary Temp Mismatch) with...						
0 RCP's in Operation **AND**						
Tave of 60 - 100°F	***From Reference 6***					***468.3***
Tave to 100 - 180°F						***522.5***
Tave to 180 - 250°F						***514.9***
1-4 RCP's in Operation	***Transient not credible with an NC pump initially running.***					-

Table 3: Unit 1 Heatup and Cooldown Restriction

Number of RCP's Running	RCS Temperature Range			
	Without Margin	OAC Points	Chart Recorder	Heat-up Limit
0 to 4	> 60°F	> 72°F	> 74°F	100 °F / hr
				Cooldown Limit
0	> 150°F	> 162	> 164	100 °F / hr
	150 - 115	162 - 127	164 - 129	60 °F / hr
	115 - 60	127 - 72	129 - 74	40 °F / hr
1 - 2	> 140°F	> 152	> 154	100 °F / hr
	140 – 75	152 – 87	154 – 89	60 °F / hr
	75 – 60	87- 72	89 – 74	40 °F / hr
3	> 145°F	> 157	> 159	100 °F / hr
	145 - 95	157 - 107	159 - 109	60 °F / hr
	95 - 60	107 - 72	109 - 74	40 °F / hr
4	> 150°F	> 162	> 164	100 °F / hr
	150 - 115	162 - 127	164 - 129	60 °F / hr
	115 - 60	127 - 72	129 - 74	40 °F / hr

From the previous LTOP LAR (Reference 12), the technical justification for verifying that the enable temperature is adequate to provide overpressure protection states that the licensed basis for the enable temperature ensures that the Appendix G limits are protected by either the PORV setpoint or the pressurizer safety valves (PSV). The PSVs are required to relieve pressure between 2435 - 2559 psig. The corresponding Appendix G steady state pressure limit is 2,240 psig at 275 °F and 2,412 psig at 280 °F (Reference 1). Linear interpolation of the P-T limits for a nominal PSV lift setpoint of 2485 psig results in a corresponding temperature limit of 282 °F. Using this methodology, there is no interruption of overpressure protection if the enable temperature is set at 282 °F (294 °F including instrument error) or above.

The LTOP enable temperature T_e is the temperature at or above which the safety relief valves provide adequate protection against non-ductile failure. ASME Code Case N-641 specifies that LTOP system shall be effective below the higher temperature determined in accordance with (1) a coolant temperature of 200F and (2) a coolant temperature corresponding to a reactor vessel metal temperature, for all vessel beltline materials, where T_e is calculated on a plant specific basis for the axial or circumferential reference flaws using the following equation:

$$T_e = RT_{NDT} + 50 \text{ Ln } [((F.M_m(pR_i/t)) - 33.2)/20.734]$$

where, for McGuire Unit 1:

RT_{NDT} = 202 °F

F = 1.1, accumulation factor for safety relief valves

M_m = the value of M_m determined in accordance with G-2214.1. As provided in WCAP-15192, Revision 2, Section 3.2, M_m = 0.926(Square Root of t) = 2.72

p = vessel design pressure, 2.485 ksi

R_i = vessel inner radius, 86.5 in.

t = vessel wall thickness, 8.625 in.

Instrument error of 12 °F and a maximum temperature lag of 30.1 °F between the 1/4 T reactor vessel location and coolant temperature during 100 °F/hr heatup are added to the above equation

Thus, T_e for Unit 1 = 202 + 34.5 + 12 + 30.1 = 278.6 °F

The current McGuire Unit 1 LTOP enable temperature of 300 °F bounds each of the methods discussed above, and is acceptable for operation up to 34 EFPY.

It is desirable to operate with one centrifugal charging pump and one safety injection pump in service for brief periods during plant heatup (i.e., for accumulator fill and check valve testing). This has shown to be acceptable provided the RHR suction relief valve is available to provide additional relief capacity.

The peak vessel pressure with the RHR suction relief at maximum relieving capacity is approximately 560 psig, which includes static and dynamic pressure effects. The capacity of the RHR suction relief valve (902 gpm) alone is adequate to relieve the full flow of either the centrifugal charging pump (runout limit = 560 gpm), the safety injection pump (runout limit = 675 gpm), two centrifugal charging pumps, or two safety injection pumps. Therefore, RHR relief valve in conjunction with one assured PORV is adequate to relieve the combined flow of any combination of two centrifugal charging and safety injection pumps. The single failure criterion is satisfied in that any two of the three relief valves will be adequate for LTOP. The peak pressure considering the RHR relief valve (560 psig) is somewhat higher than the PORV's (524.9 psig for 2.0 second stroke time); therefore, tighter restrictions must be placed on when the second pump may be made operational.

Additional restrictions for use of RHR suction relief valve are provided in TS 3.4.12 Required Actions A.2.2.1, A.2.2.2, and F.1:

1. From Table 1, the Appendix G limits for steady-state condition and 100 °F/hr heatup rate are 621 psig and 579 psig, respectively (at 60 °F). Therefore, the RHR suction relief valve is adequate for all steady-state and heatup conditions.

2. From Table 1, the Appendix G limit for a 100 °F/hr cooldown is 582 psig at 160 °F (172 °F - 174 °F with instrument uncertainties). Therefore, 100 °F/hr cooldown rate must not be used below 174 °F indicated temperature.

3. From Table 1, the Appendix G limit for a cooldown rate of 20 °F/hr is 579 psig at 60 °F (72 °F - 74 °F with instrument uncertainties). Therefore, cooldown rates of 20 °F/hr or less must be used between 174 °F and 74 °F indicated temperature.

LTOP Limits - Unit 2

Tables 4 and 5 below summarize the Appendix G limits, and peak vessel pressures for the mass input and heat input transients for Unit 2, respectively. Similar to Unit 1, the 385 psig allowable value is adequate for LTOP protection provided the maximum heatup and cooldown rates in Table 6 are followed.

Table 4: Unit 2 Appendix G Limits

RV Beltline Region Temperature (°F)		RV Beltline Region Pressure Limit per WCAP -15201 (psig)						
Actual	Indicated (w / 12°F Margin)	Steady state	Cooldown @ 20 F/Hr	Cooldown @ 40 F/hr	Cooldown @ 60F/hr	Cooldown @ 100F/hr	Heatup @ 60F/hr	Heatup @ 100F/hr
60	72	0	0	0	0	0	0	0
60	72	621	621	621	591	512	621	621
65	77	621	621	621	605	530	621	621
70	82	621	621	621	621	549	621	621
75	87	621	621	621	621	570	621	621
80	92	621	621	621	621	593	621	621
85	97	621	621	621	621	620	621	621
90	102	621	621	621	621	621	621	621
95	107	621	621	621	621	621	621	621
100	112	621	621	621	621	621	621	621
105	117	621	621	621	621	621	621	621
110	122	621	621	621	621	621	621	621
115	127	621	621	621	621	621	621	621
120	132	621	621	621	621	621	621	621
125	137	993	980	970	963		962	822
130	142	1034	1025	1019	1018		1004	846

Table 5: Unit 2 Peak Transient Pressures

Transient Description	Nominal PORV Setpoint (psig)	Pressure Overshoot (psi)	Dynamic Pressure Effects (psi)	Static Pressure Effects (psi)	Instrument Error (psi)	Peak Pressure (psig)
Safety Injection with....						
0 RCP's in Operation	380	***51.3***	***0.1***	4.7	30	**466.1**
1-2 RCP's in Operation	380	***51.3***	***15.9***	4.7	30	**481.9**
3 RCP's in Operation	380	***51.3***	***34.0***	4.7	30	**500.0**
4 RCP's in Operation	380	***51.3***	***58.9***	4.7	30	**524.9**
Charging/Letdown Mismatch with....						
0 RCP's in Operation	380	***45.6***	***0.1***	4.7	30	***460.4***
1-2 RCP's in Operation	380	***45.6***	***15.9***	4.7	30	***476.2***
3 RCP's in Operation	380	***45.6***	***34.0***	4.7	30	***494.3***
4 RCP's in Operation	380	***45.6***	***58.9***	4.7	30	***519.2***
Heat Input (Primary - Secondary Temp Mismatch) with...						
0 RCP's in Operation <u>AND</u>						
Tave of 60 - 100°F	***From Reference 6***					***468.3***
Tave to 100 - 180°F						***522.5***
Tave to 180 - 250°F						***514.9***
1-4 RCP's in Operation	***Transient not credible with an NC pump initially running.***					-

Table 6: Unit 2 Heatup and Cooldown Restrictions

Number of RCP's Running	RCS Temperature Range			Heat-up Limit
	Without Margin	OAC Points	Chart Recorder	Max Rate
0 to 4	> 60°F	> 72°F	> 74°F	100 °F / hr

Number of RCP's Running	RCS Temperature Range			Cooldown Limit
	Without Margin	OAC Points	Chart Recorder	Max Rate
0 - 3	> 60°F	> 72	> 74	100 °F / hr
4	> 65°F 65 - 60	> 77 77 - 72	> 79 79 - 74	100 °F / hr 60 °F / hr

From the previous LTOP LAR (Reference 12), the technical justification for verifying that the enable temperature is adequate to provide overpressure protection states that the licensed basis for the enable temperature ensures that the Appendix G limits are protected by either the PORV setpoint or the pressurizer safety valves (PSV). The PSVs are required to relieve pressure between 2435 - 2559 psig. The corresponding Appendix G steady state pressure limit is 2188 psig at 195 °F and 2,355 psig at 200 °F (Reference 2). Linear interpolation of the P-T limits for a nominal PSV lift setpoint of 2485 psig results in a corresponding temperature limit of 204 °F. Using this methodology, there will be no interruption of overpressure protection if the enable temperature is set at 204 °F (216 °F including instrument error) or above.

Similar to Unit 1:

$T_e = RT_{NDT} + 50 \text{ Ln } [((F.M_m(pR_i/t)) - 33.2)/20.734]$

where, for McGuire Unit 2:

$RT_{NDT} = 123$ °F

F = 1.1, accumulation factor for safety relief valves

M_m = the value of M_m determined in accordance with G-2214.1. As provided in WCAP-15201, Revision 2, Section 3.2, M_m = 0.926(Square Root of t) = 2.69

p = vessel design pressure, 2.485 ksi

R_i = vessel inner radius, 86.5 in.

t = vessel wall thickness, 8.465 in.

Instrument error of 12 °F and a maximum temperature lag of 29.0 °F between the 1/4 T reactor vessel location and coolant temperature during 100 °F/hr heatup are added to the above equation

Thus, T_e for Unit 2 = 123 + 35.2 + 12 + 29.0 = 199.2 °F

The current McGuire Unit 2 LTOP enable temperature of 300 °F bounds each of the methods discussed above, and is acceptable for operation up to 34 EFPY.

Similar to Unit 1, additional restrictions for use of RHR suction relief valve are provided in TS 3.4.12 Required Actions A.2.2.1, A.2.2.2, and F.1:

1. From Table 4, the Appendix G limit for steady-state condition and 100 °F/hr heatup rate is 621 psig (at 60 °F). Therefore, the RHR suction relief valve is adequate for all steady-state and heatup conditions.

2. From Table 4, the Appendix G limit for a 100 °F/hr cooldown is 570 psig at 75 °F (87 °F - 89 °F with instrument uncertainties). Therefore, 100 °F/hr cooldown rate must not be used below 89 °F indicated temperature.

3. From Table 4, the Appendix G limit for a cooldown rate of 60 °F/hr is 591 psig at 60 °F (72 °F - 74 °F with instrument uncertainties). Therefore, cooldown rates of 60 °F/hr or less must be used between 89 °F and 74 °F indicated temperature.

<u>Two Centrifugal and Safety Injection Pumps Capable of Injecting into the RCS during LTOP Mode</u>

Two new Required Actions (A.5.1 and A.5.2) are proposed for TS 3.4.12. This LCO currently requires two operable PORVs (one for single failure concern) or a RCS vent of greater than or equal to 2.75 square inches for a centrifugal charging or safety injection pump capable of injecting into the RCS. The note in the LCO specifies that a PORV secured in the open position with its block valve opened and power removed is equivalent to a RCS vent of greater than or equal to 2.75 square inches. The combination of proposed Action A.5.1, a vent of greater than or equal to 2.75 square inches and proposed Action A.5.2, two operable PORVs is judged to be adequate to prevent RCS overpressurization when any combination of two centrifugal charging and safety injection pumps capable of injecting into the RCS. The effective mass input rate into the RCS from any combination of two centrifugal charging and safety injection pumps would be less than twice that of a single pump.

Steam Line Isolation on Steam Line Pressure Negative Rate - High

The current TS 3.3.2, Function 4.d.(2), Footnote c specifies that this trip function may be blocked below P-11 when Safety Injection Steam Line Pressure - Low is not blocked. Footnote c should specify that this trip function may be blocked below P-11 when Steam Line Isolation on Steam Line Pressure - Low is not blocked. The Safety Injection on Steam Line Pressure - Low function was proposed to be deleted in TS change submittal dated October 6, 1997. The NRC approved this change by letter dated September 22, 1998 (Amendments 182/164). In the conversion from the former Technical Specifications to NUREG-1431-based Technical Specifications, the NUREG-1431 mark-up page was incorrectly marked to reflect the proposed deletion of Safety Injection on Steam Line Pressure - Low as proposed in the October 6, 1997 submittal. The NRC approved the NUREG-1431-based Technical Specifications for McGuire by letter dated September 30, 1998 (Amendments 184/166). Attachment 5 includes the NUREG-1431 mark-up page showing this error.

Impact on UFSAR Accident Analyses

The UFSAR Loss of Coolant Accident analysis does not assume a break in the reactor vessel. Compliance with 10 CFR 50 Appendix G, as modified by ASME Code Case N-641, continues to ensure that the integrity of the reactor vessel is maintained. Therefore, the proposed changes do not impact UFSAR accident analyses.

Summary of Technical Analysis

The proposed changes to the P-T and LTOP limits satisfy the requirements of 10 CFR 50 Appendix G, Appendix H, and ASME Section XI Appendix G, as modified by Code Case N-641. The calculation of ART is consistent with the method in RG 1.99, Revision 2. The calculation of fluence values is consistent with the guidance in RG 1.190. The LTOP changes are performed in accordance with approved procedures under Duke QA program and are consistent with the method in ASME Code Case N-641. Duke concludes that the proposed changes conform to the underlying purpose of NRC regulations and maintain the safe operation of the station.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Duke has evaluated whether or not a significant hazards consideration is involved with the proposed amendments by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed changes to the reactor coolant system (RCS) pressure and temperature (P-T) limits and low temperature overpressure protection (LTOP) limits are developed utilizing the methodology of American Society of Mechanical Engineers (ASME) Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N-641. Usage of these methodologies provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. The Loss of Coolant Accident analysis and other accident analyses in the Updated Final Safety Analysis Report (UFSAR) do not assume failure of the reactor vessel. The P-T and LTOP limits are not initiators or contributors to accident analyses addressed in the UFSAR. The proposed changes do not alter any assumption previously made in the radiological consequence evaluations nor affect the mitigation of the radiological consequences of an accident previously evaluated. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The changes to RCS P-T limits and LTOP limits are proposed to prevent non-ductile failure of the reactor vessel. The proposed changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The proposed changes do not introduce any new mode of system operation or failure mechanism. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed changes are developed utilizing the methodology of ASME Section XI, Appendix G, in conjunction with the methodology of ASME Code Case N-641. Usage of these methodologies provides compliance with the underlying intent of 10 CFR 50 Appendix G and provides operational limits established to prevent non-ductile failure of the reactor vessel. This Code case constitutes relaxation from the current requirements of 10 CFR 50 Appendix G. The alternate methodology allowed by the Code case is based on

industry experience gained since the inception of the 10 CFR 50 Appendix G requirements and replaces some requirements that have now been determined to be excessively conservative. The more appropriate assumptions and provisions allowed by the Code case maintain a margin of safety that is consistent with the intent of 10 CFR 50 Appendix G. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke concludes that the proposed amendments present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The NRC has established requirements in 10 CFR 50 to protect the integrity of the RCS pressure boundary in nuclear power plants. The TS P-T limits for heatup, cooldown, inservice leak testing, and hydrostatic testing are established to define an acceptable operational region for prevention of non-ductile failure of the RCS pressure boundary, and to comply with the fracture toughness requirements of 10 CFR 50 Appendix G. These limits are calculated using the adjusted reference nil-ductility temperature, RT_{NDT}, corresponding to the limiting beltline region material of the RPV. Because of the neutron embrittlement effect on the RPV material toughness, the RT_{NDT} increases as the reactor exposure to neutron fluence increases, and the P-T limits must be adjusted, as necessary, based on exposure evaluation.

The LTOP system, with appropriate relief capacity and setpoints, is designed to automatically prevent the RCS pressure from exceeding the P-T limits, and prevent the RPV from being exposed to conditions of fast propagating brittle fracture. Once the system is enabled, no operator action is involved for the LTOP system to perform its intended pressure mitigation function. Whenever the P-T limits are revised, an evaluation is necessary to determine whether the LTOP enable temperature and setpoint remain acceptable.

The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: Appendix G to 10 CFR 50; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2; and Standard Review Plan (SRP), Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that

could affect their RPV integrity evaluations. These data are used by the staff as the basis for the review of P-T limit curves. Appendix G to 10 CFR 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code, 1995 Edition through the 1996 Addenda.

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic material in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G to Section XI of the ASME Code requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing. Appendix G also requires a safety factor of 1.0 on stress intensities resulting from thermal loads for normal and transient operating conditions, as well as for hydrostatic testing. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress (i.e., of axial orientation). This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to Section XI of the ASME Code requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (Delta RT_{NDT}), and a margin (M) term.

The Delta RT_{NDT} is a product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The M term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2, or surveillance data. The M term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures. RG 1.99, Revision 2, describes the methodology to be used in calculating the M term.

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). The proposed amendment meets the criteria for categorical exclusion if it does not involve the following:

1. A significant hazards consideration:

The proposed changes do not involve a significant hazards consideration. This conclusion is supported by the no significant hazards consideration evaluation above.

2. A significant change in the types or significant increase in the amounts of any effluent that may be released offsite:

The proposed changes provide operational limits established to prevent non-ductile failure of the reactor vessel. The changes do not modify the RCS pressure boundary, nor make any physical changes to the facility. The proposed changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. Therefore, the proposed change does not involve a significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

3. A significant increase in individual or cumulative occupational radiation exposure:

In addition to the above, the proposed changes do not involve any new mode of system operation or failure mechanism. Therefore, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendments meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendments.

7.0 REFERENCES

1) WCAP-15192, Revision 2, McGuire Unit 1 Heatup and Cooldown Limit Curves for Normal Operation, September 2002.

2) WCAP-15201, Revision 2, McGuire Unit 2 Heatup and Cooldown Curves for Normal Operation, September 2002.

3) WCAP-14040, Revision 3, Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Unit Curves, April 2002.

4) MCC-1223.03-00-0005, McGuire Nuclear Station Unit 1 Pressurizer Power Operated Relief Valve Setpoint Determination for the LTOP System, November 05, 2002.

5) MCC-1223.03-00-0033, McGuire Nuclear Station Unit 2 Pressurizer Power Operated Relief Valve Setpoint Determination for the LTOP System, November 07, 2002.

6) MCC-1552.08-00-0305, Rev 0, McGuire Nuclear Station Replacement Steam Generator LTOP Heat Input Transient, January 28, 2000.

7) MCC-1201.01-00-0045, Revision 0, Evaluation of McGuire 16 EFPY Operating Curves, September 2002.

8) NRC letter to North Anna Power Station dated May 2, 2001, Issuance of Amendments 226 and 207. These amendments approve new P-T limits, LTOP system setpoints, and LTOP system effective temperature in the Technical Specifications. McGuire's submittal is very similar to this submittal.

9) NRC letter to Arkansas Nuclear One dated April 15, 2002, Issuance of Amendment 242. This amendment approves revised P-T limits for the reactor pressure vessel, and to approve additional Technical Specification restrictions for operation of the LTOP system. McGuire's submittal is similar to this submittal except LTOP effective temperature determination.

10) NRC letter to Point Beach dated October 6, 2002, Exemption from the Requirements of 10 CFR 50.60. This exemption allows the use of ASME Code Case N-641 in support of the licensee's application for revised P-T limits. McGuire's exemption is very similar to this exemption.

11) NRC letter to Turkey Point dated October 24, 2000, Exemption from the Requirements of 10 CFR 50.60 and Appendix G. This exemption allows the licensee to apply the methodology of ASME Code Cases N-588 and N-641 as the basis for establishing the P-T limits specified in Technical Specifications and the cold overpressure mitigation system. McGuire's exemption is similar to this exemption.

12) Duke letter to NRC dated March 29, 1995, McGuire Nuclear Station Units 1 and 2, Technical Specification 3.4.9.3 Amendment, Heatup and Cooldown Curves and LTOP.

ENCLOSURE 3

EXEMPTION REQUEST FOR USE OF ASME CODE CASE N-641

Request for Exemption from 10 CFR 50 Appendix G Requirements

Pursuant to 10 CFR 50.60, the following information provides the basis for the request for exemption from 10 CFR 50 Appendix G requirements, for use of ASME Code Case N-641. 10 CFR 50.60 states that alternates to the requirements in Appendix G of this part may be used when an exemption is granted by the Commission pursuant to 10 CFR 50.12. 10 CFR 50.12 states that the Commission may grant an exemption from requirements contained in 10 CFR 50 provided that:

1. The requested exemption is authorized by law:

   No law exists which precludes the activities covered by this exemption request. 10 CFR 50.60(b) allows the use of alternates to 10 CFR 50 Appendix G and Appendix H when an exemption is granted by the Commission under 10 CFR 50.12.

2. The requested exemption does not present an undue risk to the public health and safety:

   The proposed P-T limits and LTOP limits rely on the requested exemption. The proposed P-T limits have been developed using the K_{IC} fracture toughness curve shown on ASME XI, Appendix A, Figure A-2200-1, in lieu of the K_{IA} fracture toughness curve of ASME XI, Appendix G, Figure G-2210-1. Margins that exist in the ASME XI, Appendix G P-T limit determination process are unaffected by this request.

   Use of the K_{IC} fracture toughness curve in the development of P-T and LTOP limits is more realistic than the assumption under the use of the K_{IA} fracture toughness curve. The K_{IC} fracture toughness curve models the slow heatup and cooldown process of a reactor coolant system, with the fastest rate allowed being 100 °F per hour. The rate of change of pressure and temperature is often constant, so the stress is essentially constant in this case. Both the heatup and cooldown and the pressure testing are essentially static processes. During development of ASME Code Case N-641 and the accompanying Appendix G Code change, the ASME Section XI Working Group on Operating Plant Criteria (WGOPC) performed assessments of the margins inherent to K_{IA} using realistic heatup and cooldown curves. These assessments led to the conclusion that utilization of the K_{IA} fracture toughness curve was excessively conservative and the K_{IC} fracture toughness curve provided adequate margin for protection from brittle fracture.

   The K_{IA} fracture toughness curve was codified in 1974. The initial K_{IA} conservatism was necessary due to limited experience and knowledge of the fracture toughness of reactor pressure vessel materials over time. The conservatism also provided margin thought to be necessary to

cover uncertainties and a number of postulated but unquantified effects. Since 1974, additional knowledge has been gained from examination and testing of reactor pressure vessels that had been subject to the effects of neutron embrittlement in both an operating and test environment. The K_{IA} fracture toughness curve was based on 125 data points. The K_{IC} fracture toughness curve is based on more than 1500 data points. The additional data has significantly reduced the uncertainties associated with embrittlement effects and reduced other uncertainties. The new information indicates the lower bound on fracture toughness provided by the K_{IA} curve is extremely conservative and is well beyond the margin of safety required to protect the public health and safety from potential reactor pressure vessel failure.

3. <u>The requested exemption will not endanger the common defense and security</u>:

This request does not modify any physical plant architectural features, surveillance or alarm features. Therefore, the common defense and security are not endangered by this exemption request.

4. <u>Special circumstances are present which necessitate the request for an exemption from 10 CFR 50 Appendix G</u>:

Compliance with the regulation would result in undue hardship or other cost that are significant. The RCS P-T operating window is defined by the P-T limit curve developed in accordance with the ASME Section XI Appendix G methodology and the minimum P-T curve for pump operation. Continued operation of McGuire with these P-T curves without the relief provided by ASME Code Case N-641 would unnecessarily restrict the operating window that results from these operating P-T limits. This constitutes an unnecessary burden that can be alleviated by the application of ASME Code Case N-641 in the development of the proposed P-T curves and LTOP setpoints. Implementation of the proposed P-T curves and LTOP setpoints as allowed by ASME Code Case N-641 does not significantly reduce the margin of safety.

The exemption provides only temporary relief from the applicable regulation until the Code case is codified in the applicable regulation. The underlying purpose of the regulation will continue to be achieved. Application of ASME Code Case N-641 methodology satisfies the underlying requirement for: 1) The RCS pressure boundary be operated in a regime having sufficient margin to ensure, when stressed, the vessel boundary behaves in a non-brittle manner and the probability of a rapidly propagating fracture is minimized; and 2) P-T limits provide adequate margin in consideration

of uncertainties in determining the effects of irradiation on material properties.

Conclusion for Exemption Acceptability:

Compliance with the requirements of 10 CFR 50 Appendix G would result in hardship and unusual difficulty without a compensating increase in the level of quality and safety. ASME Code Case N-641 allows a reduction in the fracture toughness used by ASME Section XI Appendix G in the determination of RCS P-T limits. This proposed alternative is acceptable because it reduces the excess conservatism in the current Appendix G. The safety margin that exists with the revised methodology is still very large. Restrictions on allowable operating conditions and equipment operability requirements are established to ensure RCS pressure and temperature are within the heatup and cooldown P-T limits specified in Technical Specifications. Therefore, exemption from 10 CFR 50 Appendix G requirements for use of ASME Code Case N-641 in the development of P-T and LTOP limits should be granted.

ENCLOSURE 4

WESTINGHOUSE WCAP-15192, REVISION 2

WCAP-15192, Revision 2

McGuire Unit 1
Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

SEPTEMBER 2002

Prepared by the Westinghouse Electric Company LLC
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PREFACE

This report has been technically reviewed and verified by:

J.H. Ledger _____

RECORD OF REVISION

Revision 0: Original Issue

Revision 1: An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15192 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15192 Rev. 0.

Note that only the heatup curves and associated data point tables have changed. The cooldown curves and data points remain valid and were not changed.

Revision 2: Revised PT Curves to account for the use of "Calculated" Neutron Fluence Projections and to included the full flange requirement from 10 CFR 50 Appendix G. Test and tables have been updated to account for the changes described above. In addition to these changes, the Initial RT_{NDT} of the Closure Head Flange has been revised from 40°F to 10°F, while the initial RT_{NDT} of the Vessel Flange has been revised from 29°F to 10°F (See Appendix A). The credibility analysis for the weld heat 21935/12008 from Diablo Canyon Unit 2 has been updated due to an additional surveillance capsule data point (See Reference 7).

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EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature (PT) limit curves for normal operation of the McGuire Unit 1 reactor vessel. The PT curves were generated based on the latest available reactor vessel information and updated calculated fluences. The new McGuire Unit 1 heatup and cooldown pressure-temperature limit curves were generated using the "axial flaw" methodology of 1995 ASME Code, Section XI through the 1996 Addenda. In addition, the PT curves were developed using ASME Code Case N-641, which allows the use of the K_{Ic} methodology. The materials with the highest adjusted reference temperature (ART) were the lower shell longitudinal weld seams. The PT limit curves were generated for 34 EFPY using heatup rates of 60, 80 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. These curves can be found in Figures 5-1 through 5-3.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."[1] Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) **The fluence values used in this report are calculated fluence values (i.e. comply with Reg. Guide 1.190), not the best estimate fluence values.** 2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641[3] (which covers Code Cases N-640 and N-588). 3) The 1996 Version of Appendix G to Section XI[4] will be used rather than the 1989 version.

The purpose of this report is to present the calculations and the development of the McGuire Unit 1 heatup and cooldown curves for 34 EFPY. This report documents the calculated ART values and the development of the PT limit curves for normal operation. The PT curves herein were generated without instrumentation errors. The PT curves include a hydrostatic leak test limit curve from 2485 psig to 2000 psig, along with the pressure-temperature limits for the vessel flange region per the requirements of 10 CFR Part 50, Appendix G[5].

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan [6]. The beltline material properties of the McGuire Unit 1 reactor vessel are presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for four capsules (Capsules U, X, V and Y) already removed from the McGuire Unit 1 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 3. These CF values are summarized in Table 4. It should be noted that in addition to McGuire Unit 1, surveillance weld data from Diablo Canyon Unit 2 was used in the determination of CF for the lower shell longitudinal weld seams of heat # 21935/12008. It should also be noted that Diablo Canyon Unit 2 operates at a lower temperature than McGuire Unit 1. Thus, for conservatism (i.e reduction in ΔRT_{NDT}), no temperature adjustments were made to the chemistry factor for weld heat # 21935/12008.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1
Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the McGuire Unit 1 Reactor Vessel Materials

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange B5002	---	0.75	10°F
Vessel Flange B4701	---	0.73	10°F
Intermediate Shell Plate B5012-1	0.11	0.61	34°F
Intermediate Shell Plate B5012-2	0.14	0.61	0°F
Intermediate Shell Plate B5012-3	0.11	0.66	-13°F
Lower Shell Plate B5013-1	0.14	0.58	0°F
Lower Shell Plate B5013-2	0.10	0.51	30°F
Lower Shell Plate B5013-3	0.10	0.55	15°F
Intermediate Shell Longitudinal Welds, 2-442A, B & C[b]	0.199	0.846	-50°F
Lower Shell Longitudinal Welds, 3-442A, B & C[b]	0.213	0.867	-50°F
Circumferential Weld 9-442[b]	0.051	0.096	-70°F
McGuire 1: Surveillance Program Weld Metal	0.198	0.874[c]	---
Diablo Canyon 2: Surveillance Program Weld Metal	0.219	0.867	---

Notes:

(a) The Initial RT_{NDT} values for the plates and weld are based on measured data. The closure head flange and vessel flange values were revised to 10°F from 40°F and 29°F, respectively. See Appendix A for this re-evaluation of Initial RT_{NDT} for the flanges.

(b) The intermediate shell longitudinal weld seams 2-442A, B and C were fabricated with weld wire heat numbers 20291 and 12008, Flux Type 1092 Lot Number 3854. The intermediate to lower shell circumferential weld seam 9-442 was fabricated with weld wire heat number 83640, Flux Type 0091 Lot Number 3490. The lower shell longitudinal weld seams 3-442A, B and C were fabricated with weld wire heat number 21935 and 12008, Flux Type 1092 Lot Number 3889. The McGuire Unit 1 surveillance weld metal was made with the same weld heat as the intermediate shell longitudinal weld seams, while the Diablo Canyon Unit 2 surveillance weld metal was made with the same weld wire heat as the lower shell longitudinal weld seams (Justification in WCAP-13949, Ref. 8) Per Reg. Guide 1.99, Revision 2, "weight percent copper" and "weight percent nickel" are the best-estimate values for the material, which will normally be the mean of the measured values for a plate or forging or for weld samples made with the weld wire heat number that matches the critical vessel weld."

(c) Value is the average of five measurements (see Tables 4-1 and 4-2 in WCAP-14993[9]).

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 3 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases. Included in Table 2 are the Calculated fluence values for McGuire Unit 1, along with Diablo Canyon Unit 2. All capsule fluence values were determined using ENDF/B-VI cross-sections and followed the guidance in Regulatory Guide 1.190[10].

It should be noted that in the calculation of chemistry factor in Table 3, the McGuire Unit 1 and Diablo Canyon Unit 2 data had the ratio procedure applied. In both cases the ratio was slightly less than 1.0. As for Temperature adjustments are concerned, the McGuire Unit 1 data does not require any adjustments since it is being applied to its own plant. The Diablo Canyon surveillance data, on the other hand, should be adjusted for temperature. However, in this instance, it would cause a further reduction in ΔRT_{NDT} (or Chemistry Factor) since Diablo Canyon Unit 2 operated at a lower temperature than McGuire (for the capsules already withdrawn from Diablo Canyon Unit 2). Thus, it was decided for conservatism, not to reduce the chemistry factor any further beyond what was done with the ratio procedure.

Credibility

Surveillance capsule data exists from McGuire Unit 1 for the intermediate shell plate B5012-1 and the intermediate shell longitudinal welds (Heat # 20291/12008). Additionally, surveillance capsule weld data (Heat # 21935/12008) was obtained from Diablo Canyon Unit 2. This is the same heat as the McGuire Unit 1 lower shell longitudinal welds. The McGuire surveillance data has been deemed credible in a Technical Report (ATI-98-012-T005, Revision 1, Dated November 1998 by Tim Greisbach), which has been issue to the NRC through Duke Energy (From H.B. Baron on 1/7/99). The surveillance weld data from Diablo Canyon was evaluated for credibility under WCAP-15423[7]. In this report, it was determined that the first of four weld data points falls outside the scatter band of +/- 28°F. Regulatory Guide 1.99, Revision 2 treats this data as non-credible, however, the data is very predictable when analyzed using more recent embrittlement correlations, such as the recently approved ASTM E-900 correlation. Thus, the Diablo Canyon Unit 2 surveillance weld data will be utilized for determination of the limiting adjusted reference temperature (ART), with additional conservatism using the full σ_Δ in the margin term.

TABLE 2

Calculated Integrated Neutron Exposure of the Surveillance Capsules

@ McGuire Unit 1 and Diablo Canyon Unit 2

Capsule	Fluence
McGuire Capsule Fluences[a]	
U	4.05×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.50×10^{19} n/cm^2, (E > 1.0 MeV)
V	2.08×10^{19} n/cm^2, (E > 1.0 MeV)
Z[c]	2.38×10^{19} n/cm^2, (E > 1.0 MeV)
Y	2.86×10^{19} n/cm^2, (E > 1.0 MeV)
Diablo Canyon Unit 2[b]	
U	3.38×10^{18} (E > 1.0 MeV)
X	9.19×10^{18} (E > 1.0 MeV)
Y	1.55×10^{19} (E > 1.0 MeV)
V	2.41×10^{19} (E > 1.0 MeV)

NOTES:

(a) Re-Calculated in WCAP-15253[12].

(b) Per WCAP-15423[7].

(c) Dosimeters only.

TABLE 3

Calculation of Chemistry Factors using McGuire Unit 1 and Diablo Canyon Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell	U	0.405	0.749	30.95	23.18	0.561
Plate B5012-1	X	1.50	1.11	33.51	37.20	1.23
	V	2.08	1.20	81.01	97.21	1.44
(Longitudinal)	Y	2.86	1.28	93.10	119.17	1.64
Intermediate Shell	U	0.405	0.749	48.44	36.28	0.561
Plate B5012-1	X	1.50	1.11	60.69	67.37	1.23
	V	2.08	1.20	74.60	89.52	1.44
(Transverse)	Y	2.86	1.28	108.58	138.98	1.64
				SUM:	608.91	9.742
	$CF_{B5012-1} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (608.91) \div (9.742) = 62.5°F$					
Inter. Shell Longitudinal	U	0.405	0.749	159.71[d]	119.62	0.561
Welds 2-442A, B & C	X	1.50	1.11	168.98[d]	187.57	1.23
	V	2.08	1.20	178.35[d]	214.02	1.44
	Y	2.86	1.28	188.52[d]	241.31	1.64
				SUM:	762.52	4.871
	$CF_{Weld\ 2-442} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (762.52) \div (4.871) = 156.5°F$					
Lower Shell Long.	U	0.338[e]	0.701	170.57[e]	119.57	0.491
Welds 3-442A, B & C	X	0.919[e]	0.976	200.38[e]	195.57	0.953
(Using Diablo Canyon	Y	1.55[e]	1.121	208.43[e]	233.65	1.257
Unit 2 Surv. Data)	V	2.41[e]	1.237	221.33[e]	273.79	1.530
				SUM:	822.58	4.231
	$CF_{Weld\ 3-442} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (822.58) \div (4.231) = 194.4°F$					

Notes:

(a) f = Calculated fluence (x 10^{19} n/cm^2, E > 1.0 MeV), See Table 2.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values.

(d) Ratio = 0.99 (201.3 ÷ 204.2).

(e) Updated Per Capsule V, WCAP-15423. Ratio = 0.986 (208.2 ÷ 211.2). For Conservatism, No Temperature Adjustment used.

TABLE 4
Summary of the McGuire Unit 1 Reactor Vessel Beltline Material Chemistry Factors

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell Plate B5012-1	74.2°F	62.5°F
Intermediate Shell Plate B5012-2	100.3°F	---
Intermediate Shell Plate B5012-3	74.9°F	---
Lower Shell Plate B5013-1	99.1°F	---
Lower Shell Plate B5013-2	65.0°F	---
Lower Shell Plate B5013-3	65.0°F	---
Inter. Shell Plate Long. Weld Seams 2-442A, B, C (Heat # 20291/12008)[b]	201.3°F	156.5°F
Lower Shell Plate Long. Weld Seams 3-442A, B, C (Heat # 21935/12008)[c]	208.2°F	194.4°F[a]
Inter. to Lower Shell Plate Circ. Weld Seam 9-442 (Heat # 83640)	37.5°F	---
McGuire 1 Surveillance Program Weld Metal[b]	204.2°F	---
Diablo Canyon 2 Surveillance Program Weld Metal[c]	211.2°F	---

Notes:

(a) This was determined using surveillance capsule data from Diablo Canyon Unit 2.

(b) The McGuire Unit 1 surveillance capsule weld material fabricated with the same weld wire heat as the intermediate shell longitudinal weld seams 2-442A, B, C (Heat # 20291/12008).

(c) The Diablo Canyon Unit 2 surveillance capsule weld material was fabricated with the same weld wire heat as the lower shell longitudinal weld seams 3-442A, B, C of the McGuire Unit 1 (Heat # 21935/12008).

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 OVERALL APPROACH

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_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1"[3 & 4] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

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

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \quad (3)$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \quad (4)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e , inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code 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 the 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 inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation Furthermore, if conditions exist so that the increase in K_{Ic} 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 and, 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 ensures conservative operation of the system for the entire cooldown period.

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 wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values 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 ensure 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 the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface 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 therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate 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 by constructing a composite curve 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 wherein, 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.

3.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G[5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which is 621 psig for McGuire Unit 1. The limiting unirradiated RT_{NDT} of 10°F (See Appendix A) occurs in both the vessel flange and closure head flange of the McGuire Unit 1 reactor vessel, so the minimum allowable temperature of this region is 130°F at pressures greater than 621 psig. This limit is shown in Figures 6-1 and 6-2 wherever applicable.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[11]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \quad (9)$$

where x inches (vessel beltline thickness is 8.63 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections and the results of the **calculated peak fluence values** at various azimuthal locations on the vessel clad/base metal interface are presented in Table 5. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves". Table 6 contains the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the ART values for all beltline materials in the McGuire Unit 1 reactor vessel.

TABLE 5

Summary of the Peak Pressure Vessel Neutron Fluence Values at the Clad/Base Metal Interface at 34 EFPY (n/cm^2, E > 1.0 MeV)

0°	15°	30°	45°
1.21×10^{19}	1.79×10^{19}	1.73×10^{19}	1.95×10^{19}

TABLE 6
Summary of the Peak Pressure Vessel Neutron Fluence Values
at 34 EFPY used for the Calculation of ART Values
(n/cm^2, E > 1.0 MeV)

Material	Surface	¼ T	¾ T
Intermediate Shell Plate B5012-1	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Intermediate Shell Plate B5012-2	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Intermediate Shell Plate B5012-3	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Lower Shell Plate B5013-1	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Lower Shell Plate B5013-2	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Lower Shell Plate B5013-3	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	1.21×10^{19}	0.721×10^{19}	0.256×10^{19}
Intermediate Shell Longitudinal Weld Seam 2-442B & C (120° & 240° Azimuth)	1.73×10^{19}	1.03×10^{19}	0.366×10^{19}
Intermediate to Lower Shell Cirumferential Weld Seam 9-442	1.95×10^{19}	1.16×10^{19}	0.412×10^{19}
Lower Shell Longitudinal Weld Seams 3-442A & C (60° & 300° Azimuth)	1.73×10^{19}	1.03×10^{19}	0.366×10^{19}
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	1.21×10^{19}	0.721×10^{19}	0.256×10^{19}

Contained in Table 7 is a summary of the fluence factor (FF) values used in the calculation of adjusted reference temperatures for the McGuire Unit 1 reactor vessel beltline materials for 34 EFPY.

TABLE 7
Summary of the Calculated Fluence Factors used for the Generation of the 34 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm2, E > 1.0 MeV)	¼ T FF[a]	¾ T f (n/cm2, E > 1.0 MeV)	¾ T FF[b]
Intermediate Shell Plate B5012-1	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Intermediate Shell Plate B5012-2	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Intermediate Shell Plate B5012-3	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Lower Shell Plate B5013-1	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Lower Shell Plate B5013-2	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Lower Shell Plate B5013-3	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	0.721×10^{19}	0.91	0.256×10^{19}	0.63
Intermediate Shell Longitudinal Weld Seams 2-442B & C (120° & 240°Azimuth)	1.03×10^{19}	1.01	0.366×10^{19}	0.72
Intermediate to Lower Shell Cirumferential Weld Seam 9-442	1.16×10^{19}	1.04	0.412×10^{19}	0.75
Lower Shell Longitudinal Weld Seams 3-442A & C (60° & 300° Azimuth)	1.03×10^{19}	1.01	0.366×10^{19}	0.72
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	0.721×10^{19}	0.91	0.256×10^{19}	0.63

Notes:
(a) Fluence Factor at the 1/4T vessel thickness location.
(b) Fluence Factor at the 3/4T vessel thickness location.

Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_I 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 8 and 9 are the calculations of the 34 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT} (°F)	Margin (°F)	ART[b] (°F)
Intermediate Shell Plate B5012-1	Position 1.1	74.2	1.04	34	77.2	34	145
	Position 2.1	62.5	1.04	34	65.0	17	116
Intermediate Shell Plate B5012-2	Position 1.1	100.3	1.04	0	104.3	34	138
Intermediate Shell Plate B5012 -3	Position 1.1	74.9	1.04	-13	77.9	34	99
Lower Shell Plate B5013-1	Position 1.1	99.1	1.04	0	103.1	34	137
Lower Shell Plate B5013-2	Position 1.1	65.0	1.04	30	67.6	34	132
Lower Shell Plate B5013 -3	Position 1.1	65.0	1.04	15	67.6	34	117
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	Position 1.1	201.3	0.91	-50	183.2	56	189
	Position 2.1	156.5	0.91	-50	142.4	28	120
Intermediate Shell Longitudinal Weld Seam 2-442B & C (120° & 240° Azimuth)	Position 1.1	201.3	1.01	-50	203.3	56	209
	Position 2.1	156.5	1.01	-50	158.1	28	136
Intermediate to Lower Shell Circumferential Weld Seam 9-442	Position 1.1	37.5	1.04	-70	39.0	39.0	8
Lower Shell Longitudinal Weld Seams 3-442A & C (60°& 300° Azimuth)	Position 1.1	208.2	1.01	-50	210.3	56	216
	Position 2.1	194.4	1.01	-50	196.3	56[d]	202
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	Position 1.1	208.2	0.91	-50	189.5	56	196
	Position 2.1	194.4	0.91	-50	176.9	28	155

Notes:
(a) Initial RT_{NDT} values are measured values
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF
(d) Deemed Not Credible (Reference 7), thus the full σ_Δ was used in the Margin Term.

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT} (°F)	Margin (°F)	ART[b] (°F)
Intermediate Shell Plate B5012-1	Position 1.1	74.2	0.75	34	55.7	34	124
	Position 2.1	62.5	0.75	34	46.9	17	98
Intermediate Shell Plate B5012-2	Position 1.1	100.3	0.75	0	75.2	34	109
Intermediate Shell Plate B5012 -3	Position 1.1	74.9	0.75	-13	56.2	34	77
Lower Shell Plate B5013-1	Position 1.1	99.1	0.75	0	74.3	34	108
Lower Shell Plate B5013-2	Position 1.1	65.0	0.75	30	48.8	34	113
Lower Shell Plate B5013 -3	Position 1.1	65.0	0.75	15	48.8	34	98
Intermediate Shell Longitudinal Weld Seam 2-442A (0° Azimuth)	Position 1.1	201.3	0.63	-50	126.8	56	133
	Position 2.1	156.5	0.63	-50	98.6	28	77
Intermediate Shell Longitudinal Weld Seam 2-442B & C (120° & 240° Azimuth)	Position 1.1	201.3	0.72	-50	144.9	56	151
	Position 2.1	156.5	0.72	-50	112.7	28	91
Intermediate to Lower Shell Circumferential Weld Seam 9-442	Position 1.1	37.5	0.75	-70	28.1	28.1	-14
Lower Shell Longitudinal Weld Seams 3-442A & C (60°& 300° Azimuth)	Position 1.1	208.2	0.72	-50	149.9	56	156
	Position 2.1	194.4	0.72	-50	140.0	56[d]	146
Lower Shell Longitudinal Weld Seam 3-442B (180° Azimuth)	Position 1.1	208.2	0.63	-50	131.2	56	137
	Position 2.1	194.4	0.63	-50	122.5	28	101

Notes:
(a) Initial RT_{NDT} values are measured values.
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF
(d) Deemed Not Credible (Reference 7), thus the full σ_Δ was used in the Margin Term

The Lower Shell Longitudinal Welds (Seams 3-442A and C) are the limiting beltline material for all the PT limit curves to be generated. Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the McGuire Unit 1 reactor vessel PT limit curves. These limiting curves will be presented in Section 5.

TABLE 10
Summary of the Limiting ART Values Used in the
Generation of the McGuire Unit 1 Heatup/Cooldown Curves

¼ T Limiting ART	¾ T Limiting ART
202	146

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3.0 and 4.0 of this report This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figures 1 and 2 present the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60, 80 and 100°F/hr applicable for the first 34 EFPY. These curves were generated using a combination of the1996 ASME Code Section XI, Appendix G with the limiting ART values and the ASME Code Case N-641. Figure 3 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 34 EFPY. Again, these curves were generated using a combination of the1996 ASME Code Section XI, Appendix G with the limiting ART values and the ASME Code Case N-641. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit line shown in Figures 1 through 3. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 and 2. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-641[3] (approved in February 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\,(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 5. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 4.0 of this report For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperatures for the in service hydrostatic leak tests for the McGuire Unit 1 reactor vessel at 34 EFPY is 262°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve constitutes the limit for core operation for the reactor vessel.

Figures 1 through 3 define all of the above limits for ensuring prevention of nonductile failure for the McGuire Unit 1 reactor vessel for 34 EFPY. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 3 are presented in Tables 11, 12 and 13.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

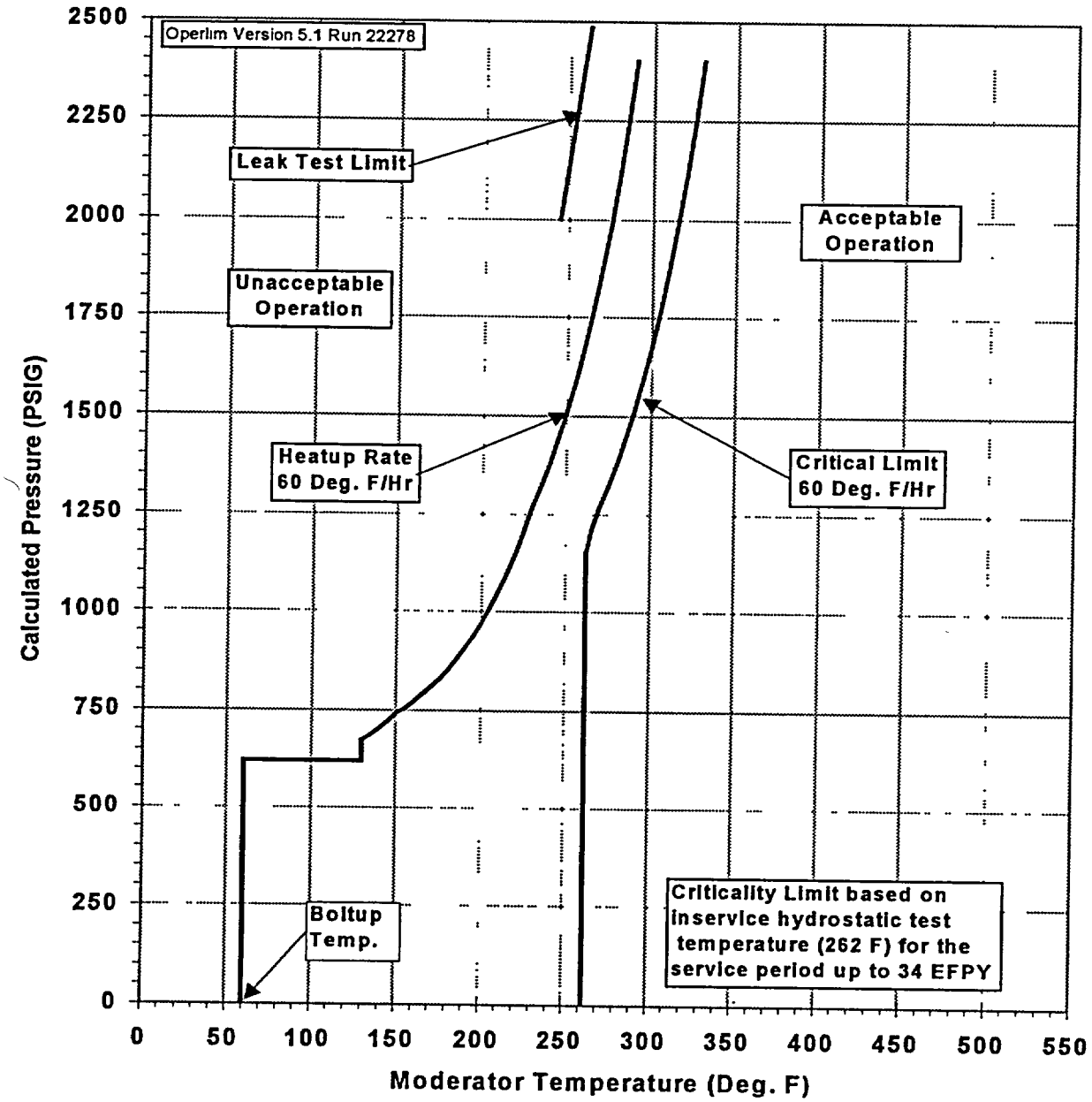

Figure 1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

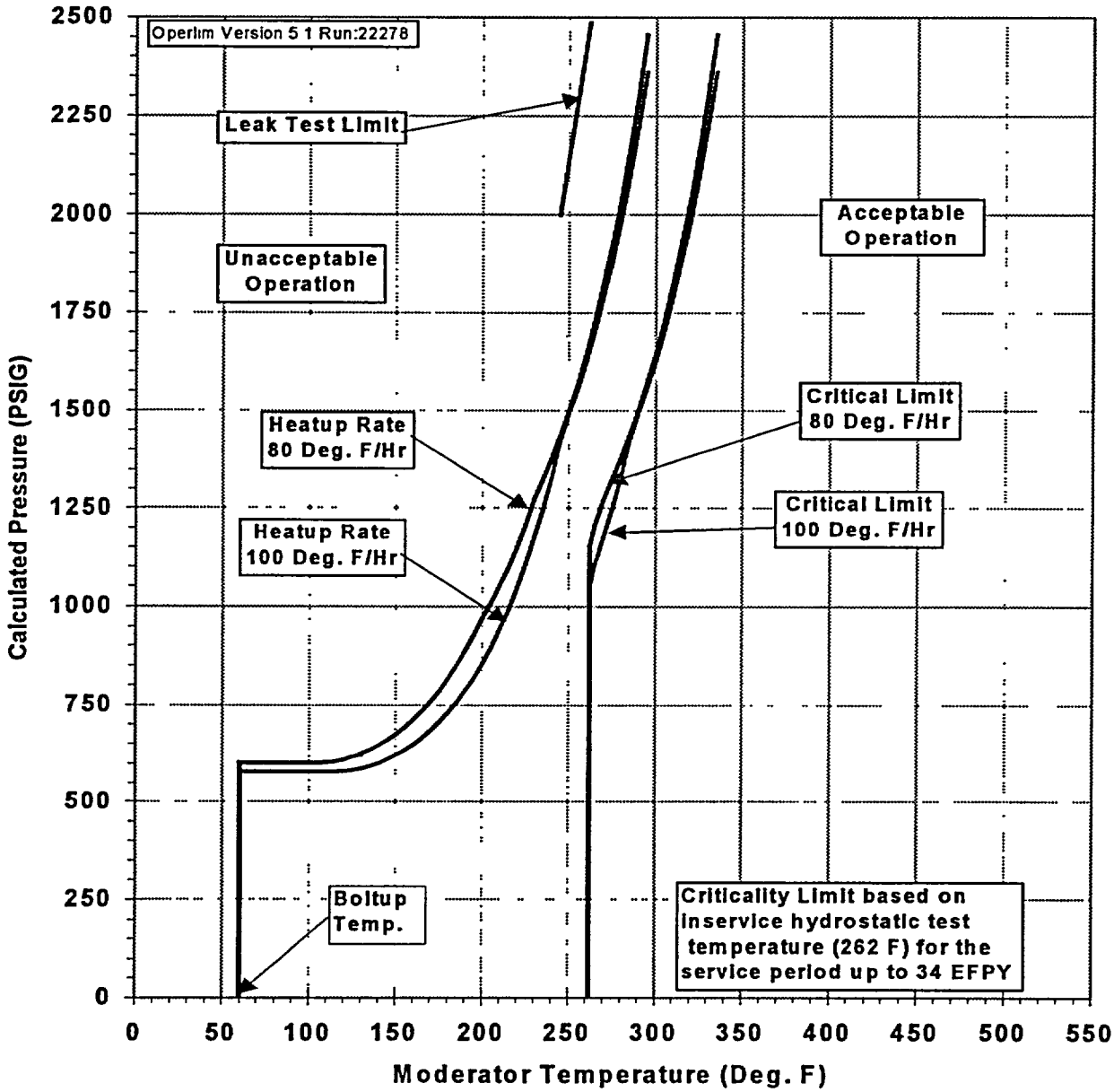

Figure 2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T,202°F
3/4T,146°F

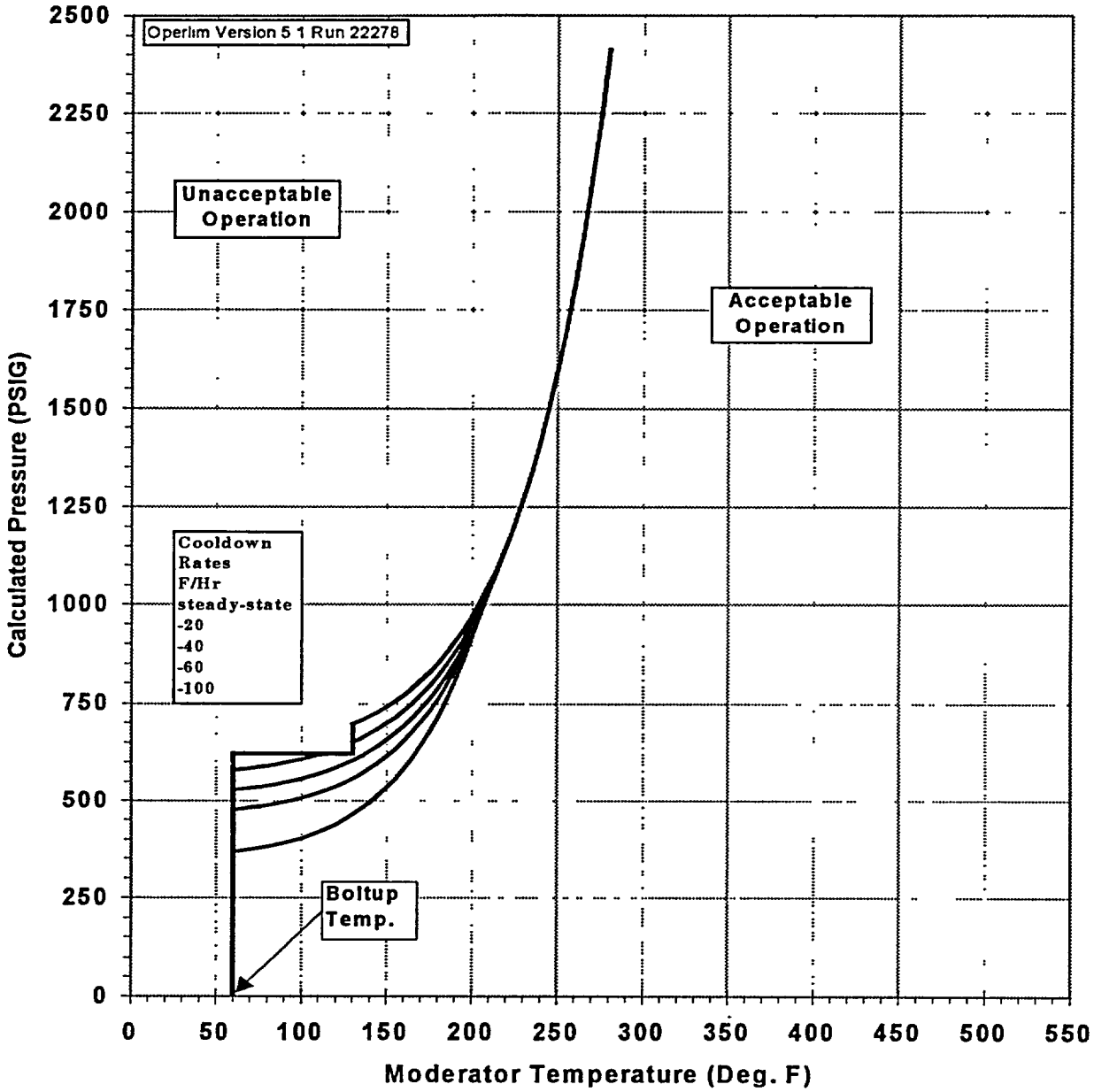

Figure 3 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

TABLE 11

34 EFPY, 60°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641

(without Uncertainties for Instrumentation Errors)

60 °F/hr		Criticality Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	262	0	245	2000
60	621	262	621	262	2485
65	621	262	621		
70	621	262	621		
75	621	262	621		
80	621	262	621		
85	621	262	621		
90	621	262	621		
95	621	262	621		
100	621	262	621		
105	621	262	621		
110	621	262	621		
115	621	262	621		
120	621	262	621		
125	621	262	621		
130	621	262	621		
130	621	262	621		
130	673	262	673		
135	687	262	687		
140	703	262	703		
145	721	262	721		
150	741	262	741		
155	755	262	755		
160	771	262	771		
165	788	262	788		
170	807	262	807		
175	828	262	828		
180	851	262	851		
185	877	262	877		
190	905	262	905		
195	937	262	937		
200	972	262	972		
205	1010	262	1010		
210	1052	262	1152		

TABLE 11 - (Continued)
34 EFPY, 60°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641
(without Uncertainties for Instrumentation Errors)

60 °F/hr		Criticality Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
215	1099	265	1208		
220	1151	270	1271		
225	1208	275	1323		
230	1271	280	1380		
235	1323	285	1443		
240	1380	290	1513		
245	1443	295	1589		
250	1513	300	1674		
255	1589	305	1767		
260	1674	310	1870		
265	1767	315	1983		
270	1870	320	2109		
275	1983	325	2247		
280	2109	330	2399		
285	2247				
290	2399				

TABLE 12

34 EFPY, 80 and 100°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641

(without Uncertainties for Instrumentation Errors)

80 °F/hr		Criticality Limit		100 °F/hr		Critical. Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	262	0	60	0	262	0	245	2000
60	601	262	601	60	579	262	579	262	2485
65	601	262	601	65	579	262	579		
70	601	262	602	70	579	262	579		
75	601	262	603	75	579	262	579		
80	601	262	605	80	579	262	579		
85	601	262	607	85	579	262	579		
90	601	262	610	90	579	262	579		
95	601	262	614	95	579	262	579		
100	601	262	616	100	579	262	579		
105	601	262	621	105	579	262	579		
110	602	262	621	110	579	262	579		
115	605	262	621	115	579	262	579		
120	610	262	621	120	580	262	580		
125	616	262	621	125	583	262	583		
130	621	262	624	130	587	262	587		
130	621	262	634	135	593	262	593		
130	624	262	645	140	600	262	600		
135	634	262	658	145	609	262	609		
140	645	262	673	150	620	262	620		
145	658	262	691	155	633	262	633		
150	673	262	710	160	647	262	647		
155	691	262	732	165	664	262	664		
160	710	262	757	170	683	262	683		
165	732	262	784	175	704	262	704		
170	757	262	814	180	728	262	728		
175	784	262	848	185	755	262	755		
180	814	262	886	190	784	262	784		
185	848	262	927	195	818	262	818		
190	886	262	972	200	854	262	854		
195	927	262	1010	205	895	262	895		
200	972	262	1052	210	941	262	941		
205	1010	262	1099	215	991	262	991		
210	1052	262	1151	220	1047	262	1047		

TABLE 12 – (Continued)
34 EFPY, 80 and 100°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641
(without Uncertainties for Instrumentation Errors)

80 °F/hr		Criticality Limit		100 °F/hr		Critical Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
215	1099	265	1208	225	1108	265	1108		
220	1151	270	1271	230	1176	270	1176		
225	1208	275	1328	235	1251	275	1251		
230	1271	280	1380	240	1334	280	1334		
235	1328	285	1437	245	1425	285	1425		
240	1380	290	1500	250	1494	290	1494		
245	1437	295	1570	255	1557	295	1557		
250	1500	300	1647	260	1627	300	1627		
255	1570	305	1731	265	1704	305	1704		
260	1647	310	1824	270	1789	310	1789		
265	1731	315	1927	275	1882	315	1882		
270	1824	320	2040	280	1985	320	1985		
275	1927	325	2165	285	2098	325	2098		
280	2040	330	2303	290	2223	330	2223		
285	2165	335	2455	295	2360	335	2360		
290	2303								
295	2455								

TABLE13
34 EFPY Cooldown Curve Data Points Using 1996 App. G & ASME Code Case N-641
(without Uncertainties for Instrumentation Errors)

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	579	60	528	60	476	60	367
65	621	65	582	65	530	65	478	65	370
70	621	70	584	70	533	70	481	70	373
75	621	75	587	75	536	75	484	75	376
80	621	80	590	80	539	80	488	80	380
85	621	85	594	85	543	85	492	85	385
90	621	90	598	90	547	90	496	90	390
95	621	95	602	95	552	95	501	95	396
100	621	100	607	100	557	100	507	100	403
105	621	105	613	105	563	105	513	105	410
110	621	110	619	110	570	110	520	110	419
115	621	115	621	115	577	115	528	115	428
120	621	120	621	120	585	120	537	120	439
125	621	125	621	125	594	125	547	125	450
130	621	130	621	130	604	130	557	130	464
130	621	130	621	135	615	135	570	135	478
130	697	130	651	140	628	140	583	140	495
135	706	135	661	145	641	145	598	145	513
140	717	140	672	150	657	150	615	150	534
145	728	145	685	155	674	155	634	155	557
150	741	150	699	160	692	160	654	160	582
155	755	155	714	165	713	165	677	165	611
160	771	160	731	170	736	170	703	170	642
165	788	165	750	175	762	175	731	175	677
170	807	170	771	180	790	180	762	180	716
175	828	175	794	185	821	185	797	185	759
180	851	180	820	190	856	190	835	190	807
185	877	185	848	195	894	195	878	195	861
190	905	190	879	200	937	200	925	200	920
195	937	195	914	205	984	205	978		
200	972	200	952	210	1036				
205	1010	205	995						
210	1052	210	1042						
215	1099	215	1093						
220	1151								

TABLE13 – (Continued)
34 EFPY Cooldown Curve Data Points Using 1996 App. G & ASME Code Case N-641
(without Uncertainties for Instrumentation Errors)

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
225	1208								
230	1271								
235	1341								
240	1418								
245	1503								
250	1598								
255	1702								
260	1817								
265	1944								
270	2085								
275	2240								
280	2412								

6 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

2. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

3. ASME Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1", January 17, 2000.
   [Sub Reference 1: ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.]
   [Sub Reference 2: ASME Boiler and Pressure Vessel Code, Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.]

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure." Dated December 1995, through 1996 Addendum.

5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

6. "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

7. WCAP-15423, "Analysis of Capsule V from Pacific Gas and Electric Company Diablo Canyon Unit 2 Reactor Vessel Radiation Surveillance Program," B Burgos, et. al., September 2000.

8. WCAP-13949, "Analysis of Capsule V Specimens and Dosimeters and analysis of Capsule Z Dosimeters From Duke Power Company McGuire Unit 1 Reactor Vessel Radiation Surveillance Program," E. Terek, et al., February 1994.

9. WCAP-14993 Analysis of Capsule Y from The Duke Power Company McGuire Unit 1 Reactor Vessel Radiation Surveillance Program," T.J. Laubham, et. al., December 1998.

10. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

11. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

12. WCAP-15253, "Duke Power Company Reactor Cavity Neutron Measurement Program for William B McGuire Unit 1 Cycle 12," J.D. Perock, et, al., July 1999.

APPENDIX A

INITIAL RT_{NDT} DETERMINATION FOR THE VESSEL FLANGES

2172 (Rev.B 1-65)

:—REVISED—:—:

BETHLEHEM STEEL CORPORATION

Bethlehem PLANT—METALLURGICAL DEPARTMENT

REPORT OF TESTS

REPORT No. 750-A

PURCHASER Combustion Engineering Inc.

June 23, 1969

Purchaser's Requisition	Date	Description	B.S.C. Order
47-64621	4/24/67	Closure Head Flange - Code B5002	3102-386
Contract 2167-NP		Dwg: SC-2067-2-0 - Item 2	

REQUIREMENTS

C	Mn	P	S	Si	Ni	Cr	V	Mo				
				ASTM	Spec.	A508-64	(S1) Code Case	1332				

Y.S. (.2%)	Tensile	Elong	Red.	V-N Charpys		Cust. Spec.
Tangential 50,000	80,000	18.0	38.0	@ +10°F 30 ft.lbs. Avg.		P3C9 (D)

CHEMICAL COMPOSITION

Heat No.	C	Mn	P	S	Si	Ni	Cr	V	Mo	Co
123X369	.21	.65	.008	.011	.28	.71	.38	.01	.60	.01
Check Analyses:										
T12	.20	.73	.01	.010	.29	.75	.39	/.01	.66	.013
T6	.21	.71	.01	.010	.29	.74	.39	/.01	.63	.012

TEST RESULTS

Identification No.	Specimen size	Y.S. (.2%)	Tensile	Elong	Red.
123X369VA1 T12	.505x2"	67,500	90,000	27.0	72.2
" T6	"	68,000	90,000	26.5	72.7

Test specimens were stress relieved at 1150°F for 40 hours - then furnace cooled to below 600°F

V-Notch	Charpys Ft.Lbs.	Broken Fibrous	@ +10°F L. E.
T12	119	66	.084
T12	102	55	.073
T12	116	66	.077
T6	112	61	.082
T6	.68	33	.052
T6	152	100	.092

AVERAGE ENERGY @ 10°F is > 100 ft lb

Subscribed and Sworn to before me this 23rd day of June, 1969.

James M. Mangan

James M. Mangan
Notary Public

My commission expires: Jan. 29th, 1973
City of Bethlehem
Northampton County

Magnetic Particle Inspection was performed in accordance with customer specification M & P 2.4.2.4 (B) paragraph 4.1 (Prod Method)

I CERTIFY THE ABOVE RESULTS TO BE CORRECT AS CONTAINED IN THE RECORDS OF THE COMPANY.

REMARKS:— Drop Weight test material was not cyclic tempered.

E. A. Reid
CHIEF METALLURGIST

1575°F. HELD FOR 24 HOURS - WATER QUENCHED
1220°F. " " 30 " - FURNACE COOLED

B-5002
2167

1175 (10-68)

BETHLEHEM STEEL CORPORATION
REPORT OF TRANSITION TEMPERATURE SURVEY

Report No. 750

June 10, 1969

CUSTOMER: Combustion Engineering, Inc.

CUSTOMER ORDER NO: 47-64621

BSC ORDER NO: 3102-386

TYPE FORGING: Closure Head Flange - 123X369VA1

I.S.T.T. = Minus 92°F

CODE	TEMP.	FT. LBS.	FT LBS AVERAGE	% FIBROUS	LATERAL EXPANSION	CODE	TEMP.	FT. LBS	FT LBS AVERAGE	% FIBROUS	LATERAL EXPANSION
T6	-120°F	3		0	.000						
T12		3		0	.000						
T6		6	4.0	5	.005						
T12	-90°F	54		21	.050						
T6		40		11	.034						
T12		7	33.7	0	.003						
T12	-60°F	41		16	.030						
T6		75		25	.057						
T12		86	67.3	31	.068						
T6	+20°F	122		61	.081						
T12		88		40	.069						
T6		124	111.3	64	.055						
T12	+60°F	150		100	.093						
T6		154		100	.095						
T12		148	150.7	100	.091						
T6	+120°F	154		100	.091						
T12		158		100	.094						
T6		146	156.0	82	.089						

Upper Shelf obtained @ 80°F & higher w/Energy > 100 ft-lbs

TRANSITION CURVE ATTACHED.

2172 (Rev.8 1-65)

BETHLEHEM STEEL CORPORATION
Bethlehem PLANT—METALLURGICAL DEPARTMENT

REPORT OF TESTS

REPORT No. 966

PURCHASER Combustion Engineering, Inc. 7/25 1968

Purchaser's Requisition	Date	Description	B.S.C. Order
47 - 64620	4/24/67	Vessel Flange--Item 1-Code B4701	3102-383
		Drawing: SC-2067-1- Rev.1	

REQUIREMENTS

C	Mn	P	S	Si	Ni	Cr	V	Mo

	Y.S. (.2%)	Tensile	Elong	Red.	V-N Charpys @ +10°F	Specification
Tangential	50000	80000	18.0	38.0	30 ft.lbs.avg.	P3C9D (Modified)

CHEMICAL COMPOSITION

Heat No.	C	Mn	P	S	Si	Ni	Cr	V	Mo	Co
122W201	.21	.58	.008	.010	.23	.71	.31	.01	.60	.01
Check Analyses										
T12	.23	.61	.01	.010	.25	.73	.32	.02	.59	.011
T6	.21	.61	.01	.009	.25	.73	.34	.02	.61	.011

TEST RESULTS

Identification No.		Specimen size	Y.S. (.2%)	Tensile	Elong	Red.
122W201VA1	T12	.505x2"	67500	89000	26.5	72.0
"	T6	"	63000	86500	27.0	73.1

Test specimens were tempered at 1125°F., for 40 hours. Then Furnace Cooled to below 600°F.

HEAT TREATMENT (OF FORGING)
1550°F., held for 30 hours - Water Quenched
1220°F., held for 40 hours - Furnace Cooled

Average Energy @ 10°F is 7 100 ft-lbs

Drop Weight test material was not Cyclic Tempered. Was given the same treatment as the forging.

V-Notch Charpys at +10°F

	Ft.Lbs.	Fibrous	L.E.
T12	100	58%	.080
T12	115	58%	.084
T12	105	55%	.079
T6	104	58%	.077
T6	112	52%	.085
T6	98	46%	.072

Magnetic Particle Inspection was performed in accordance with Spec. M & P 2.4.2.4 (c) paragraph 4.1 (Prod Method).

Subscribed and sworn to before me this 25th day of July, 1968.
Granville W. Arnold
Granville W. Arnold
Notary Public
My commission expires: Jan.20th,1969
City of Bethlehem-Northampton County

I CERTIFY THE ABOVE RESULTS TO BE CORRECT AS CONTAINED IN THE RECORDS OF THE COMPANY.

REMARKS:—

E.A.Reid
CHIEF METALLURGIST
Per R.P. Morrison

B-4701
2167
7c ± 1-5
V-70330

2172 (Rev B 1-65)

BETHLEHEM STEEL CORPORATION

Bethlehem PLANT—METALLURGICAL DEPARTMENT

Supplement

REPORT OF TESTS

Report No 966-A

Purchaser Combustion Engineering, Inc.

8/20 19 68

Purchaser's Requisition	Date	Description	B.S.C. Order
47 - 64620	4/24/67	Vessel Flange----Item 1---Code B4701	3102-383
		Drawing: SC-2067-1 Rev.1	

REQUIREMENTS

C	Mn	P	S	Si	Ni	Cr	V	Mo				

	Tensile	Elong.	Red.			

CHEMICAL COMPOSITION

Heat No.	C	Mn	P	S	Si	Ni	Cr	V	Mo

TEST RESULTS

Identification No	Specimen size		Tensile	Elong	Red.			
122W201VA1	Transition Temperature Survey							
Temp. Ft.Lbs	Ft.Lbs Avg.	Fibrous	L.E.	Temp.	Ft.Lbs.	Ft.Lbs. Avg.	Fibrous	L.E.
-120°F 4		0%	.002	+20°F	110		61%	.078
3		0%	.000		104		55%	.073
~~6~~	~~4.0~~	~~0%~~	~~.003~~		~~92~~	~~102.0~~	~~46%~~	~~.072~~
-80°F 23		11%	.017	+80°F	159		100%	.094
9		5%	.005		153		100%	.092
~~21~~	~~17.7~~	~~11%~~	~~.018~~		~~137~~	~~149.7~~	~~82%~~	~~.089~~
-40°F 80		31%	.062					
54		25%	.041					
~~97~~	~~77.0~~	~~43%~~	~~.074~~					

Upper Shelf is obtained @ 80°F w/ Energy > 100 ft-lbs

I.S.T.T. = Minus 69°F

Transition Curve Attached.

I CERTIFY THE ABOVE RESULTS TO BE CORRECT AS CONTAINED IN THE RECORDS OF THE COMPANY.

This report issued as a supplement to test report No. 966 dated July 25th, 1968. Please attach to subject test report.

E.A.REID
CHIEF METALLURGIST

Per

ENCLOSURE 5

WESTINGHOUSE WCAP-15201, REVISION 2

WCAP-15201, Revision 2

McGuire Unit 2
Heatup and Cooldown Limit Curves for Normal Operation

T. J. Laubham

SEPTEMBER 2002

Prepared by the Westinghouse Electric Company LLC
for the Duke Energy

Approved: _____
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PREFACE

This report has been technically reviewed and verified by:

J.H. Ledger _____

RECORD OF REVISION

Revision 0: Original Issue

Revision 1: An error was detected in the "OPERLIM" Computer Program that Westinghouse uses to generate pressure-temperature (PT) limit curves. This error potentially effects the heatup curves when the 1996 Appendix G Methodology is used in generating the PT curves. It has been determined that WCAP-15201 Rev. 0 was impacted by this error. Thus, this revision provides corrected curves from WCAP-15201 Rev. 0.

Note that only the heatup curves and associated data point tables have changed, however the 100 Deg.F/Hr. Heatup Rate was not affected by this error for McGuire Unit 2. The cooldown curves and data points remain valid and were not changed.

Revision 2: Revised PT Curves to account for the use of "Calculated" Neutron Fluence Projections and to included the full flange requirement from 10 CFR 50 Appendix G. Text and tables have been updated to account for the changes described above. Title was changed to remove reference to Code Case N-640.

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Figure 3 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App. G Methodology & ASME Code Case N-641 24

EXECUTIVE SUMMARY

This report provides the methodology and results of the generation of heatup and cooldown pressure temperature (PT) limit curves for normal operation of the McGuire Unit 2 reactor vessel. The PT curves were generated based on the latest available reactor vessel information and updated calculated fluences. The new McGuire Unit 2 heatup and cooldown pressure-temperature limit curves were generated using the "axial flaw" methodology of 1995 ASME Code, Section XI through the 1996 Addenda. In addition, the PT curves were developed using ASME Code Case N-641, which allows the use of the K_{Ic} methodology. The material with the highest adjusted reference temperature (ART) was the lower shell forging 04. The PT limit curves were generated for 34 EFPY using heatup rates of 60, 80 and 100°F/hr and cooldown rates of 0, 20, 40, 60 and 100°F/hr. These curves can be found in Figures 5-1 through 5-3.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials."[1] Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values ($IRT_{NDT} + \Delta RT_{NDT}$ + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface.

The heatup and cooldown curves documented in this report were generated using the most limiting ART values and the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2[2], "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves" with exception of the following: 1) **The fluence values used in this report are calculated fluence values (i.e. comply with Reg. Guide 1.190), not the best estimate fluence values.** 2) The K_{Ic} critical stress intensities are used in place of the K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-641[3] (which covers Code Cases N-640 and N-588). 3) The 1996 Version of Appendix G to Section XI[4] will be used rather than the 1989 version.

The purpose of this report is to present the calculations and the development of the McGuire Unit 2 heatup and cooldown curves for 34 EFPY. This report documents the calculated ART values and the development of the PT limit curves for normal operation. The PT curves herein were generated without instrumentation errors. The PT curves include a hydrostatic leak test limit curve from 2485 psig to 2000 psig, along with the pressure-temperature limits for the vessel flange region per the requirements of 10 CFR Part 50, Appendix G[5].

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan [6]. The beltline material properties of the McGuire Unit 2 reactor vessel are presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for four capsules (Capsules V, X, U and W) already removed from the McGuire Unit 2 reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 3. These CF values are summarized in Table 4.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1
Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the McGuire Unit 2 Reactor Vessel Materials

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange	---	---	1°F
Vessel Flange	---	---	-4°F
Intermediate Shell Forging 05	0.153	0.793	-4°F
Lower Shell Forging 04	0.15	0.88	-30°F
Intermediate to Lower Shell Girth Weld	0.039	0.724	-68°F
McGuire Unit 2 Surveillance Weld[b]	0.04	0.74	- -

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) The surveillance weld was made with the same weld wire and flux as the intermediate to lower shell girth weld (weld wire heat # 895075, Type Grau L.O. # LW320 Flux, Lot P46).

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 3 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases. Included in Table 2 are the Calculated fluence values for McGuire Unit 2. All capsule fluence values were determined using ENDF/B-VI cross-sections and followed the guidance in Regulatory Guide 1.190[10].

It should be noted that in the calculation of chemistry factor in Table 3, the ratio was not required since it was equal to 1.0. As for Temperature adjustments are concerned, the McGuire Unit 2 data does not require any adjustments since it is being applied to its own plant.

TABLE 2
Calculated Integrated Neutron Exposure of the Surveillance Capsules
@ McGuire Unit 2

Capsule	Fluence
McGuire Capsule Fluences[a]	
V	3.23×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.47×10^{19} n/cm^2, (E > 1.0 MeV
U	2.04×10^{19} n/cm^2, (E > 1.0 MeV)
Y[b]	2.08×10^{19} n/cm^2, (E > 1.0 MeV)
Z[b]	2.41×10^{19} n/cm^2, (E > 1.0 MeV)
W	3.07×10^{19} n/cm^2, (E > 1.0 MeV)

NOTES:

(a) Re-Calculated in WCAP-15334[7].

(b) Dosimeters only.

TABLE 3
Calculation of Chemistry Factors using McGuire Unit 2
Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell Forging 05 (Axial)	V	0.323	0.689	58 64	40.40	0.475
	X	1.47	1.11	91.12	101.14	1.23
	U	2.04	1.19	84.14	100.13	1.42
	W	3.07	1.30	130.33	169.43	1.69
Intermediate Shell Forging 05 (Tangential)	V	0.323	0.689	68.97	47.52	0.475
	X	1.47	1.11	98.28	109.09	1.23
	U	2.04	1.19	91.18	108.50	1.42
	W	3.07	1.30	102.03	132.64	1.69
				SUM	808.85	9.63
	$CF_{Forging\ 05} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (808.85) \div (9.63) = \mathbf{84.0°F}$					
Circumferential Weld Seam[d]	V	0.323	0 689	38.51	26.53	0.475
	X	1.47	1.11	35.93	39.88	1.23
	U	2.04	1.19	23.81	28.33	1.42
	W	3 07	1.30	43.76	56.89	1.69
				SUM	151.63	4.815
	$CF_{S/P\ Weld} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (151.63) \div (4.815) = \mathbf{31.5°F}$					

Notes:
(a) f = Calculated fluence (x 10^{19} n/cm^2, E > 1.0 MeV). See WCAP-15334[7].
(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$
(c) ΔRT_{NDT} values are measured, See WCAP-14799[8], Appendix B.
(d) Per WCAP-15201[11], the ratio procedure was shown not to be applicable (i.e. Ratio = 1.0).

TABLE 4
Summary of the McGuire Unit 2 Reactor Vessel Beltline Material Chemistry Factors

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Intermediate Shell Forging 05	117°F	84.0°F
Lower Shell Forging 04	115.8°F	(a)
Beltline Region Circ. Weld Metal	52.7°F	31.5°F

Notes:
(a) No surveillance material for forging 04, thus Position 2.1 does not apply.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 OVERALL APPROACH

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_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1"[3 & 4] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 METHODOLOGY FOR PRESSURE-TEMPERATURE LIMIT CURVE DEVELOPMENT

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

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

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \quad (5)$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \quad (6)$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[2] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code 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 the 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 inner diameter. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the ΔT (temperature) developed during cooldown results in a higher value of K_{Ic} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist so that the increase in K_{Ic} 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 and, 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 ensures conservative operation of the system for the entire cooldown period.

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 wall. The heatup results in compressive stresses at the inside surface that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{Ic} for the 1/4T crack during heatup is lower than the K_{Ic} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist so that the effects of compressive thermal stresses and lower K_{Ic} values 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 ensure 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 the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface 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 therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate 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 by constructing a composite curve 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 wherein, 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.

3.3 CLOSURE HEAD/VESSEL FLANGE REQUIREMENTS

10 CFR Part 50, Appendix G[5] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which is 621 psig for McGuire Unit 2. The limiting unirradiated RT_{NDT} of 1°F occurs in the closure head flange of the McGuire Unit 2 reactor vessel, so the minimum allowable temperature of this region is 121°F at pressures greater than 621 psig. This limit is shown in Figures 1 and 2 wherever applicable.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \quad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = CF * f^{(0.28 - 0.10 \log f)} \quad (8)$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4T or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth\ x)} = f_{surface} * e^{(-0.24x)} \quad (9)$$

where x inches (vessel beltline thickness is 8.465 inches) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis Group evaluated the vessel fluence projections and the results of the **calculated peak fluence values** at various azimuthal locations on the vessel clad/base metal interface are presented in Table 5. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves". Table 6 contains the 1/4T and 3/4T calculated fluences and fluence factors, per the Regulatory Guide 1.99, Revision 2, used to calculate the ART values for all beltline materials in the McGuire Unit 2 reactor vessel.

TABLE 5

Summary of the Peak Pressure Vessel Neutron Fluence Values at the Clad/Base Metal Interface at 34 EFPY (n/cm^2, E > 1.0 MeV)

0°	15°	30°	45°
1.14×10^{19}	1.68×10^{19}	1.65×10^{19}	1.85×10^{19}

TABLE 6

Summary of the Peak Pressure Vessel Neutron Fluence Values
at 34 EFPY used for the Calculation of ART Values
(n/cm^2, E > 1.0 MeV)

Material	Surface	¼ T	¾ T
Intermediate Shell Forging 05	1.85×10^{19}	1.11×10^{19}	0.403×10^{19}
Lower Shell Forging 04	1.85×10^{19}	1.11×10^{19}	0.403×10^{19}
Intermediate to Lower Shell Cirumferential Weld	1.85×10^{19}	1.11×10^{19}	0.403×10^{19}

Contained in Table 7 is a summary of the fluence factor (FF) values used in the calculation of adjusted reference temperatures for the McGuire Unit 2 reactor vessel beltline materials for 34 EFPY.

TABLE 7
Summary of the Calculated Fluence Factors used for the Generation of the 34 EFPY Heatup and Cooldown Curves

Material	¼ T f (n/cm2, E > 1.0 MeV)	¼ T FF[a]	¾ T f (n/cm2, E > 1.0 MeV)	¾ T FF[b]
Intermediate Shell Forging 05	1.11×10^{19}	1.03	0.403×10^{19}	0.75
Lower Shell Forging 04	1.11×10^{19}	1.03	0.403×10^{19}	0.75
Intermediate to Lower Shell Cirumferential Weld	1.11×10^{19}	1.03	0.403×10^{19}	0.75

Notes:
(a) Fluence Factor at the 1/4T vessel thickness location.
(b) Fluence Factor at the 3/4T vessel thickness location.

Margin is calculated as, $M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$. The standard deviation for the initial RT_{NDT} margin term, is σ_I 0°F when the initial RT_{NDT} is a measured value, and 17°F when a generic value is available. The standard deviation for the ΔRT_{NDT} margin term, σ_Δ, is 17°F for plates or forgings, and 8.5°F for plates or forgings when surveillance data is used. For welds, σ_Δ is equal to 28°F when surveillance capsule data is not used, and is 14°F (half the value) when credible surveillance capsule data is used. σ_Δ need not exceed 0.5 times the mean value of ΔRT_{NDT}.

Contained in Tables 8 and 9 are the calculations of the 34 EFPY ART values used for generation of the heatup and cooldown curves.

TABLE 8
Calculation of the ART Values for the 1/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT}[c] (°F)	Margin (°F)	ART[b] (°F)
Intermediate Shell Forging 05	Position 1.1	117.0	1.03	-4	120.5	34	151
(Heat # 526840)	Position 2.1	84.0	1.03	-4	86.5	17	100
Lower Shell Forging 04 (Heat # 411337/11)	Position 1.1	115.8	1.03	-30	119.3	34	123
Intermediate to Lower Shell	Position 1.1	52.7	1.03	-68	54.3	54.3	41
Circumferential Weld Seam W05	Position 2.1	31.5	1.03	-68	32.4	28	-8

Notes:
(a) Initial RT_{NDT} values are measured values
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF

TABLE 9
Calculation of the ART Values for the 3/4T Location @ 34 EFPY

Material	RG 1.99 R2 Method	CF (°F)	FF	IRT_{NDT}[a] (°F)	ΔRT_{NDT}[c] (°F)	Margin (°F)	ART[b] (°F)
Intermediate Shell Forging 05	Position 1.1	117.0	0.750	-4	87.8	34	118
(Heat # 526840)	Position 2.1	84.0	0.750	-4	63	17	76
Lower Shell Forging 04 (Heat # 411337/11)	Position 1.1	115.8	0.750	-30	86.9	34	91
Intermediate to Lower Shell	Position 1.1	52.7	0.750	-68	39.5	39.5	11
Circumferential Weld Seam W05	Position 2.1	31.5	0.750	-68	23.6	23.6	-21

Notes:
(a) Initial RT_{NDT} values are measured values.
(b) ART = Initial RT_{NDT} + ΔRT_{NDT} + Margin (°F)
(c) ΔRT_{NDT} = CF * FF

The Lower Shell Forging 04 is the limiting beltline material for all the PT limit curves to be generated. Contained in Table 10 is a summary of the limiting ARTs to be used in the generation of the McGuire Unit 2 reactor vessel PT limit curves. These limiting curves will be presented in Section 5.

TABLE 10
Summary of the Limiting ART Values Used in the
Generation of the McGuire Unit 2 Heatup/Cooldown Curves

¼ T Limiting ART	¾ T Limiting ART
123	91

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Sections 3.0 and 4.0 of this report. This approved methodology is also presented in WCAP-14040-NP-A, Revision 2 with exception to those items discussed in Section 1 of this report.

Figures 1 and 2 present the limiting heatup curves without margins for possible instrumentation errors using heatup rates of 60, 80 and 100°F/hr applicable for the first 34 EFPY. These curves were generated using a combination of the1996 ASME Code Section XI, Appendix G with the limiting ART values and the ASME Code Case N-641. Figure 3 presents the limiting cooldown curves without margins for possible instrumentation errors using cooldown rates of 0, 20, 40, 60 and 100°F/hr applicable for 34 EFPY. Again, these curves were generated using a combination of the1996 ASME Code Section XI, Appendix G with the limiting ART values and the ASME Code Case N-641. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit line shown in Figures 1 through 3. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed below in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 1 and 2. The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-641[3] (approved in February 1999) as follows:

$$1.5\ K_{Im} < K_{Ic}$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734\ e^{[0.02\,(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature.

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 5. The pressure-temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 4.0 of this report. For the heatup and cooldown curves without margins for instrumentation errors, the minimum temperatures for the in service hydrostatic leak tests for the McGuire Unit 2 reactor vessel at 34 EFPY is 184°F. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve constitutes the limit for core operation for the reactor vessel.

Figures 1 through 3 define all of the above limits for ensuring prevention of nonductile failure for the McGuire Unit 2 reactor vessel for 34 EFPY. The data points used for the heatup and cooldown pressure-temperature limit curves shown in Figures 1 through 3 are presented in Tables 11, 12 and 13.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

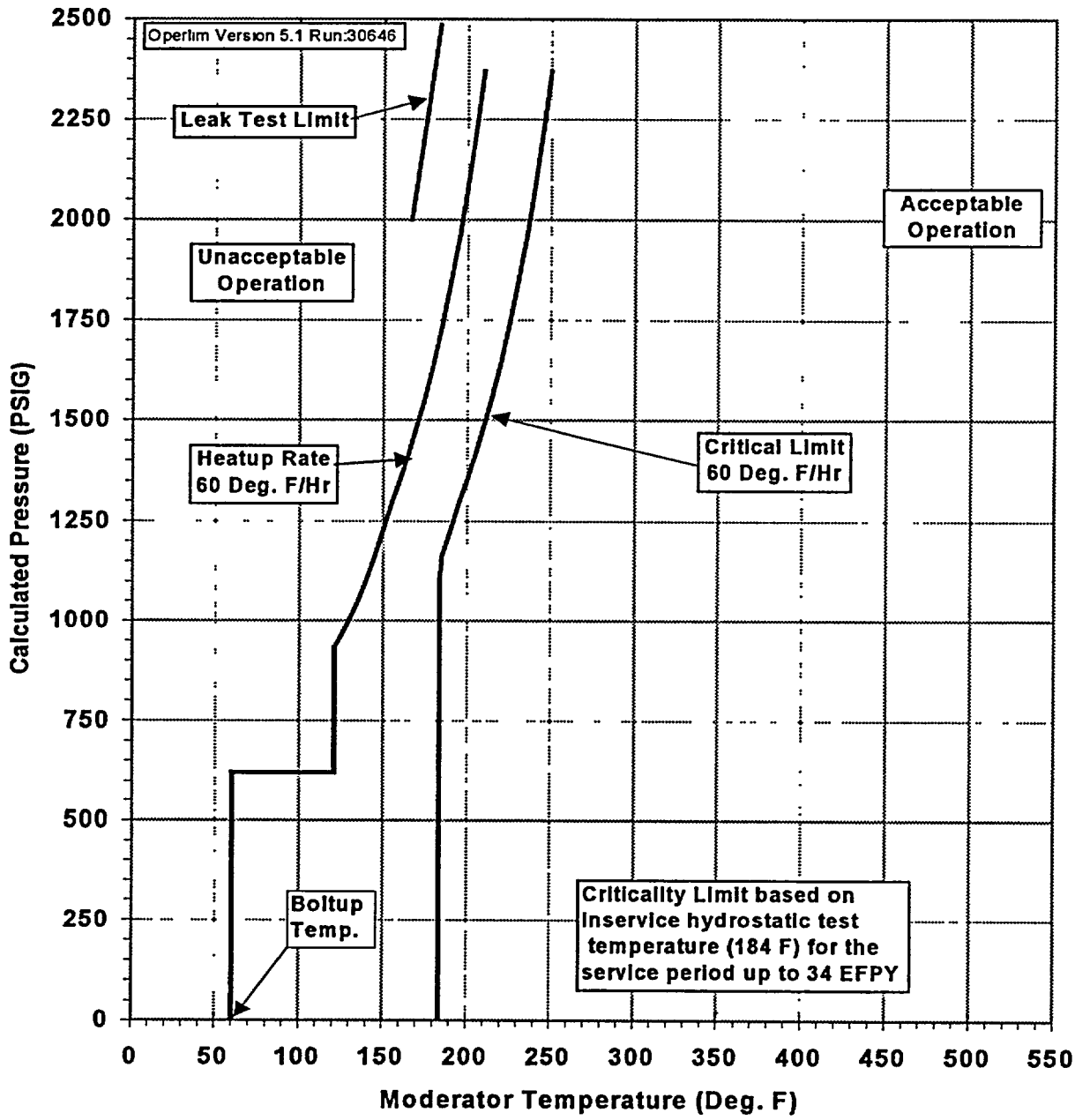


Figure 1 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

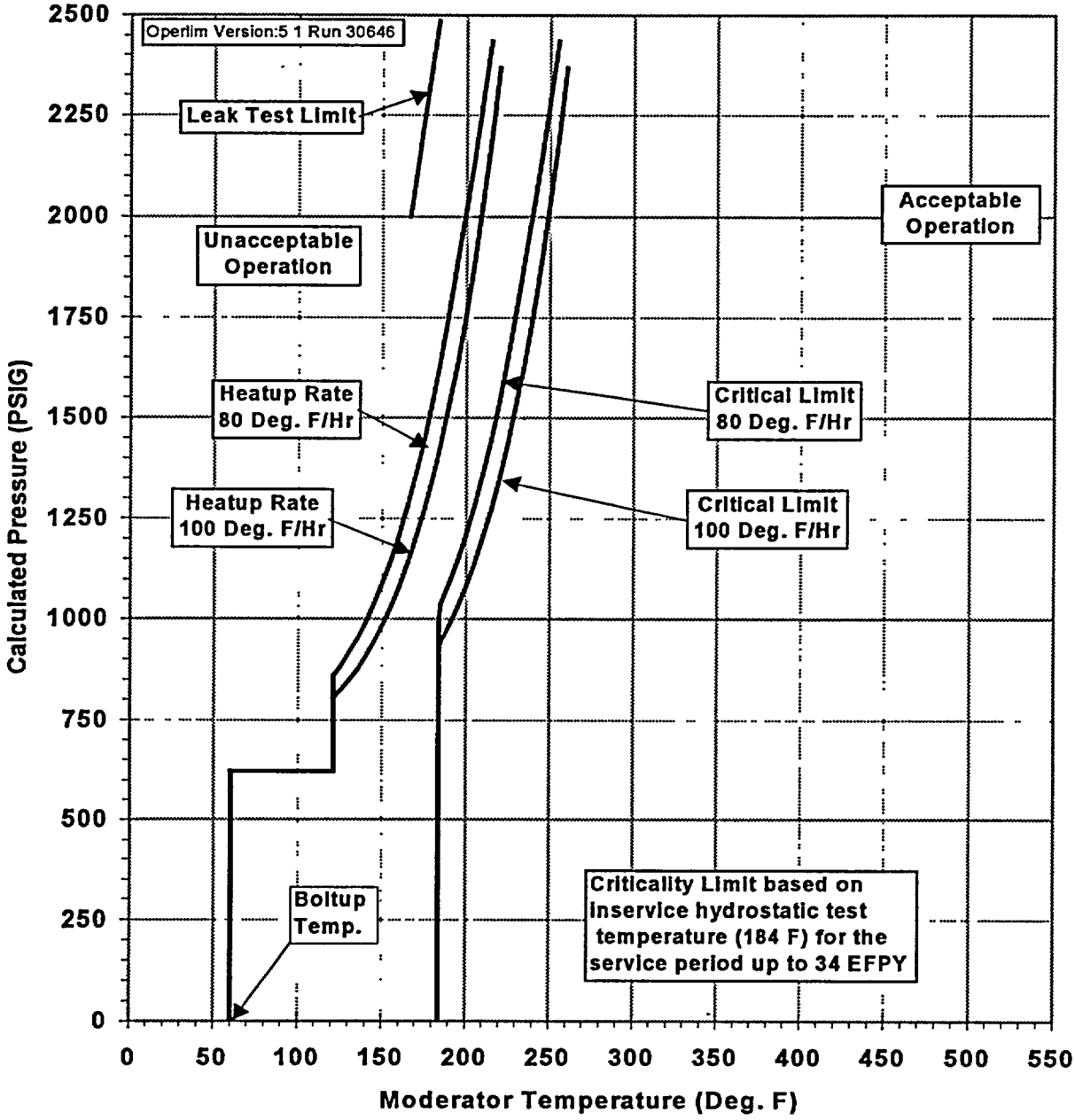


Figure 2 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T,123°F
3/4T, 91°F

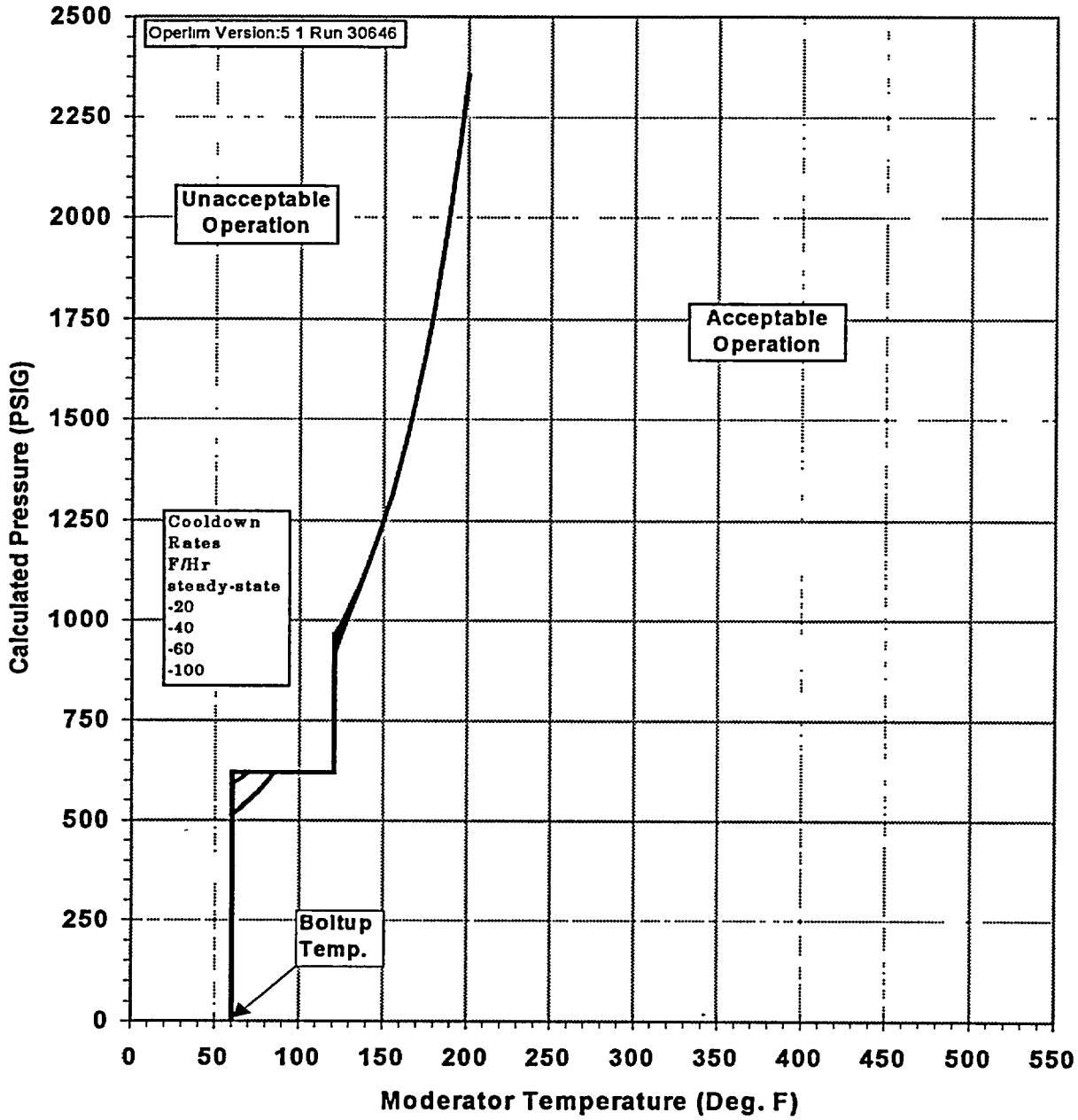


Figure 3 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

TABLE 11

34 EFPY, 60°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641

(without Uncertainties for Instrumentation Errors)

60 °F/hr		Criticality Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	184	0	167	2000
60	621	184	621	184	2485
65	621	184	621		
70	621	184	621		
75	621	184	621		
80	621	184	621		
85	621	184	621		
90	621	184	621		
95	621	184	621		
100	621	184	621		
105	621	184	621		
110	621	184	621		
115	621	184	621		
120	621	184	621		
121	621	184	932		
121	932	184	962		
125	962	184	1004		
130	1004	184	1051		
135	1051	184	1103		
140	1103	185	1161		
145	1161	190	1226		
150	1226	195	1296		
155	1296	200	1353		
160	1353	205	1416		
165	1416	210	1486		
170	1486	215	1562		
175	1562	220	1646		
180	1646	225	1739		
185	1739	230	1842		
190	1842	235	1955		
195	1955	240	2080		
200	2080	245	2218		
205	2218	250	2370		
210	2370				

TABLE 12

34 EFPY, 80 and 100°F/hr Heatup Curve Data Points Using 1996 App. G & ASME Code Case N-641

(without Uncertainties for Instrumentation Errors)

80 °F/hr		Criticality Limit		100 °F/hr		Critical Limit		Leak Test Limit	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	184	0	60	0	184	0	167	2000
60	621	184	621	60	621	184	621	184	2485
65	621	184	621	65	621	184	621		
70	621	184	621	70	621	184	621		
75	621	184	621	75	621	184	621		
80	621	184	621	80	621	184	621		
85	621	184	621	85	621	184	621		
90	621	184	621	90	621	184	621		
95	621	184	621	95	621	184	621		
100	621	184	621	100	621	184	621		
105	621	184	621	105	621	184	621		
110	621	184	621	110	621	184	621		
115	621	184	621	115	621	184	621		
120	621	184	621	120	621	184	621		
121	621	184	859	121	621	184	805		
121	859	184	881	121	805	184	822		
125	881	184	913	125	822	184	846		
130	913	184	950	130	846	184	874		
135	950	184	991	135	874	184	906		
140	991	185	1037	140	906	185	943		
145	1037	190	1089	145	943	190	984		
150	1089	195	1146	150	984	195	1030		
155	1146	200	1210	155	1030	200	1081		
160	1210	205	1281	160	1081	205	1139		
165	1281	210	1360	165	1139	210	1203		
170	1360	215	1447	170	1203	215	1273		
175	1447	220	1544	175	1273	220	1352		
180	1544	225	1650	180	1352	225	1439		
185	1650	230	1769	185	1439	230	1536		
190	1769	235	1899	190	1536	235	1642		
195	1899	240	2017	195	1642	240	1760		
200	2017	245	2143	200	1760	245	1890		
205	2143	250	2282	205	1890	250	2034		
210	2282	255	2435	210	2034	255	2193		
215	2435			215	2193	260	2368		
				220	2368				

TABLE 13
34 EFPY Cooldown Curve Data Points Using 1996 App. G & ASME Code Case N-641
(without Uncertainties for Instrumentation Errors)

Steady State		20°F/hr.		40°F/hr.		60°F/hr.		100°F/hr.	
T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)	T (°F)	P (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	621	60	621	60	591	60	512
65	621	65	621	65	621	65	605	65	530
70	621	70	621	70	621	70	621	70	549
75	621	75	621	75	621	75	621	75	570
80	621	80	621	80	621	80	621	80	593
85	621	85	621	85	621	85	621	85	620
90	621	90	621	90	621	90	621	90	621
95	621	95	621	95	621	95	621	95	621
100	621	100	621	100	621	100	621	100	621
105	621	105	621	105	621	105	621	105	621
110	621	110	621	110	621	110	621	110	621
115	621	115	621	115	621	115	621	115	621
120	621	120	621	120	621	120	621	120	621
121	621	121	621	121	621	121	621	121	621
121	963	121	947	121	934	121	924	121	918
125	993	125	980	125	970	125	963		
130	1034	130	1025	130	1019	130	1018		
135	1080	135	1075	135	1074				
140	1130								
145	1185								
150	1247								
155	1315								
160	1390								
165	1473								
170	1564								
175	1665								
180	1777								
185	1901								
190	2037								
195	2188								
200	2355								

6 REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

2. WCAP-14040-NP-A, Revision 2, "Methodology used to Develop Cold Overpressure Mitigating system Setpoints and RCS Heatup and Cooldown Limit Curves", J.D. Andrachek, et. al., January 1996.

3. ASME Code Case N-641, "Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements Section XI, Division 1", January 17, 2000.
[*Sub Reference 1: ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.*]
[*Sub Reference 2: ASME Boiler and Pressure Vessel Code, Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.*]

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure." Dated December 1995, through 1996 Addendum.

5. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D.C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

6. "Fracture Toughness Requirements", Branch Technical Position MTEB 5-2, Chapter 5.3.2 in Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, NUREG-0800, 1981.

7. WCAP-15334, "Duke Power Company Reactor Cavity Neutron Measurement Program for William B McGuire Unit 2 Cycle 12," A.H. Fero, November 1999.

8. WCAP-14799, "Analysis of Capsule W From The Duke Power Company McGuire Unit 2 Reactor Vessel Radiation Surveillance Program," E. Terek, et. al., March 1997.

9. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels."

10. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

11. WCAP-15201, Revision 1, "McGuire Unit 2 Heatup and Cooldown Curves for Normal Operation Using Code Case N-640," J. H. Ledger, April 2001.

ENCLOSURE 6

ASME CODE CASE N-641

Approval Date: January 17, 2000

See Numeric Index for expiration and any reaffirmation dates

Case N-641
Alternative Pressure-Temperature Relationship and Low Temperature Overpressure Protection System Requirements
Section XI, Division 1

Inquiry: What alternatives to Appendix G-2215 may be used for determination of pressure-temperature relationships and low temperature overpressure protection system effective temperatures and allowable pressures?

Reply: It is the opinion of the Committee that, as an alternative to Appendix G-2215, the following may be used.

-1000 INTRODUCTION

-1100 Scope

This Case presents alternative procedures for calculating pressure-temperature relationships and low temperature overpressure protection (LTOP) system effective temperatures and allowable pressures. These procedures take into account alternative fracture toughness properties, circumferential and axial reference flaws, and plant-specific LTOP effective temperature calculations.

-2215 Allowable Pressure

-2215.1 Pressure-Temperature Relationship. The equations below provide the basis for determination of the allowable pressure at any temperature at the depth of the postulated defect during Service Conditions for which Level A and Level B Service Limits are specified. In addition to the conservatism of these assumptions, it is recommended that a factor of 2 be applied to the calculated K_I values produced by primary stresses. In shell and head regions remote from discontinuities, the only significant loadings are: (1) general primary membrane stress due to pressure; and (2) thermal stress due to thermal gradient through the thickness during startup and shutdown. Therefore, the requirement to be satisfied and from which the allowable pressure for any assumed rate of temperature change can be determined is:

$$2K_{Im} + K_{It} < K_{Ic} \qquad (1)$$

throughout the life of the component at each temperature with K_{Im} from G-2214.1, K_{It} from G-2214.3, and K_{Ic} from Fig. G-2210-1.

The allowable pressure at any temperature shall be determined as follows.

(a) For the startup condition,

(1) consider postulated defects in accordance with G-2120;

(2) perform calculations for thermal stress intensity factors due to the specified range of heat-up rates from G-2214.3;

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding locations of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location. The allowable pressure-temperature relationship is the minimum pressure at any temperature determined from

(a) the calculated steady-state ($K_{It} = 0$) results for the ¼ thickness inside surface postulated defects using the equation:

$$P = \frac{K_{Ic}}{2M_m}\left(\frac{t}{R_i}\right)$$

(b) the calculated results from all vessel beltline materials for the heatup stress intensity factors using the corresponding ¼ thickness outside-surface postulated defects and the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left(\frac{t}{R_i}\right)$$

(b) For the cooldown condition;

(1) consider postulated defects in accordance with G-2120;

(2) perform calculations for thermal stress intensity factors due to the specified range of cooldown rates from G-2214.3,

(3) calculate the K_{Ic} toughness for all vessel beltline materials from G-2212 using temperatures and RT_{NDT} values for the corresponding location of interest; and

(4) calculate the pressure as a function of coolant inlet temperature for each material and location using the equation:

$$P = \frac{K_{Ic} - K_{It}}{2M_m}\left(\frac{t}{R_i}\right)$$

The allowable pressure-temperature relationship is the minimum pressure at any temperature, determined from all vessel beltline materials for the cooldown stress intensity factors using the corresponding ¼ thickness inside-surface postulated defects.

-2215.2 Low Temperature Overpressure Protection System. Plants having LTOP systems may use the following temperature and pressure conditions to provide protection against failure during reactor startup and shutdown operation due to low temperature overpressure events that have been classified Service Level A or B.

(a) LTOP System Effective Temperature. The LTOP system effective temperature T_e is the temperature at or above which the safety relief valves provide adequate protection against nonductile failure. LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (2) below. Alternatively, LTOP systems shall be effective below the higher temperature determined in accordance with (1) and (3) below.

(1) a coolant temperature[1] of 200°F;

(2) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is defined for inside axial surface flaws as RT_{NDT} + 40°F, and T_e is defined for inside circumferential surface flaws as RT_{NDT} − 85°F;

(3) a coolant temperature[1] corresponding to a reactor vessel metal temperature[2], for all vessel beltline materials, where T_e is calculated on a plant specific basis for the axial and circumferential reference flaws using the following equation:

$$T_e = RT_{NDT} + 50 \ln [((F \cdot M_m (pR_i / t)) - 33.2) / 20.734]$$

where

F = 1.1, accumulation factor for safety relief valves
M_m = the value of M_m determined in accordance with G-2214.1
p = vessel design pressure, ksi
R_i = vessel inner radius, in
t = vessel wall thickness, in.

(b) LTOP System Allowable Pressure. LTOP systems shall limit the maximum pressure in the vessel to 100% of the pressure determined to satisfy Eq. (1) if K_{Ic} is used for determination of allowable pressure, or 110% of the pressure determined to satisfy Eq. (1) if K_{Ia} is used (as an alternative to K_{Ic}) for determination of allowable pressure.

[1] The coolant temperature is the reactor coolant inlet temperature

[2] The vessel metal temperature is the temperature at a distance one-fourth of the vessel section thickness from the clad-base-metal interface in the vessel beltline region RT_{NDT} is the highest adjusted reference temperature, for weld or base metal in the beltline region, at a distance one-fourth of the vessel section thickness from the clad-base-metal interface as determined in accordance with Regulatory Guide 1 99, Rev 2.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Table 3.3 2-1 (page 3 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	3[b][c]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≤ 120[d] psi	100[d] psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2[e]	2 trains	I	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - High High (P-14)	1,2[e]	3 per SG	J	SR 3.3.2.1 SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.5 SR 3.3.2.6 SR 3.3.2.8 SR 3.3.2.9	≤ 85.6%	83.9%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. T_{avg}-Low	1,2[e]	1 per loop	J	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8	≥ 551°F	553°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
e. Doghouse Water Level-High High	1,2[e]	2 per train per Doghouse	L,M	SR 3.3.2.1 SR 3.3.2.7	≤ 13 inches	12 inches
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA	NA
b. SG Water Level - Low Low	1,2,3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.8 SR 3.3.2.9	≥ 15%	16.7%

(continued)

(b) Except when all MSIVs are closed and de-activated.
(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when ~~Safety~~ ~~Injection~~ Steam Line Isolation on Steam Line Pressure-Low is not blocked.
(d) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

Figure 3.4.3-2, Figure 3.4.3-3, or Figure 3.4.3-4

LCO 3.4.3 RCS pressure and RCS temperature shall be limited during RCS heatup and cooldown, criticality, and inservice leak and hydrostatic testing in accordance with:

a. A maximum heatup rate as specified in Figure 3.4.3-1;

b. A maximum cooldown rate as specified in ~~Figure 3.4.3-2~~; and

c. A maximum temperature change of ≤ 10°F in any 1 hour period during inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves.

Figure 3.4.3-5 or Figure 3.4.3-6

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits.	30 minutes
	AND	
	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5 with RCS pressure < 500 psig.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ----------NOTE---------- Required Action C.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ----------	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within limits.	30 minutes

MATERIAL PROPERTY BASIS
LIMITING MATERIALS; LOWER SHELL LONGITUDINAL WELDS 3-442A and LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY

¼-t, 149.5 deg. F
¾-t, 102.0 deg. F

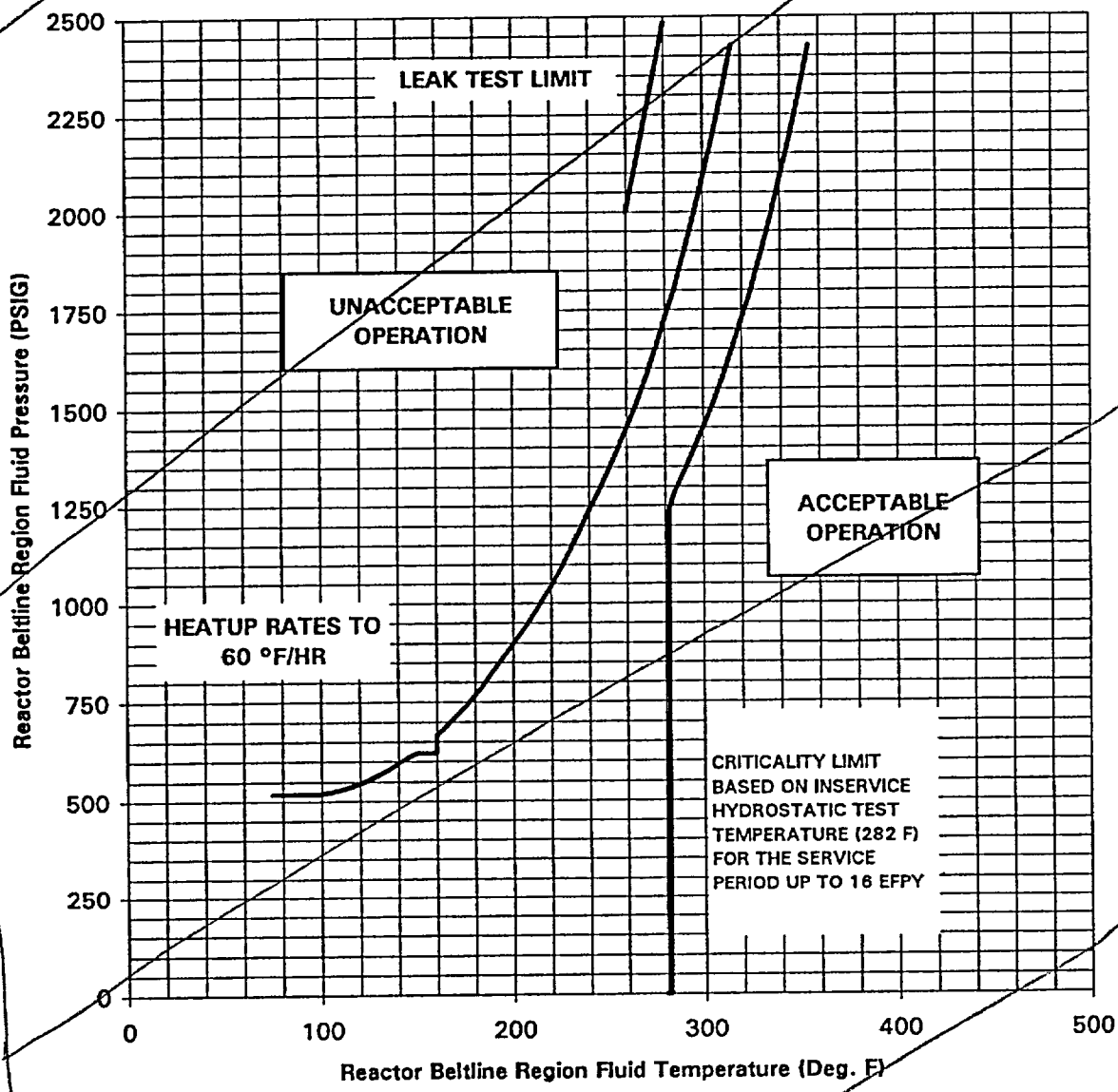
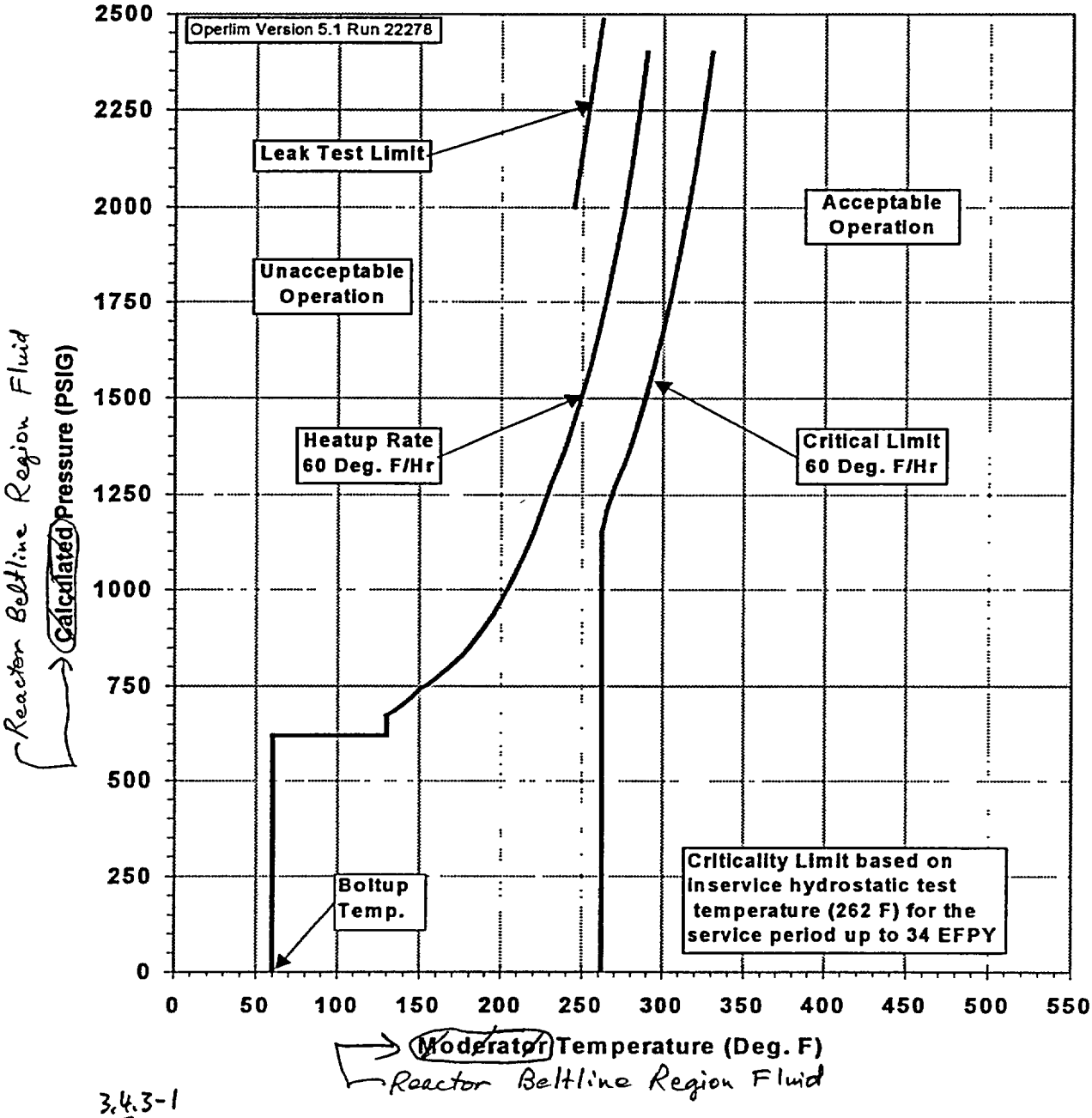


Figure 3.4.3-1 RCS Heatup Limitations
(UNIT 1 ONLY)
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the first 16 EFPY

Replaced by next two figures

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

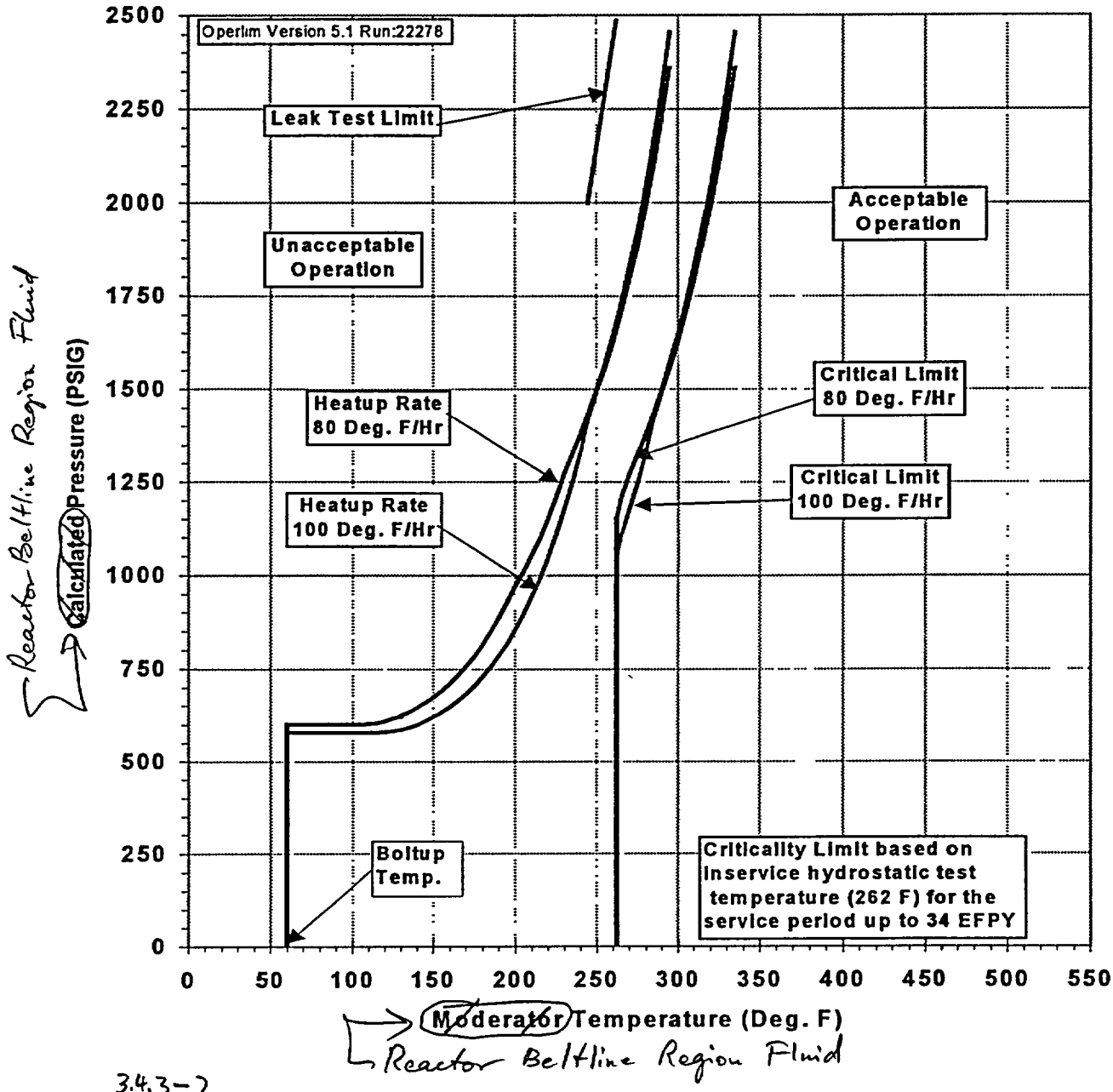


3.4.3-1

Figure 1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F



3.4.3-2

Figure 2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

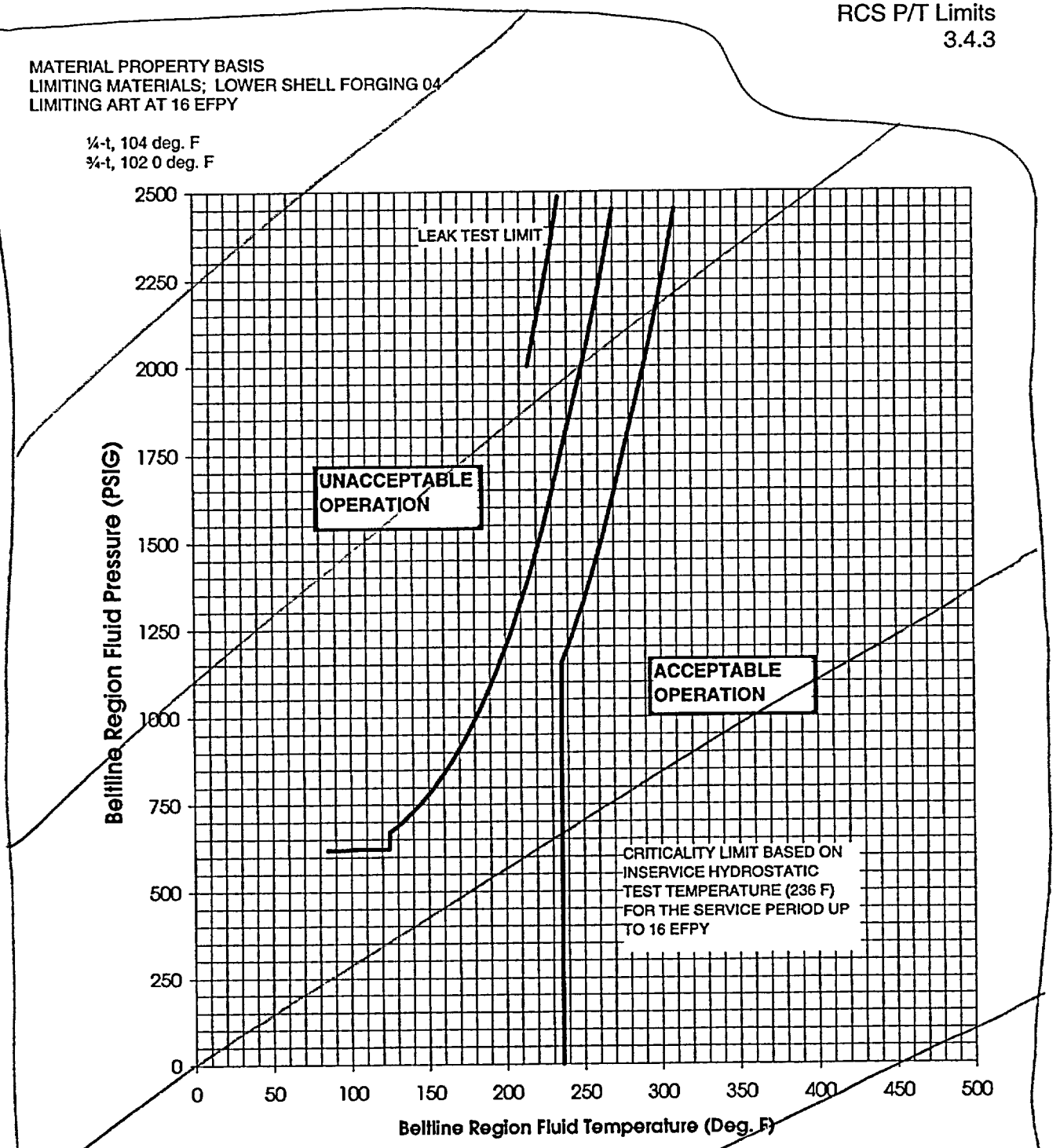
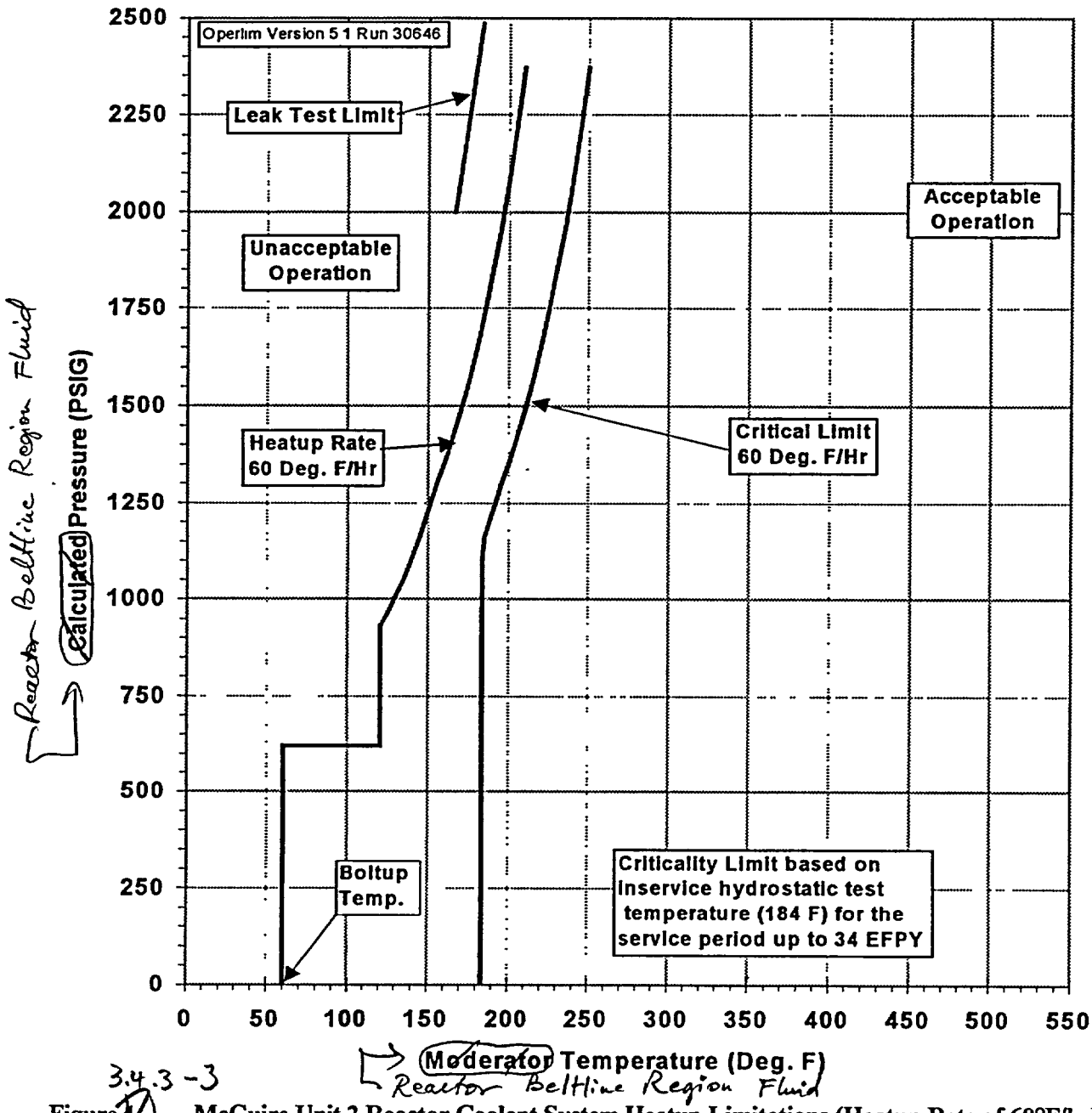


Figure 3.4.3-1 RCS Heatup Limitations
(Unit 2 Only)
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the First 16 EFPY

Replaced by next two figures

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

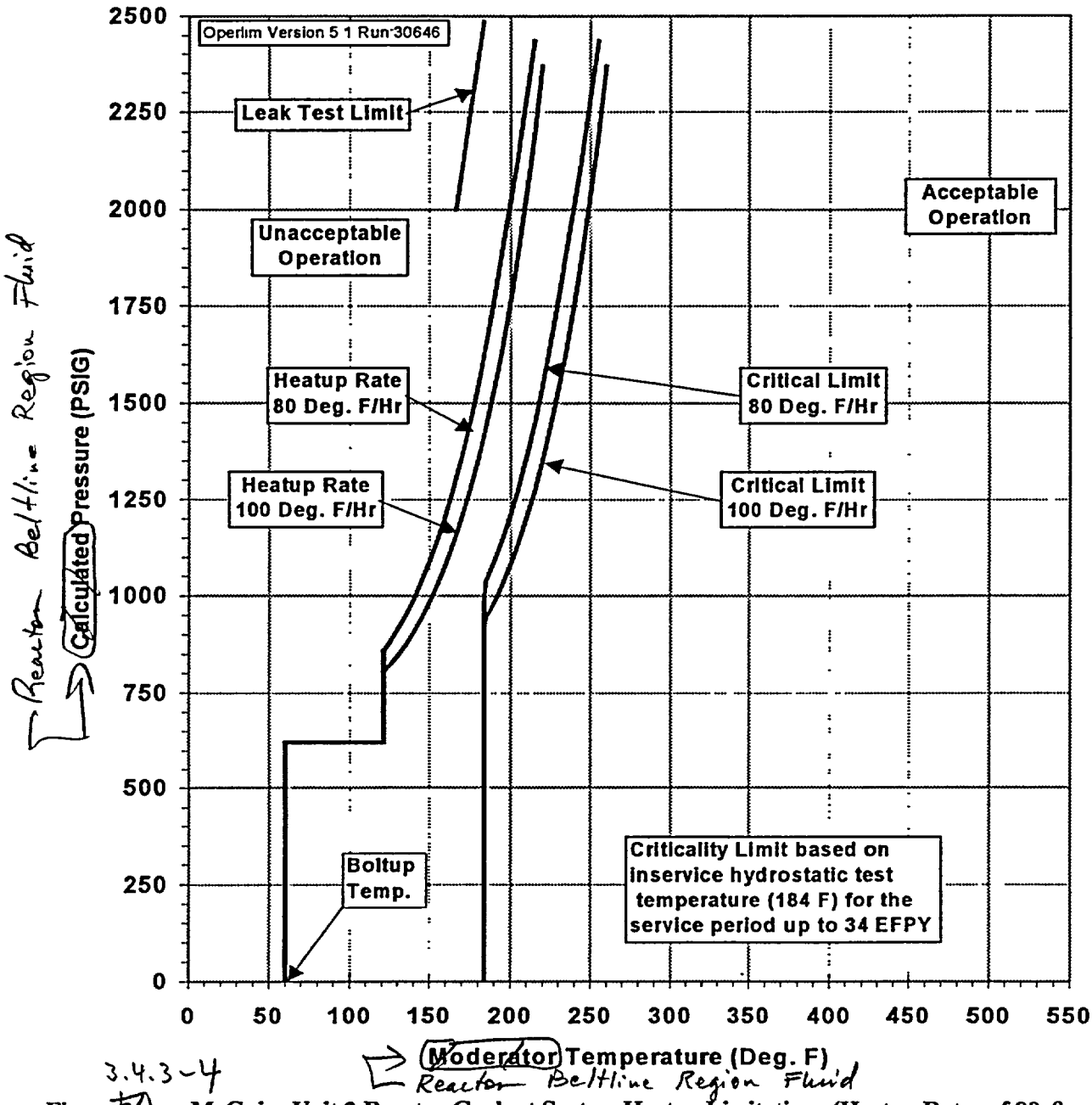


3.4.3-3

Figure 1 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F



3.4.3-4

Figure 2 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS
LIMITING MATERIALS; LOWER SHELL LONGITUDINAL WELDS 3-442A and LOWER SHELL PLATE B5013-2

LIMITING ART AT 16 EFPY

¼-t, 149 5 deg. F
¾-t, 102 0 deg. F

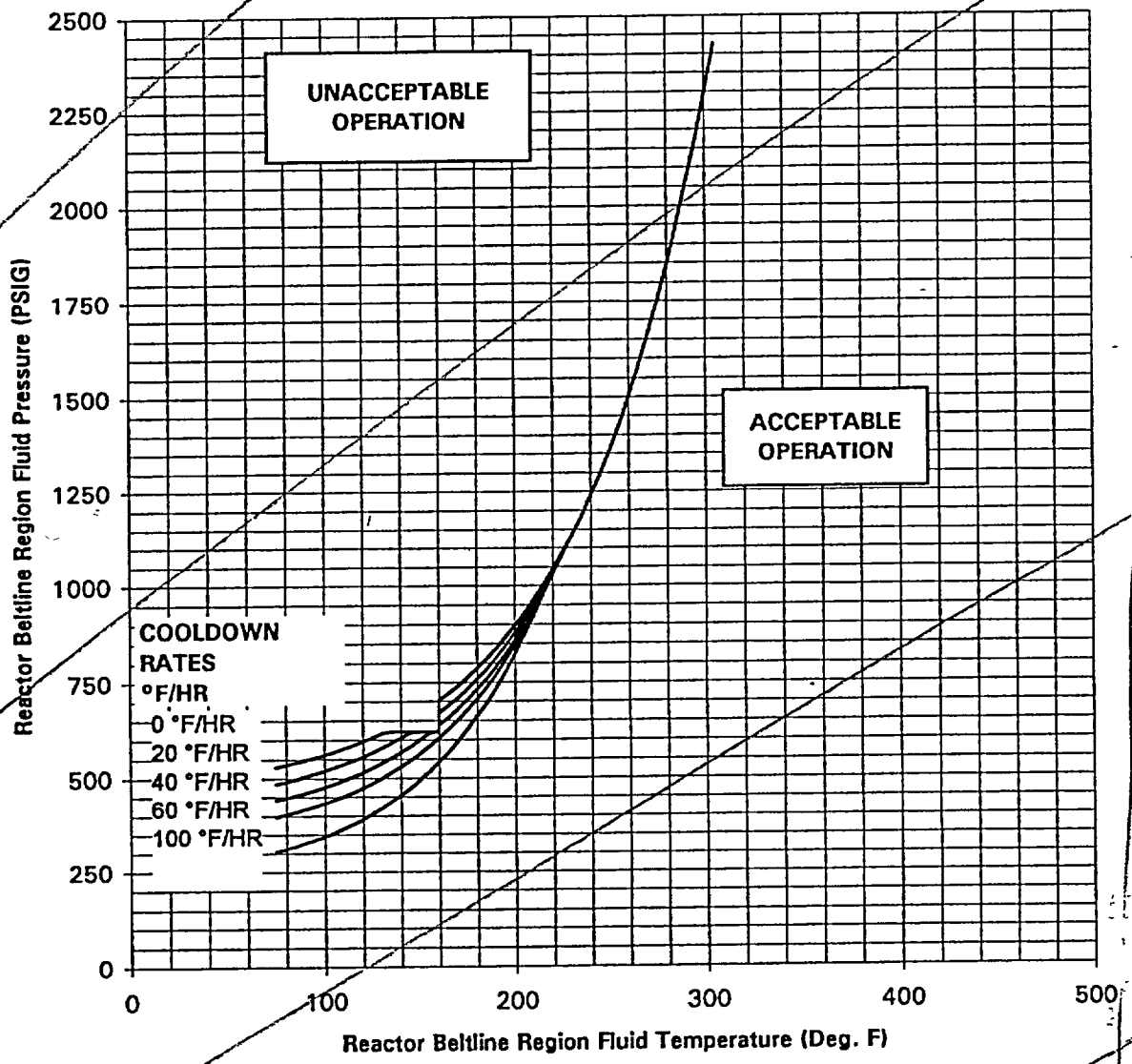
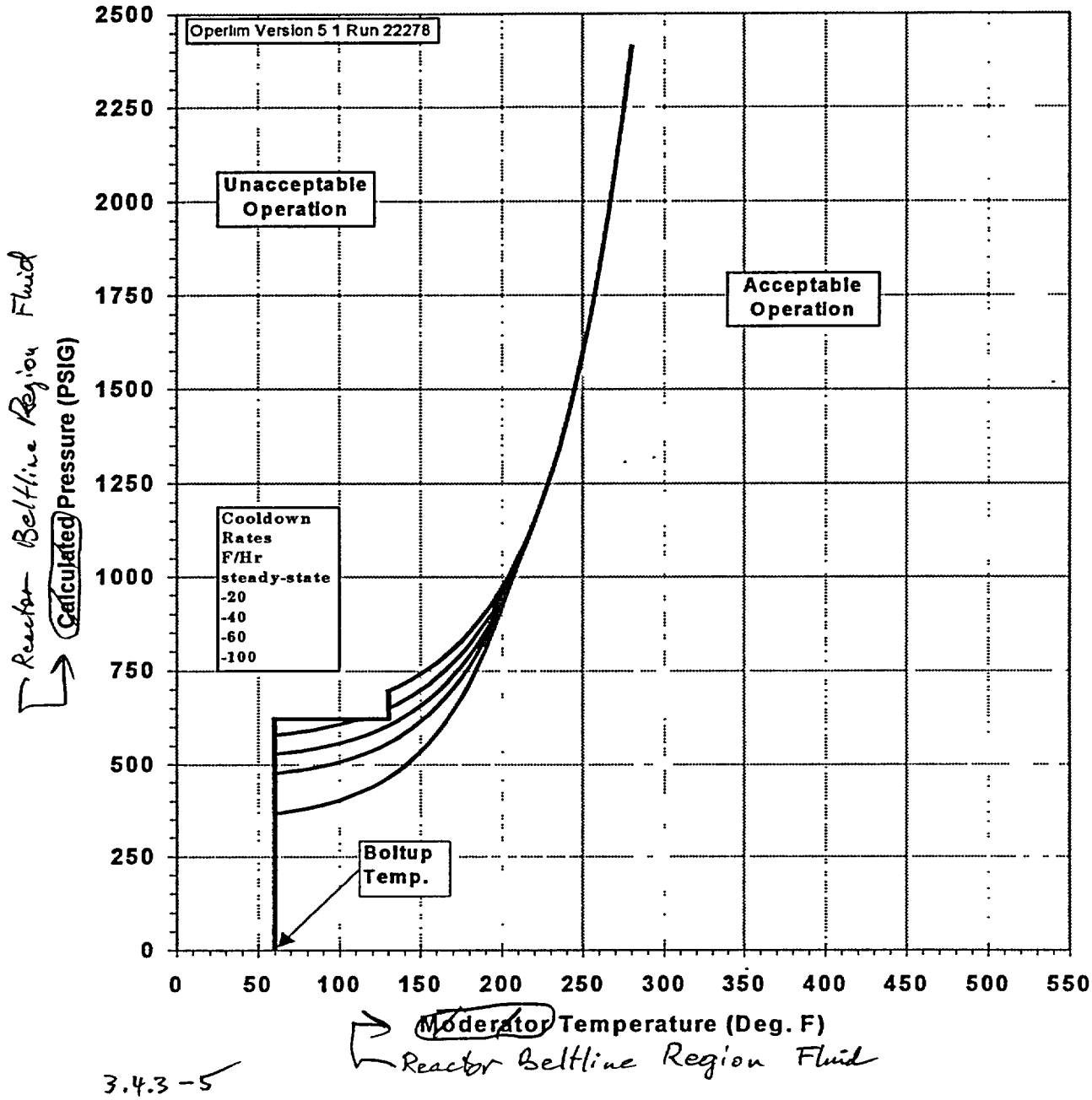


Figure 3.4.3-2 RCS Cooldown Limitations
(UNIT 1 ONLY)
Cooldown Rates up to 100 deg. F/HR
(Without margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the first 16 EFPY

Replaced by next figure

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T,202°F
3/4T,146°F



3.4.3-5

Figure 3 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIALS; LOWER SHELLFORGING 04

LIMITING ART AT 16 EFPY:
¼-t, 104 deg. F
¾-t, 73 deg. F

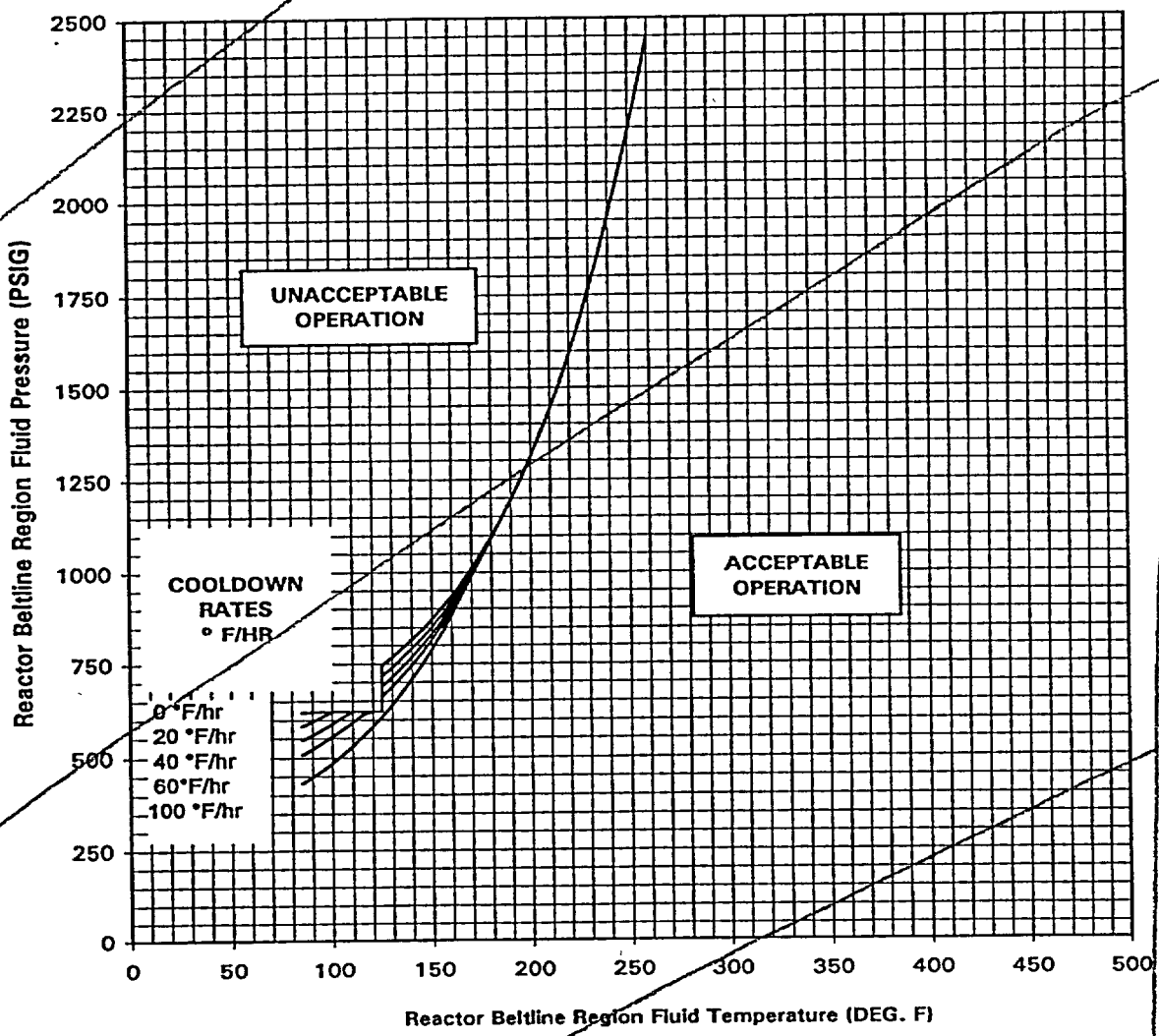
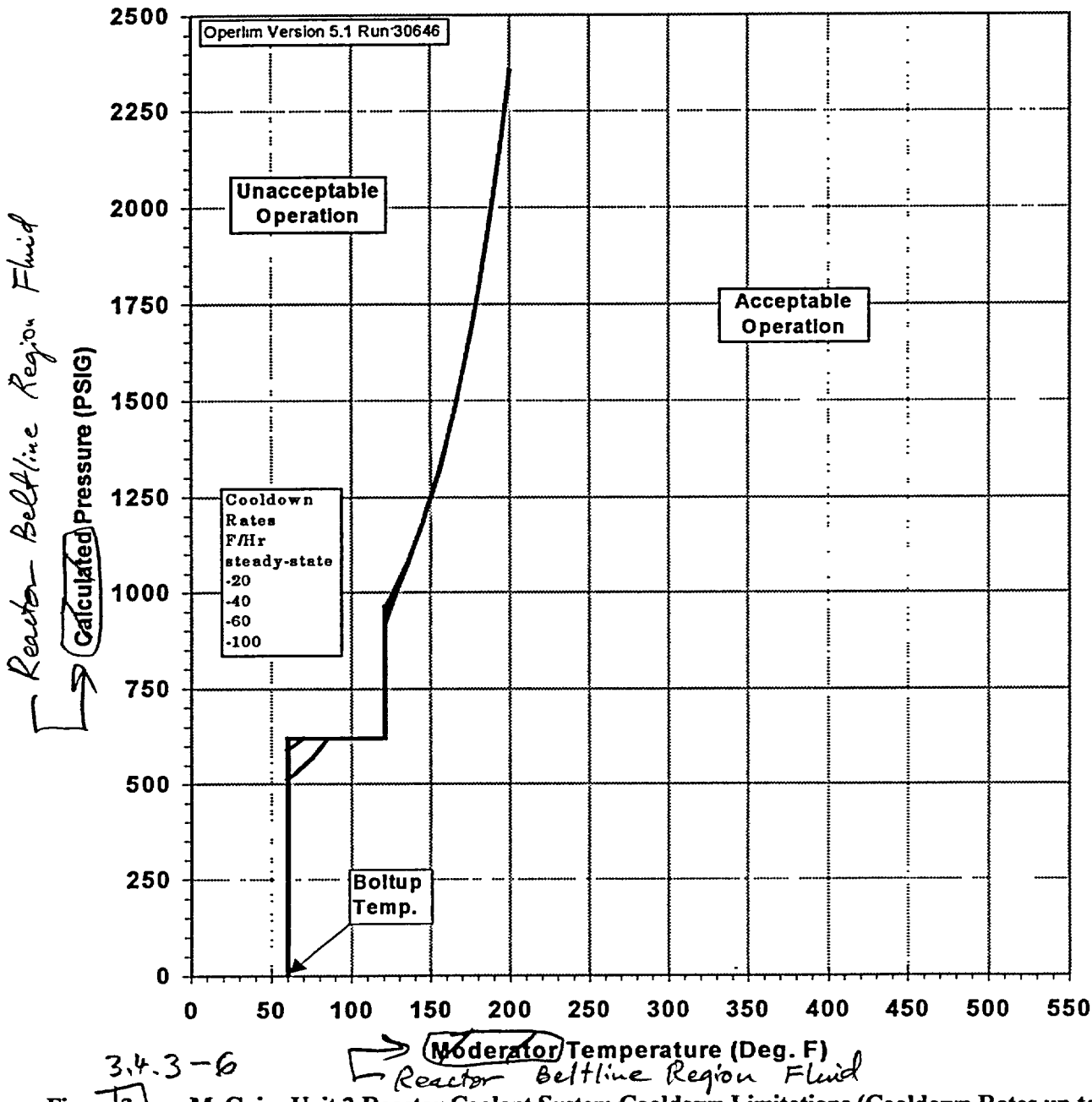


Figure 3.4.3-2 RCS Cooldown Limitations
(UNIT 2 ONLY)
Cooldown Rates up to 100 deg F/HR
(without Margins for Instrumentation Errors)
NRC REG GUIDE 1.99, Rev. 2
Applicable for the First 16 EFPY

Replaced by next figure

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F



3.4.3-6

Figure 3 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.12 An LTOP System shall be OPERABLE with a maximum of one centrifugal charging pump or one safety injection pump capable of injecting into the RCS and the accumulators isolated and either a or b below:

a. Two power operated relief valves (PORVs) with lift setting ≤ 385 psig or

b. The RCS depressurized and an RCS vent of ≥ 2.75 square inches.

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

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ 300°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

----------NOTE----------
Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in Specification 3.4.3.

ACTIONS

----------NOTE----------
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two centrifugal charging pumps capable of injecting into the RCS. OR One centrifugal charging pump and one safety injection pump capable of injecting into the RCS. OR Two safety injection pumps capable of injecting into the RCS.	A.1 ----------NOTE---------- Two centrifugal charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. ---------- Initiate action to verify a maximum of one centrifugal charging pump or one safety injection pump is capable of injecting into the RCS.	Immediately
	OR	
	A.2.1 Verify RHR suction relief valve is OPERABLE and the suction isolation valves are open.	Immediately
	AND	
	A.2.2.1 Verify RCS cold leg temperature ~~> 167°F~~.	Immediately > 174 °F (Unit 1) or > 89 °F (Unit 2).
	OR	
	A.2.2.2 Verify RCS cold leg temperature ~~> 107°F~~ and ~~cooldown rate < 20°F/hr~~.	Immediately > 74 °F and cooldown rate < 20 °F/hr (Unit 1), or > 74 °F and cooldown rate < 60 °F/hr (Unit 2).
	OR	

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Verify two PORVs secured open and associated block valves open and power removed.	Immediately
	OR	
	A.4 Depressurize RCS and establish RCS vent of ≥ 4.5 square inches.	Immediately
B. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in Specification 3.4.3.	B.1 Isolate affected accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > 300°F.	12 hours
	OR	
	C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed by Specification 3.4.3.	12 hours
D. One PORV inoperable in MODE 4.	D.1 Restore PORV to OPERABLE status.	7 days

(continued)

	OR	
	A.5.1 Depressurize RCS and establish RCS vent of ≥ 2.75 square inches.	Immediately
	AND	
	A.5.2 Verify two PORVs are OPERABLE	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One PORV inoperable in MODE 5 or 6.	E.1 Suspend all operations which could lead to a water solid pressurizer. AND E.2 Restore PORV to OPERABLE status.	Immediately 24 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1. Verify RCS cold leg temperature ~~≥ 167°F.~~ AND F.2 Verify RHR suction relief valve is OPERABLE and the suction isolation valves are open.	1 hour > 174 °F (Unit 1) or > 89 °F (Unit 2). 1 hour
G. Two PORVs inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ 2.75 square inches.	8 hours

ATTACHMENT 2

CHANGES TO TS BASES PAGES (mark-up)

For Information Only

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Manual and automatic initiation of steam line isolation must be OPERABLE in MODES 1, 2, and 3 when there is sufficient energy in the RCS and SGs to have an SLB or other accident. This could result in the release of significant quantities of energy and cause a cooldown of the primary system. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is insufficient energy in the RCS and SGs to experience an SLB or other accident releasing significant quantities of energy.

c. Steam Line Isolation-Containment Pressure-High High

This Function actuates closure of the MSIVs in the event of a LOCA or an SLB inside containment to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. The Containment Pressure - High High function is described in ESFAS Function 2.C.

Containment Pressure-High High must be OPERABLE in MODES 1, 2, and 3, when there is sufficient energy in the primary and secondary side to pressurize the containment following a pipe break. This would cause a significant increase in the containment pressure, thus allowing detection and closure of the MSIVs. The Steam Line Isolation Function remains OPERABLE in MODES 2 and 3 unless all MSIVs are closed and de-activated. In MODES 4, 5, and 6, there is not enough energy in the primary and secondary sides to pressurize the containment to the Containment Pressure-High High setpoint.

d. Steam Line Isolation-Steam Line Pressure

(1) Steam Line Pressure-Low

Steam Line Pressure-Low provides closure of the MSIVs in the event of an SLB to maintain three unfaulted SGs as a heat sink for the reactor, and to limit the mass and energy release to containment. This Function provides closure of the MSIVs in the event of a feed line break to ensure a supply of steam for the turbine driven AFW pump.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure-Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure-High High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure-Negative Rate-High signal for Steam Line Isolation below P-11 when SI has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

Steam Line Pressure - Low

(2) Steam Line Pressure-Negative Rate-High

Steam Line Pressure-Negative Rate-High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure-Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate-High signal is automatically enabled. Steam Line Pressure-Negative Rate-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in

For Information Only

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

This Specification contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and

BASES

BACKGROUND (continued)

Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 5).

Insert 1

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 7).

Insert 1:

A second program that employs excore cavity dosimetry to monitor the reactor vessel neutron fluence has been installed in both units. This program meets the requirements of 10 CFR 50 Appendix H (Ref. 4).

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the specified limit. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one centrifugal charging pump or one safety injection pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.75 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one centrifugal charging pump or one safety injection pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (Ref. 7).

BASES

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO permits a maximum of one centrifugal charging pump or one safety injection pump capable of injecting into the RCS and requires all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

   A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit.

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 2.75 square inches.

Insert 2

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 300°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 300°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 300°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

Insert 2:

The LCO is modified with a note that specifies that a PORV secured in the open position may be used to meet the RCS vent requirement provided that its associated block valve is open and power removed. With the PORV physically secured or locked in the open position with its associated block valve open and power removed, this vent path is passive and is not subject to active failure.

BASES

APPLICABILITY (continued)

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

LCO 3.0.4 is not applicable for entry into LTOP operation.

A.1, A.2.1, A.2.2.1, A.2.2.2., A.3, ~~and~~ A.4, A.5.1, and A.5.2

With two centrifugal charging pumps, safety injection pumps, or a combination of each, capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition. Two pumps may be capable of injecting into the RCS provided the RHR suction relief valve is OPERABLE with the RCS cold leg temperature > 167°F or > 107°F and cooldown rate limited to 20°F, or if two PORVs are secured open with the associated block valves open and power removed, or with an RCS vent of 4.5 square inches. For cases where no reactor coolant pumps are in operation, RCS cold leg temperature limits are to be met by monitoring of BOTH the WR Cold Leg temperatures and Residual Heat Removal Heat Exchanger discharge temperature. With both PORVS and block valves secured open, or with an RCS vent of 4.5 square inches, there are no credible single failures to limit the flow relief capacity. For the RHR relief valve to be OPERABLE, the RHR suction isolation valves must be open and the relief valve setpoint at 450 psig consistent with the safety analysis. The RHR suction relief valves are spring loaded, bellows type water relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Code (Ref. 3) for Class 2 relief valves.

Insert 3

Required Action A.1 is modified by a Note that permits two centrifugal charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.

B.1, C.1, and C.2

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum

Insert 3:

1. RCS cold leg temperature > 174 °F (Unit 1), or
2. RCS cold leg temperature > 89 °F (Unit 2), or
3. RCS cold leg temperature > 74 °F and cooldown rate < 20 °F /hr (Unit 1), or
4. RCS cold leg temperature > 74 °F and cooldown rate < 60 °F /hr (Unit 2), or
5. two PORVs secured open with associated block valves open and power removed, or
6. a RCS vent of ≥ 4.5 square inches, or
7. a RCS vent of ≥ 2.75 square inches and two OPERABLE PORVs (the RCS vent shall not be one of the two OPERABLE PORVs).

BASES

ACTIONS (continued)

>174 °F (Unit 1) or
> 89 °F (Unit 2).

F.1 and F.2

If the Required Actions and associated Completion Times of Condition E are not met, then alternative actions are necessary to establish the required redundancy in relief capacity. This is accomplished by verifying that the RHR relief valve is OPERABLE and the RHR suction isolation valves open and the RCS cold leg temperature ~~> 167°F~~. For cases where no reactor coolant pumps are in operation, RCS cold leg temperature limits are to be met by monitoring of BOTH the WR Cold Leg temperatures and Residual Heat Removal Heat Exchanger discharge temperature. The Completion Time of 1 hour reflects the importance of restoring the required redundancy at lower RCS temperatures.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition C, D, E, or F is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D, E, or F.

The vent must be sized ≥ 2.75 square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, all but one centrifugal charging pump or

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION PAGES (retyped)

Table 3 3.2-1 (page 3 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
(2) Negative Rate - High	3[b][c]	3 per steam line	D	SR 3 3.2.1 SR 3 3 2 5 SR 3 3 2 8 SR 3 3 2 9	≤ 120[d] psi	100[d] psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2[e]	2 trains	I	SR 3.3 2 2 SR 3.3 2 4 SR 3.3 2.6	NA	NA
b. SG Water Level - High High (P-14)	1,2[e]	3 per SG	J	SR 3.3 2.1 SR 3 3 2.2 SR 3 3 2.4 SR 3 3 2.5 SR 3 3 2.6 SR 3 3 2.8 SR 3 3 2.9	≤ 85 6%	83.9%
c Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements					
d T_{avg}-Low	1,2[e]	1 per loop	J	SR 3 3.2.1 SR 3 3.2.5 SR 3 3 2 8	≥ 551°F	553°F
coincident with Reactor Trip, P-4	Refer to Function 8.a (Reactor Trip, P-4) for all initiation functions and requirements.					
e Doghouse Water Level-High High	1,2[e]	2 per train per Doghouse	L,M	SR 3.3 2 1 SR 3.3 2 7	≤13 inches	12 inches
6 Auxiliary Feedwater						
a Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	H	SR 3 3 2.2 SR 3 3 2.4 SR 3 3 2.6	NA	NA
b SG Water Level - Low Low	1,2,3	4 per SG	D	SR 3 3.2.1 SR 3 3.2.5 SR 3 3.2.8 SR 3 3.2 9	≥ 15%	16.7%

(continued)

(b) Except when all MSIVs are closed and de-activated
(c) Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Steam Line Isolation on Steam Line Pressure-Low is not blocked
(d) Time constant utilized in the rate/lag controller is ≥ 50 seconds
(e) Except when all MFIVs, MFCVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure and RCS temperature shall be limited during RCS heatup and cooldown, criticality, and inservice leak and hydrostatic testing in accordance with:

a. A maximum heatup rate as specified in Figure 3.4.3-1, Figure 3.4.3-2, Figure 3.4.3-3, or Figure 3.4.3-4;

b. A maximum cooldown rate as specified in Figure 3.4.3-5 or Figure 3.4.3-6; and

c. A maximum temperature change of ≤ 10°F in any 1 hour period during inservice leak and hydrostatic testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ----------NOTE---------- Required Action A.2 shall be completed whenever this Condition is entered. ---------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits.	30 minutes
	AND	
	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5 with RCS pressure < 500 psig.	36 hours

(continued)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

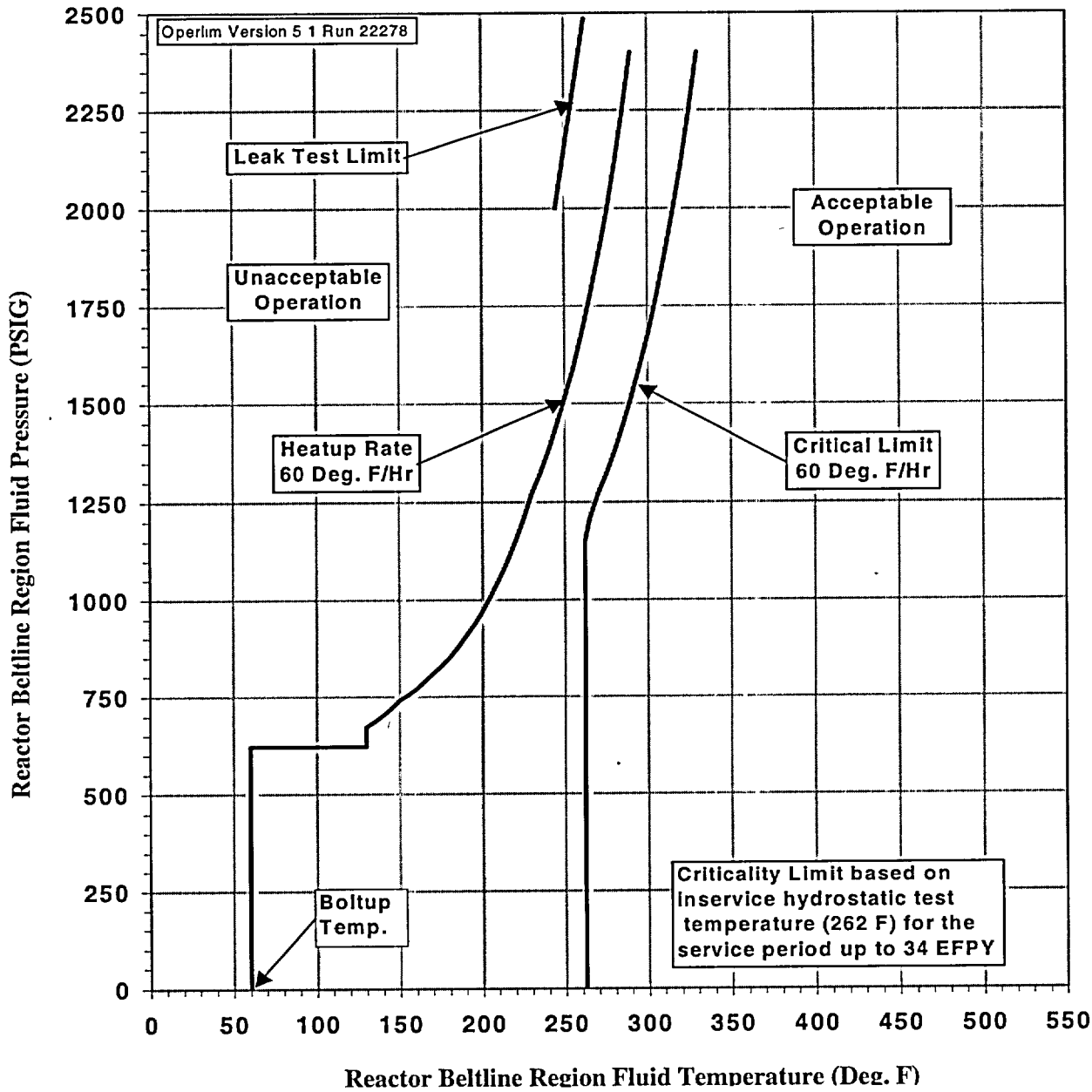


Figure 3.4.3-1 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

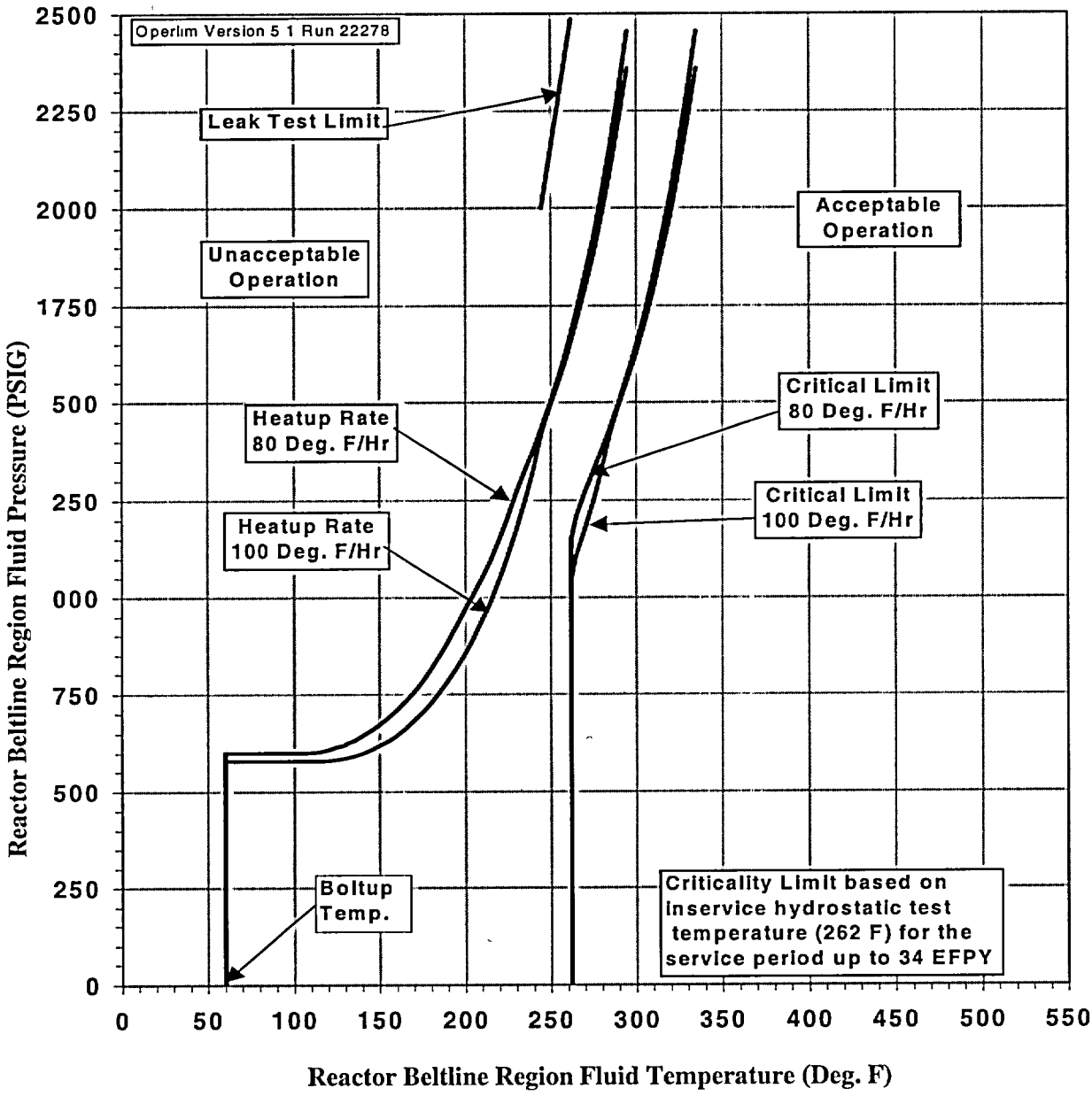


Figure 3.4.3-2 McGuire Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

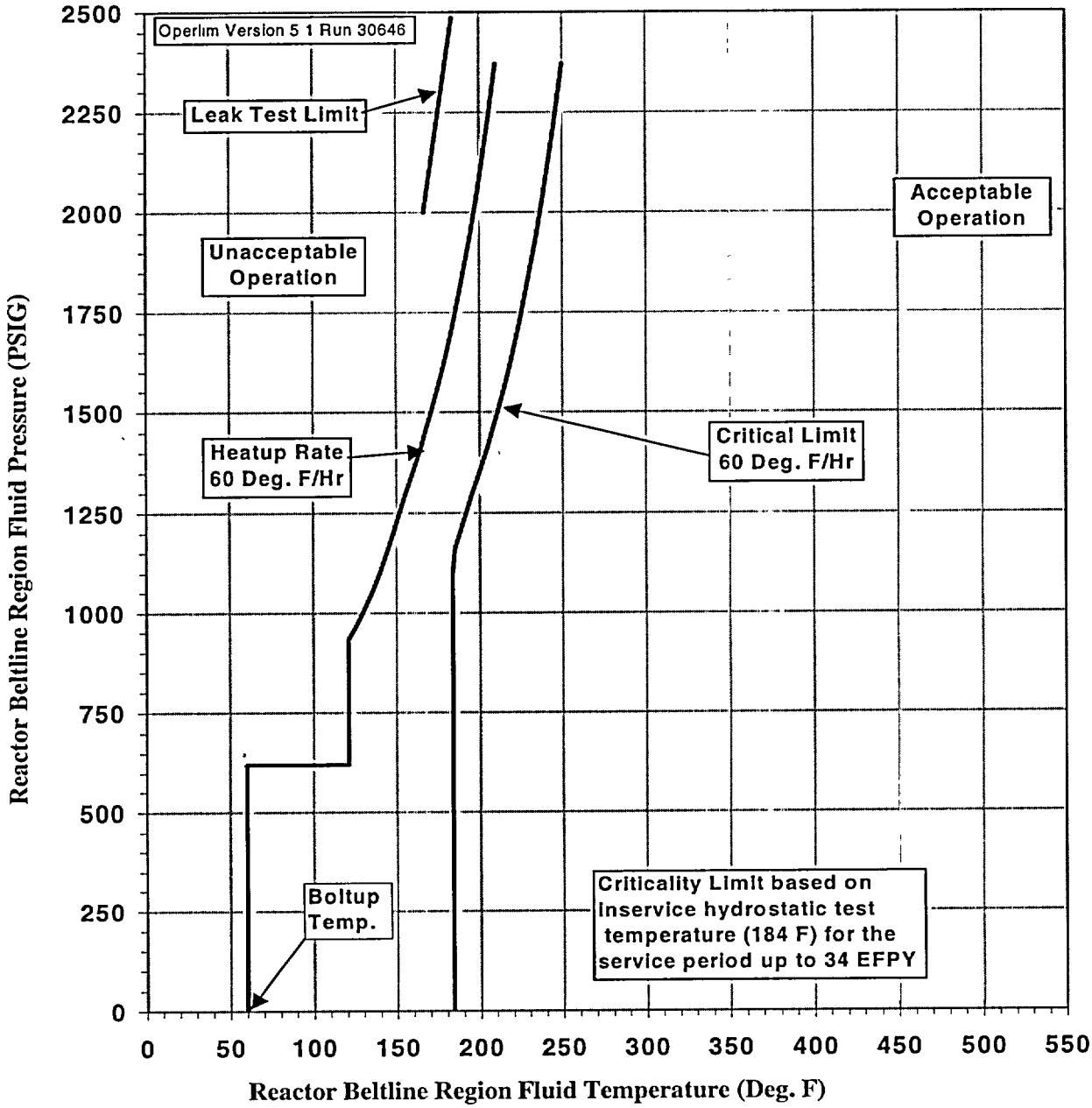


Figure 3.4.3-3 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 60°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

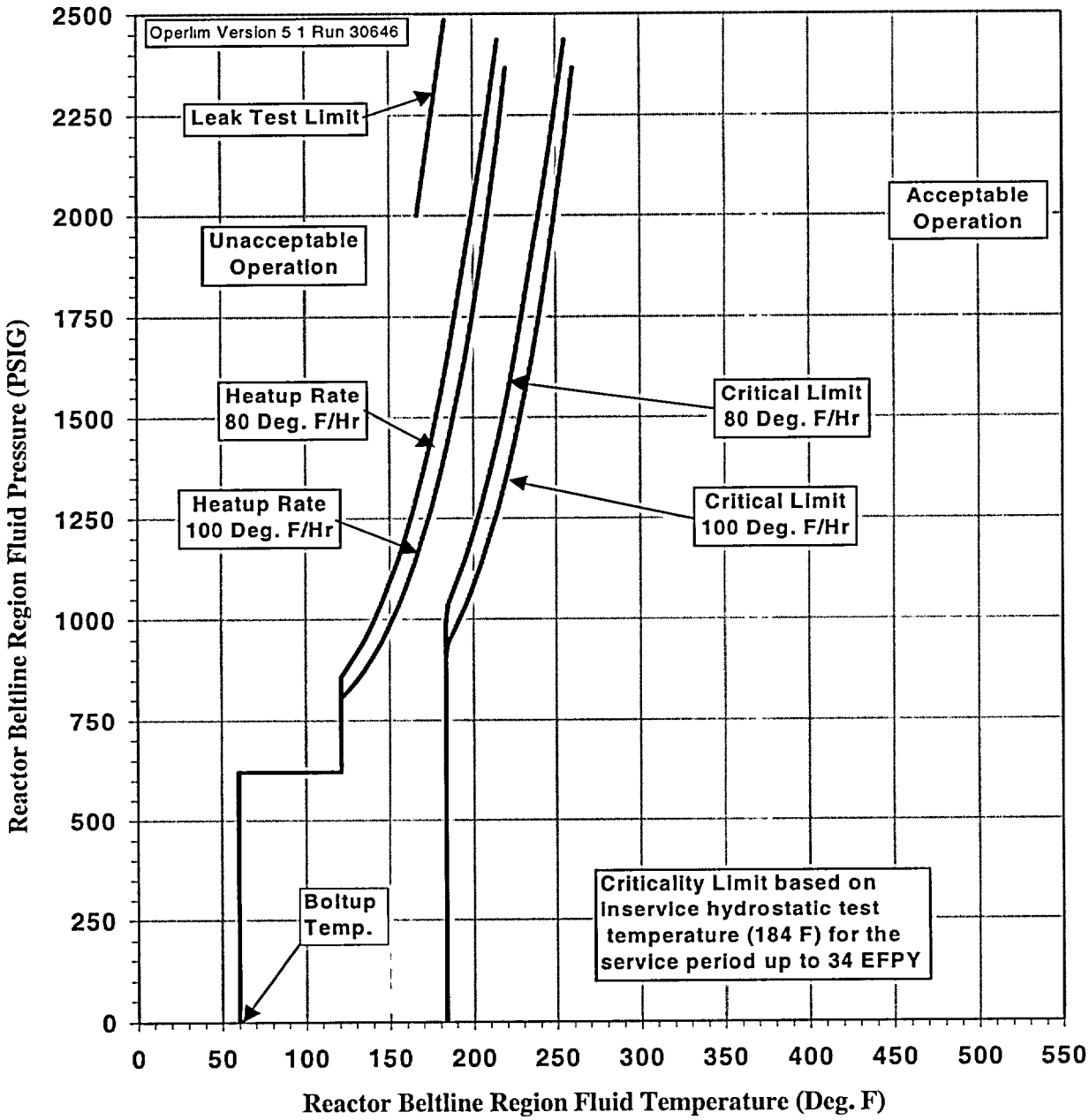


Figure 3.4.3-4 McGuire Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rates of 80 & 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL LONGITUDINAL WELD
LIMITING ART VALUES AT 34 EFPY: 1/4T, 202°F
3/4T, 146°F

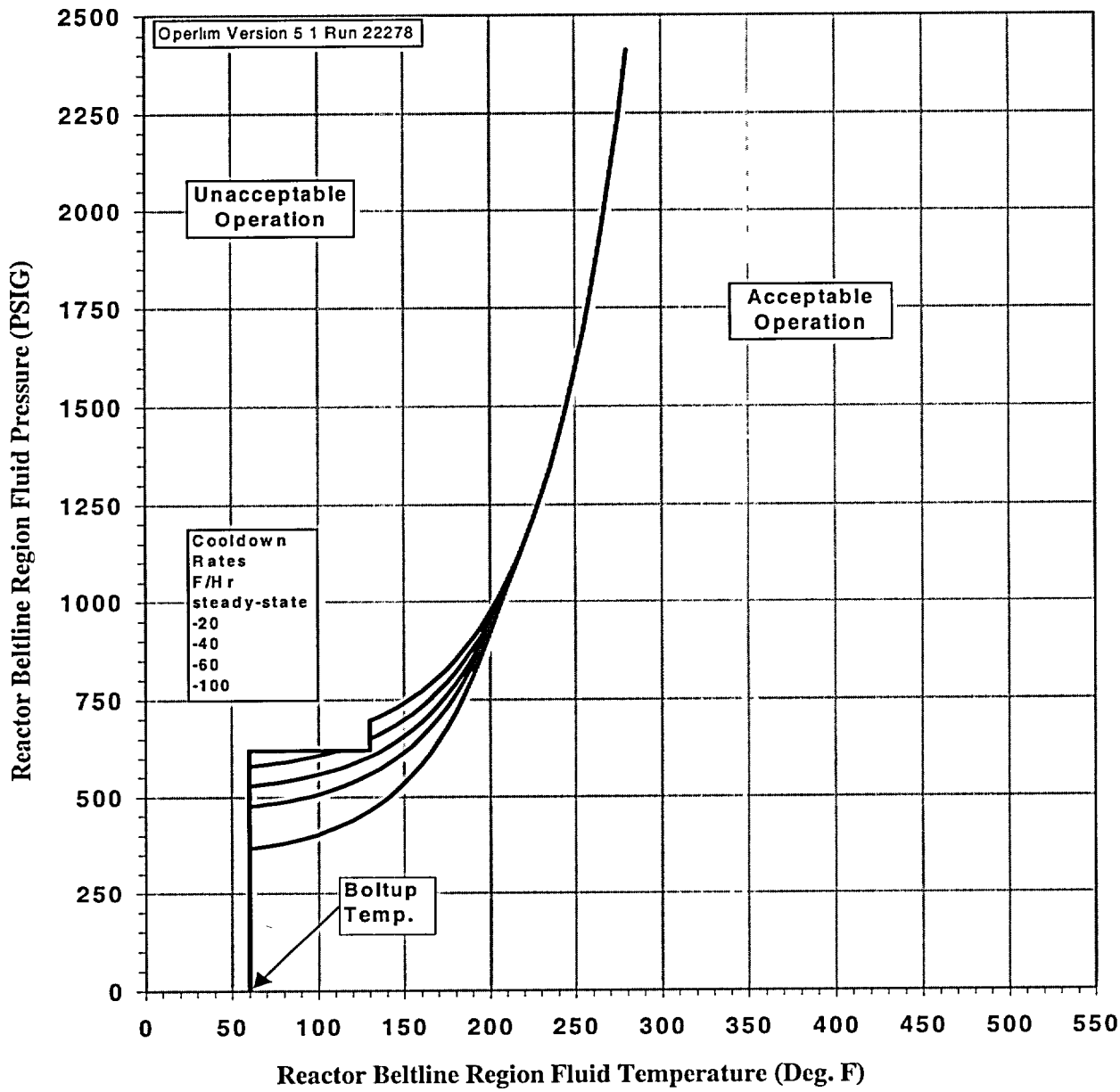


Figure 3.4.3-5 McGuire Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL FORGING 04
LIMITING ART VALUES AT 34 EFPY: 1/4T, 123°F
3/4T, 91°F

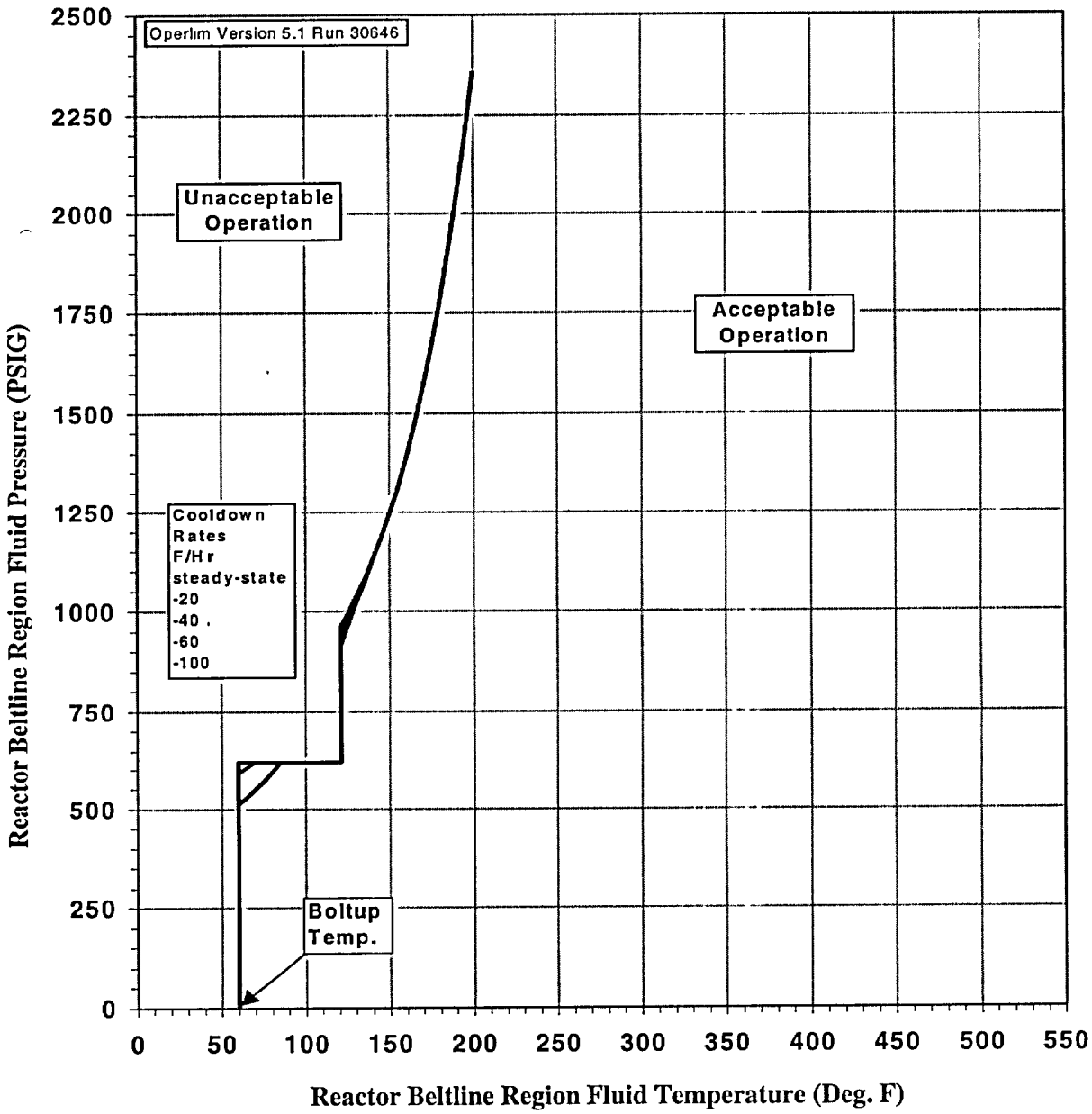


Figure 3.4.3-6 McGuire Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates up to 100°F/hr) Applicable for the First 34 EFPY (Without Margins for Instrumentation Errors) Using 1996 App.G Methodology & ASME Code Case N-641

ACTIONS

----------NOTE----------
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two centrifugal charging pumps capable of injecting into the RCS. OR One centrifugal charging pump and one safety injection pump capable of injecting into the RCS. OR Two safety injection pumps capable of injecting into the RCS.	A.1 ----------NOTE---------- Two centrifugal charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. ---------- Initiate action to verify a maximum of one centrifugal charging pump or one safety injection pump is capable of injecting into the RCS.	Immediately
	OR	
	A.2.1 Verify RHR suction relief valve is OPERABLE and the suction isolation valves are open.	Immediately
	AND	
	A.2.2.1 Verify RCS cold leg temperature > 174°F (Unit 1) or > 89°F (Unit 2).	Immediately
	OR	
	A.2.2.2 Verify RCS cold leg temperature > 74°F and cooldown rate < 20°F/ hr (Unit 1), or > 74°F and cooldown rate < 60°F/hr (Unit 2).	Immediately
	OR	(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.3 Verify two PORVs secured open and associated block valves open and power removed.	Immediately
	OR	
	A.4 Depressurize RCS and establish RCS vent of ≥ 4.5 square inches.	Immediately
	OR	
	A.5.1 Depressurize RCS and establish RCS vent of ≥ 2.75 square inches.	Immediately
	AND	
	A.5.2 Verify two PORVs are OPERABLE.	Immediately
B. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in Specification 3.4.3.	B.1 Isolate affected accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > 300°F.	12 hours
	OR	
	C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed by Specification 3.4.3.	12 hours
D. One PORV inoperable in MODE 4.	D.1 Restore PORV to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. One PORV inoperable in MODE 5 or 6.	E.1 Suspend all operations which could lead to a water solid pressurizer. AND E.2 Restore PORV to OPERABLE status.	Immediately 24 hours
F. Required Action and associated Completion Time of Condition E not met.	F.1. Verify RCS cold leg temperature > 174°F (Unit 1) or > 89°F (Unit 2). AND F.2 Verify RHR suction relief valve is OPERABLE and the suction isolation valves are open.	1 hour 1 hour
G. Two PORVs inoperable. OR Required Action and associated Completion Time of Condition C, D, E, or F not met. OR LTOP System inoperable for any reason other than Condition A, B, C, D, E, or F.	G.1 Depressurize RCS and establish RCS vent of ≥ 2.75 square inches.	8 hours

ATTACHMENT 4

TS BASES PAGES (retyped)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Steam Line Pressure-Low Function must be OPERABLE in MODES 1, 2, and 3 (above P-11), with any main steam valve open, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam lines. This signal may be manually blocked by the operator below the P-11 setpoint. Below P-11, an inside containment SLB will be terminated by automatic actuation via Containment Pressure-High High. Stuck valve transients and outside containment SLBs will be terminated by the Steam Line Pressure-Negative Rate-High signal for Steam Line Isolation below P-11 when Steam Line Pressure – Low has been manually blocked. The Steam Line Isolation Function is required in MODES 2 and 3 unless all MSIVs are closed and de-activated. This Function is not required to be OPERABLE in MODES 4, 5, and 6 because there is insufficient energy in the secondary side of the unit to have an accident.

(2) Steam Line Pressure-Negative Rate-High

Steam Line Pressure-Negative Rate-High provides closure of the MSIVs for an SLB when less than the P-11 setpoint, to maintain at least one unfaulted SG as a heat sink for the reactor, and to limit the mass and energy release to containment. When the operator manually blocks the Steam Line Pressure-Low main steam isolation signal when less than the P-11 setpoint, the Steam Line Pressure-Negative Rate-High signal is automatically enabled. Steam Line Pressure-Negative Rate-High provides no input to any control functions. Thus, three OPERABLE channels are sufficient to satisfy requirements with a two-out-of-three logic on each steam line.

Steam Line Pressure-Negative Rate-High must be OPERABLE in MODE 3 when less than the P-11 setpoint, when a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). In MODES 1 and 2, and in MODE 3, when above the P-11 setpoint, this signal is automatically disabled and the Steam Line Pressure-Low signal is automatically enabled. The Steam Line Isolation Function is required to be OPERABLE in

BASES

BACKGROUND (continued)

Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 5).

A second program that employs excore cavity dosimetry to monitor the reactor vessel neutron fluence has been installed in both units. This program meets the requirements of 10 CFR 50 Appendix H (Ref. 4).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 7).

BASES

APPLICABLE SAFETY ANALYSES (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the specified limit. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one centrifugal charging pump or one safety injection pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

The PORV setpoints will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent size of 2.75 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one centrifugal charging pump or one safety injection pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of 10 CFR 50.36 (Ref. 7).

BASES

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

To limit the coolant input capability, the LCO permits a maximum of one centrifugal charging pump or one safety injection pump capable of injecting into the RCS and requires all accumulator discharge isolation valves closed and immobilized when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in LCO 3.4.3.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs (NC-32B and NC-34A); or

   A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the specified limit and testing proves its automatic ability to open at this setpoint, and motive power is available to the valve and its control circuit.

b. A depressurized RCS and an RCS vent.

   An RCS vent is OPERABLE when open with an area of ≥ 2.75 square inches.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

The LCO is modified with a note that specifies that a PORV secured in the open position may be used to meet the RCS vent requirement provided that its associated block valve is open and power removed. With the PORV physically secured or locked in the open position with its associated block valve open and power removed, this vent path is passive and is not subject to active failure.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ 300°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above 300°F. When the reactor vessel head is off, overpressurization cannot occur.

LCO 3.4.3 provides the operational P/T limits for all MODES.
LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of

BASES

APPLICABILITY (continued)

the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 300°F.

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.

ACTIONS

LCO 3.0.4 is not applicable for entry into LTOP operation.

A.1, A.2.1, A.2.2.1, A.2.2.2, A.3, A.4, A.5.1, and A.5.2

With two centrifugal charging pumps, safety injection pumps, or a combination of each, capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition. Two pumps may be capable of injecting into the RCS provided the RHR suction relief valve is OPERABLE with:

1. RCS cold leg temperature > 174°F (Unit 1), or
2. RCS cold leg temperature > 89°F (Unit 2), or
3. RCS cold leg temperature > 74°F and cooldown rate < 20°F/hr (Unit 1), or
4. RCS cold leg temperature > 74°F and cooldown rate < 60°F/hr (Unit 2), or
5. two PORVs secured open with associated block valves open and power removed, or
6. a RCS vent of ≥ 4.5 square inches, or
7. a RCS vent of ≥ 2.75 square inches and two OPERABLE PORVs (the RCS vent shall not be one of the two OPERABLE PORVs).

For cases where no reactor coolant pumps are in operation, RCS cold leg temperature limits are to be met by monitoring of BOTH the WR Cold Leg temperatures and Residual Heat Removal Heat Exchanger discharge temperature. With both PORVS and block valves secured open, or with

BASES

ACTIONS (continued)

active failure of the remaining valve path during this time period is very low.

E.1 and E.2

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 8). Thus, with one of the two PORVs inoperable in MODE 5 or in MODE 6 with the head on, all operations which could lead to a water solid pressurizer must be suspended immediately and the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE PORV to protect against overpressure events.

F.1 and F.2

If the Required Actions and associated Completion Times of Condition E are not met, then alternative actions are necessary to establish the required redundancy in relief capacity. This is accomplished by verifying that the RHR relief valve is OPERABLE and the RHR suction isolation valves open and the RCS cold leg temperature > 174°F (Unit 1) or > 89°F (Unit 2). For cases where no reactor coolant pumps are in operation, RCS cold leg temperature limits are to be met by monitoring of BOTH the WR Cold Leg temperatures and Residual Heat Removal Heat Exchanger discharge temperature. The Completion Time of 1 hour reflects the importance of restoring the required redundancy at lower RCS temperatures.

G.1

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition C, D, E, or F is not met; or

ATTACHMENT 5

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Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure – High	1,2, 3	[4]	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [6.61] psig	≤ [6.35] psig
d. Steam Line Pressure						
(1) Low	1,2, 3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [635] psig	≥ [675] psig
(2) Negative Rate – High	3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [121.6] psi/sec	≤ [130] psi/sec
e. High Steam Flow in Two Steam Lines	1,2(i), 3(i)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	(e)	(f)
Coincident with T_{avg} – Low Low	1,2(i), 3(d)(i)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [550.6]°F	≥ [553]°F

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(b) Above the P-11 (Pressurizer Pressure) interlock.
(c) Time constants used in the lead/lag controller are $t_1 \geq [50]$ seconds and $t_2 \leq [5]$ seconds.
(d) Above the P-12 (T_{avg} – Low Low) interlock.
(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.
(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.
(g) Below the P-11 (Pressurizer Pressure) interlock.
(h) Time constant utilized in the rate/lag controller is ≤ [50] seconds.
(i) Except when all MSIVs are closed and [de-activated].

≥ 50

Trip function automatically blocked above P-11 (Pressurizer Pressure) interlock and may be blocked below P-11 when Safety Injection Steam Line Pressure-Low is not blocked.

Should be "Steam Line Isolation" based on Amendments 182/164.