R.E. Ginna Nuclear Power Plant

RCS Pressure and Temperature Limits Report

Cycle 25

Draft B

Note: This report is not part of the Technical Specifications. This report is referenced in the Technical Specifications.

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1.0 RCS Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Ginna Station has been prepared in accordance with the requirements of Technical Specification 5.6.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications addressed in this report are listed below:

3.4.3	RCS Pressure and Temperature (P/T) Limits
3.4.6	RCS Loops - MODE 4
3.4.7	RCS Loops - MODE 5, Loops Filled
3.4.10	Pressurizer Safety Valves
3.4.12	Low Temperature Overpressure Protection (LTOP) System

2.0 OPERATING LIMITS

The cycle-specific parameter limits for the specifications listed in Section 1.0 are presented in the following subsections. All changes to these limits must be developed using the NRC approved methodologies specified in Technical Specification 5.6.6. These limits have been determined such that all applicable limits of the safety analysis are met. All items that appear in capitalized type are defined in Technical Specification 1.1, "Definitions."

2.1 RCS Pressure and Temperature Limits (LCO 3.4.3 and LCO 3.4.12) (Reference 1)

2.1.1 The RCS temperature rate-of-change limits are:

a. A maximum heatup of 60°F per hour.

b. A maximum cooldown of 100°F per hour.

2.1.2 The RCS P/T limits for heatup and cooldown are specified by Figures 1 and 2, respectively.

2.1.3 The minimum boltup temperature, using the methodology of Reference 2, is 60°F.

2.2 Low Temperature Overpressure Protection System Enable Temperature (LCOs 3.4.6, 3.4.7, 3.4.10 and 3.4.12) (Reference 1)

2.2.1 The enable temperature for the Low Temperature Overpressure Protection System is ≥ 330°F.

2.3 Low Temperature Overpressure Protection System Setpoints (LCO 3.4.12)

2.3.1 Pressurizer Power Operated Relief Valve Lift Setting Limits (Reference 1)

The lift setting for the pressurizer Power Operated Relief Valves (PORVs) is ≤ 424 psig.

3.0 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

The reactor vessel material irradiation surveillance specimens shall be removed and examined to determine changes in material properties. The removal schedule is provided in Table 1. The results of these examinations shall be used to update Figures 1 and 2.

The pressure vessel steel surveillance program (Ref. 3) is in compliance with Appendix H to 10 CFR 50, entitled, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standard utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASTM E208. The empirical relationship between RT_{NDT} and the fracture toughness of the reactor vessel steel is developed in accordance with Appendix G, "Protection Against Non-Ductile Failure," to section III of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E185-82.

As shown by Reference 4, the reactor vessel material irradiation surveillance specimens indicate that the surveillance data meets the credibility discussion presented in Regulatory Guide 1.99 revision 2 where:

1. The capsule materials represent the limiting reactor vessel material.

2. Charpy energy vs. temperature plots scatter are small enough to permit determination of 30 ft-lb temperature and upper shelf energy unambiguously.

3. The scatter of ΔRT_{NDT} values are within the best fit scatter limits as shown on Table 2. The only exception is with respect to the Intermediate Shell which is not the limiting reactor vessel material.

4. The Charpy specimen irradiation temperature matches the reactor vessel surface interface temperature within ± 25°F limits.

5. The surveillance data falls within the scatter band of the material database.

4.0 SUPPLEMENTAL DATA INFORMATION AND DATA TABLES

4.1 The RT_{PTS} value for Ginna Station is 275.2°F for 32 EFPY per Reference 5. (Note - these values are based on Capsule T. The new revised RT_{PTS} values based on Capsule S will be implemented following NRC review of these values).

4.2 Tables

Table 2 contains a comparison of measured surveillance material 30 ft-lb transition temperature shifts and upper shelf energy decreases with Regulatory Guide 1.99, Revision 2 predictions.

Table 3 shows calculations of the surveillance material chemistry factors using surveillance capsule data.

Table 4 provides the reactor vessel toughness data.

Table 5 provides a summary of the fluence values used in the generation of the heatup and cooldown limit curves.

Table 6 shows example calculations of the ART values at 21 EFPY for the limiting reactor vessel material.

5.0 REFERENCES

1. Letter from A.R. Johnson, NRC, to R.C. Mecredy, RG&E, Subject: "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992.

2. WCAP-14040, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 1, December 1994 as approved by letter from C.I. Grimes, NRC to R.H. Newton, WOG, Subject: "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1", (TAC No. M91749), dated October 16, 1995.

3. WCAP-7254, "Rochester Gas and Electric, Robert E. Ginna Unit No. 1 Reactor Vessel Radiation Surveillance Program," May 1969.

4. WCAP-13902, "Analysis of Capsule S from the Rochester Gas and Electric Corporation R.E. Ginna Reactor Vessel Radiation Surveillance Program," dated December 1993.

5. Letter from George E. Lear, NRC to Roger Kober, RG&E, "Safety Evaluation by Office of Nuclear Reactor Regulation Regarding Projected Values of Material Properties for Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events, Rochester Gas and Electric Company, R.E. Ginna Nuclear Power Plant Docket No. 50-244 (TAC No. 59956)," dated November 17, 1986.

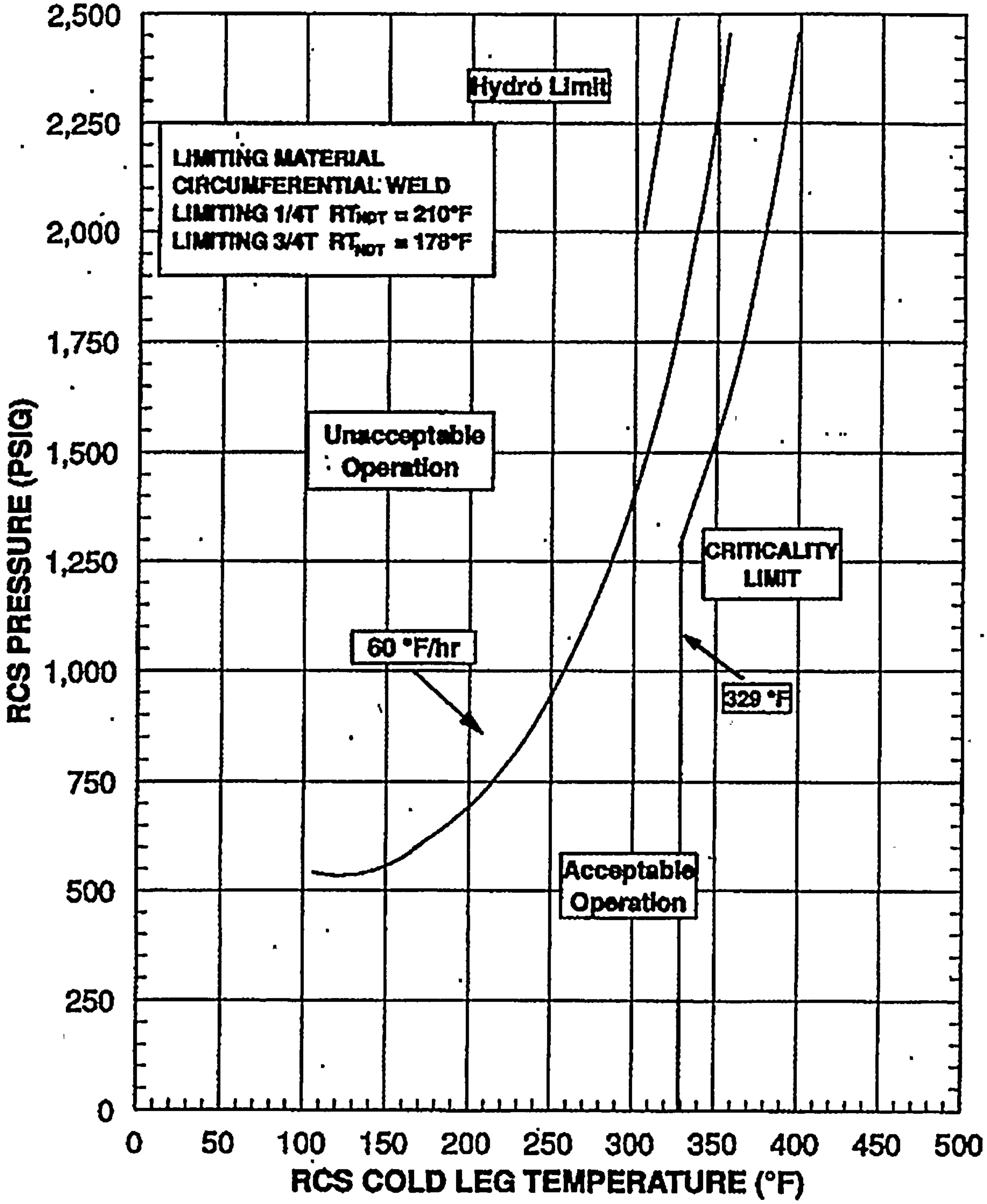


FIGURE 1
REACTOR VESSEL HEATUP LIMITATIONS
APPLICABLE FOR THE FIRST 21 EFPY

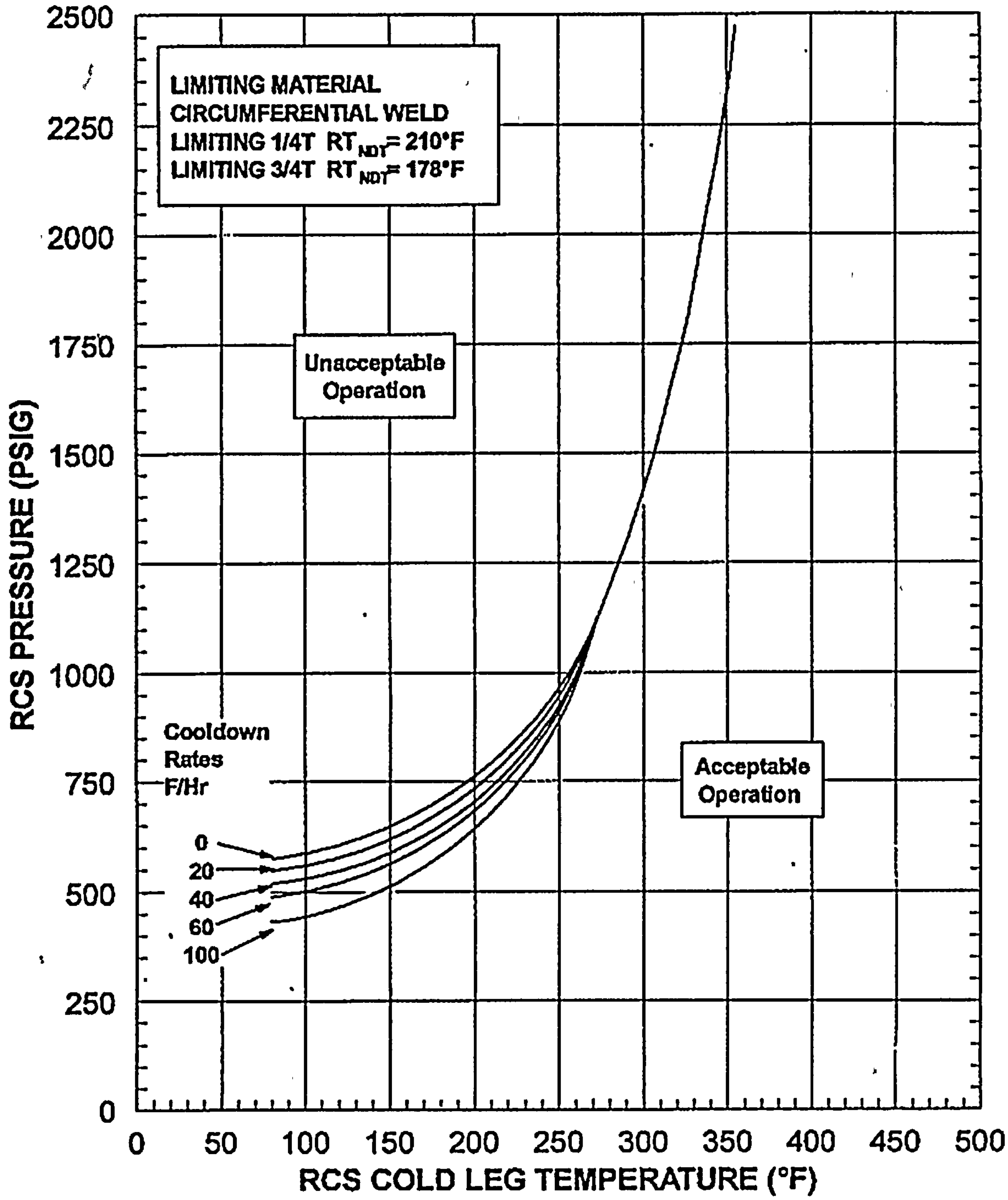


FIGURE 2

REACTOR VESSEL COOLDOWN LIMITATIONS
APPLICABLE FOR THE FIRST 21 EFPY

Table 1

Surveillance Capsule Removal Schedule

Capsule	Vessel Location (deg.)	Capsule Lead Factor	Removal Schedule[a]	Capsule Fluence E19(n/cm^2)
V	77°	2.99	1.6 (removed)	0.556
R	257°	3.00	2.7 (removed)	1.15
T	67°	1.85	7 (removed)	1.97
S	57°	1.74	17 (removed)	3.87
N	237°	1.74	TBD[b]	TBD[b]
P	247°	1.9	Standby	N/A

NOTES:

(a) Effective Full Power Years (EFPY).

(b) To be determined, there is no current requirement for removal.

(c) Currently under NRC review, not included in current heat up/cooldown curve.

TABLE 2

Surveillance Material 30 ft-lb Transition Temperature Shifts and Upper Shelf Energy Values

Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	30 lb-ft Transition Temperature Shift		
			Predicted[a] (°F)	Measured (°F)	Δ (°F)
Lower Shell	V	.556	26	25	1
	R	1.15	32	25	7
	T	1.97	37	30	7
Intermediate Shell	V	.556	37	0	37
	R	1.15	46	0	46
	T	1.97	52	0	52
Weld Metal	V	.556	135	140	-5
	R	1.15	168	165	3
	T	1.97	191	150	41
HAZ Metal	V	.556	---	0	---
	R	1.15	---	90	---
	T	1.97	---	100	---

(a) Based upon Reg. Guide 1.99, Revision 2 predictions

(b) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 3 Calculation of Chemistry Factors Using Surveillance Capsule Data						
Material	Capsule	Fluence (x 10^{19} n/cm^2, E > 1.0 MeV)	FF	ΔRT_{NDT} (°F)	FF*ΔRT_{NDT} (°F)	FF^2
Intermediate Shell Forging 05 (Tangential)	V	.703	0.901	25	22.5	.812
	R	1.01	1.003	25	25.1	1.006
	T	1.75	1.154	30	34.6	1.332
				Sum:	82.2	3.15
	Chemistry Factor[a] = 26.1					
Intermediate Shell	V	.703	0.901	0	0	.812
	R	1.01	1.003	0	0	1.006
	T	1.75	1.154	0	0	1.332
				Sum:	0.0	3.15
	Chemistry Factor[a] = 0.0					
Weld Metal	V	.703	.901	140	126.1	.812
	R	1.01	1.003	165	165.5	1.006
	T	1.75	1.154	150	173.1	1.332
				Sum:	464.7	3.15
	Chemistry Factor[a][b] = 147.5°F					

NOTES:

(a) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

(b) Using surveillance capsule values of Cu = .23 and Ni = .52.

TABLE 4 Reactor Vessel Toughness Table (Unirradiated)[a]			
Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}(°F)
Intermediate Shell	.07	.69	20
Lower Shell	.05	.69	40
Circumferential Weld	.25	.55	0

(a) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 5 Reactor Vessel Surface Fluence Values at 7 and 21 EFPY[a] x 10^{19} (n/cm^2, E > 1.0 MeV)				
EFPY	0°	14.5°	30°	44.5°
7	.866	.538	.359	.310
21	2.32[b]	1.42	0.991	.893

(a) WCAP-11026 "R.E. Ginna Reactor Vessel Fluence and RT_{PTS} Evaluations," dated December 1985 Table II.2-3 through II.2-6, as referenced from License Amendment No. 48.

(b) Letter from A. Johnson (NRC) to R. Mecredy (RG&E) "Issuance of Amendment No. 48 to Facility Operating License No. DPR-18, R.E. Ginna Nuclear Power Plant (TAC No. M79828)," dated March 6, 1992 especially Reference (5) of Section 3.1.2.

TABLE 6

Calculation of Adjusted Reference Temperatures at 21 EFPY for the Limiting Reactor Vessel Material

Parameter	Values	
Operating Time	21 EFPY	
Material	Circ. Weld	Circ. Weld
Location	1/4-T	3/4-T
Chemistry Factor (CF), °F[b]	147.5	147.5
Fluence (f), $\div 10^{19}$ n/cm^2 (E > 1.0 MeV)[a]	1.57	.720
Fluence Factor (FF)[a]	1.125	.908
ΔRT_{NDT} = CF x FF, °F	165.9	133.93
Initial RT_{NDT} (I), °F	0	0
Margin (M), °F[a]	44	44
ART = I + (CFxFF) + M, °F[a]	210	178°F

NOTES:

(a) Value calculated from Table 4 note[a] reference.

(b) Values calculated from Table 3 and Table 4.

Attachment II

RG&E Specific Methodology for Determining LTOP Setpoints

3.0 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM (LTOPS)

3.1 INTRODUCTION

The purpose of the LTOPS is to supplement the normal plant operational administrative controls to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. The LTOPS also protects the Residual Heat Removal (RHR) System from overpressurization. This has been achieved by conservatively choosing an LTOPS setpoint which prevents the RCS from exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[4] requirements, and the RHR System from exceeding 110% of its design pressure. The LTOPS is designed to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the LTOPS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the low-range pressure channels, are utilized to mitigate potential RCS overpressure transients. The LTOPS provides the relief capacity for specific transients which would not be mitigated by the RHR System relief valve. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[4] allowable is imposed above a certain temperature so that the loads on the piping from a LTOPS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for LTOPS. Each of these scenarios assumes no RHR System heat removal capability. The RHR System relief valve (203) does not actuate during the transients. The first transient consists of a heat injection scenario in which a

reactor coolant pump in a single loop is started with the RCS temperature as much as 50°F lower than the steam generator secondary side temperature. This results in a sudden heat input to a water-solid RCS from the steam generators, creating an increasing pressure transient. The second transient has been defined as a mass injection scenario into a water-solid RCS as caused by one of two possible scenarios. The first scenario is an inadvertent acutation of the safety injection pumps into the RCS. The second scenario is the simultaneous isolation of the RHR System, isolation of letdown, and failure of the normal charging flow controls to the full flow condition. Either scenario may be eliminated from consideration depending on the plant configurations which are restricted by technical specifications. Also, various combinations of charging and safety injection flows may also be evaluated on a plant-specific basis. The resulting mass injection/letdown mismatch causes an increasing pressure transient.

3.2 LTOPS Setpoint Determination

Rochester Gas and Electric and Babcock & Wilcox Nuclear Technology (BWNT) have developed the following methodology which is employed to determine PORV setpoints for mitigation of the LTOPS design basis cold overpressurization transients. This methodology maximizes the available operating margin for setpoint selection while maintaining an appropriate level of protection in support of reactor vessel and RHR System integrity.

3.2.1 Parameters Considered

The selection of proper LTOPS setpoint for actuating the PORVs requires the consideration of numerous system parameters including:

a. Volume of reactor coolant involved in transient
b. RCS pressure signal transmission delay
c. Volumetric capacity of the relief valves versus opening position, including the potential for critical flow

d. Stroke time of the relief valves (open & close)

e. Initial temperature and pressure of the RCS and steam generator

f. Mass input rate into RCS

g. Temperature of injected fluid

h. Heat transfer characteristics of the steam generators

i. Initial temperature asymmetry between RCS and steam generator secondary water

j. Mass of steam generator secondary water

k. RCP startup dynamics

l. 10CFR50, Appendix G[4] pressure/temperature characteristics of the reactor vessel

m. Pressurizer PORV piping/structural analysis limitations

n. Dynamic and static pressure differences throughout the RCS and RHRS

o. RHR System pressure limits

p. Loop assymetry for RCP start cases

q. Instrument uncertainty for temperature (conditions under which the LTOP System is placed into service) and pressure uncertainty (actuation setpoint)

These parameters are modelled in the BWNT RELAP5/MOD2-B&W computer code (Ref. 19) which calculates the maximum and minimum system pressures.

3.2.2 Pressure Limits Selection

The function of the LTOPS is to protect the reactor vessel from fast propagating brittle fracture. This has been implemented by choosing a LTOPS setpoint which prevents exceeding the limits prescribed by the applicable pressure/temperature characteristic for the specific reactor vessel material in accordance with rules given in Appendix G to 10CFR50[4]. The LTOPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. Therefore, it is appropriate to utilize the steady-state Appendix G limit. In addition, the LTOPS also provides for an operational consideration to maintain the integrity of the PORV piping, and to protect the RHR System from overpressure

during the LTOPS design basis transients. A typical characteristic 10CFR50 Appendix G curve is shown by Figure 3.1 where the allowable system pressure increases with increasing temperature. This type of curve sets the nominal upper limit on the pressure which should not be exceeded during RCS increasing pressure transients based on reactor vessel material properties. Superimposed on this curve is the PORV piping limit and RHR System pressure limit which is conservatively used, for setpoint development, as the maximum allowable pressure above the temperature at which it intersects with the 10CFR50 Appendix G curve.

When a relief valve is actuated to mitigate an increasing pressure transient, the release of a volume of coolant through the valve will cause the pressure increase to be slowed and reversed as described by Figure 3.2. The system pressure then decreases, as the relief valve releases coolant, until a reset pressure is reached where the valve is signalled to close. Note that the pressure continues to decrease below the reset pressure as the valve recloses. The nominal lower limit on the pressure during the transient is typically established based solely on an operational consideration for the reactor coolant pump #1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. In the event that the available range is insufficient to concurrently accomodate the upper and lower pressure limits, the upper pressure limits are given preference.

The nominal upper limit (based on the minimum of the steady-state 10CFR50 Appendix G requirement, the RHR System pressure limit, and the PORV piping limitations) and the nominal RCP #1 seal performance criteria create a pressure range from which the setpoints for both PORVs may be selected as shown on Figures 3.3 and 3.4. Where there is insufficient range between the upper and lower pressure limits to select PORV setpoints to provide protection against violation of both limits, setpoint selection to provide protection against the upper pressure limit violation shall take precedence.

3.2.3 Mass Input Consideration

For a particular mass input transient to the RCS, the relief valve will be signalled to open at a specific pressure setpoint. However, as shown on Figure 3.2, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters, and results in a maximum system pressure somewhat higher than the set pressure. Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and to the delay in stroking the valve closed. The maximum and minimum pressures reached (P_{MAX} and P_{MIN}) in the transient are a function of the selected setpoint (P_S) as shown on Figure 3.3. The shaded area represents an optimum range from which to select the setpoint based on the particular mass input case. Several mass input cases may be run at various input flow rates to bound the allowable setpoint range.

3.2.4 Heat Input Consideration

The heat input case is done similarly to the mass input case except that the locus of transient pressure values versus selected setpoints may be determined for several values of the initial RCS temperature. This heat input evaluation provides a range of acceptable setpoints dependent on the reactor coolant temperature, whereas the mass input case is limited to the most restrictive low temperature condition only (i.e. the mass injection transient is not sensitive to temperature). The shaded area on Figure 3.4 describes the acceptable band for a heat input transient from which to select the setpoint for a particular initial reactor coolant temperature. If the LTOPS is a single setpoint system, the most limiting result is used throughout.

3.2.5 Final Setpoint Selection

By superimposing the results of multiple mass input and heat input cases evaluated, (from a series of figures such as 3.3 and 3.4) a range of allowable PORV setpoints to satisfy both conditions can be determined. For a single setpoint

system, the most limitting setpoint is chosen, with the upper pressure limit given precedence if both limits cannot be accomodated.

The selection of the setpoints for the PORVs considers the use of nominal upper and lower pressure limits. The upper limits are specified by the minimum of the steady-state cooldown curve as calculated in accordance with Appendix G to 10CFR50[4] or the peak RCS or RHR System pressure based upon piping/structural analysis loads. The lower pressure extreme is specified by the reactor coolant pump #1 seal minimum differential pressure performance criteria. Uncertainties in the pressure and temperature instrumentation utilized by the LTOPS are accounted for consistent with the methodology of Reference 20. Accounting for the effects of instrumentation uncertainty imposes additional restrictions on the setpoint development, which is already based on conservative pressure limits (such as a safety factor of 2 on pressure stress, use of a lower bound K_{IR} curve and an assumed ¼T flaw depth with a length equal to 1 ½ times the vessel wall thickness) as discussed in Section 2 of this report.

3.3 Application of ASME Code Case N-514

ASME Code Case N-514[17] allows low temperature overpressure protection systems (LTOP) to limit the maximum pressure in the reactor vessel to 110% of the pressure determined to satisfy Appendix G, paragraph G-2215, of Section XI of the ASME Code[5]. (Note, that the setpoint selection methodology as discussed in Section 3.2.5 specifically utilizes the steady-state curve.) The application of ASME Code Case N-514 increases the operating margin in the region of the pressure-temperature limit curves where the LTOPS system is enabled. Code Case N-514 requires LTOPS to be effective at coolant temperatures less than 200°F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater. 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 vessel inside surface, as determined by Regulatory Guide 1.99, Revision 2. Although expected soon, use of Code Case N-514 has not yet been formally approved by the NRC. In the interim, an exemption

to the regulations must be granted by the NRC before Code Case N-514 can be used in the determination of the LTOPS setpoint(s) and enable temperature.

3.4 Enable Temperature for LTOPS

The enable temperature is the temperature below which the LTOPS system is required to be operable. The definition of the enabling temperature currently approved and supported by the NRC is described in Branch Technical Position RSB 5-2[18]. This position defines the enable temperature for LTOP systems as the water temperature corresponding to a metal temperature of at least RT_{NDT} + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations. This definition is mostly based on material properties and fracture mechanics, with the understanding that material temperatures of RT_{NDT} + 90°F at the critical location will be well up the transition curve from brittle to ductile properties, and therefore brittle fracture of the vessel is not expected.

The ASME Code Case N-514 supports an enable temperature of RT_{NDT} + 50°F or 200°F, whichever is greater as described in Section 3.3. This definition is also supported by Westinghouse and can be used by requesting an exemption to the regulations or when ASME Code Case N-514 is formally approved by the NRC.

The RCS cold leg temperature limitation for starting an RCP is the same value as the LTOPS enable temperature to ensure that the basis of the heat injection transient is not violated. The Standard Technical Specifications (STS) prohibit starting an RCP when any RCS cold leg temperatures is less than or equal to the LTOPS enable temperature unless the secondary side water temperature of each steam generator is less than or equal to 50°F above each of the RCS cold leg temperatures.

Figure 3.1

TYPICAL APPENDIX G
P/T CHARACTERISTICS

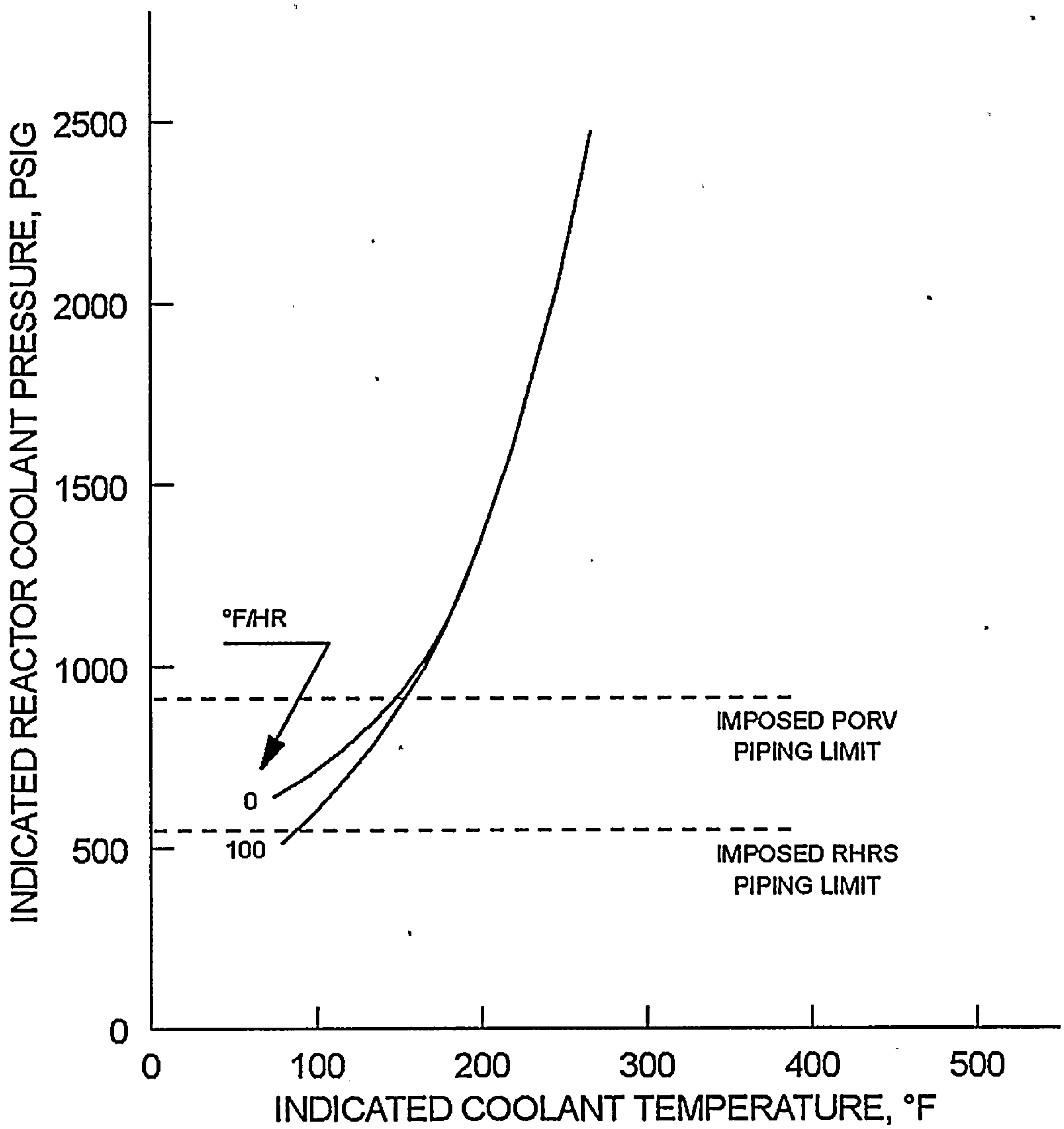


Figure 3.2

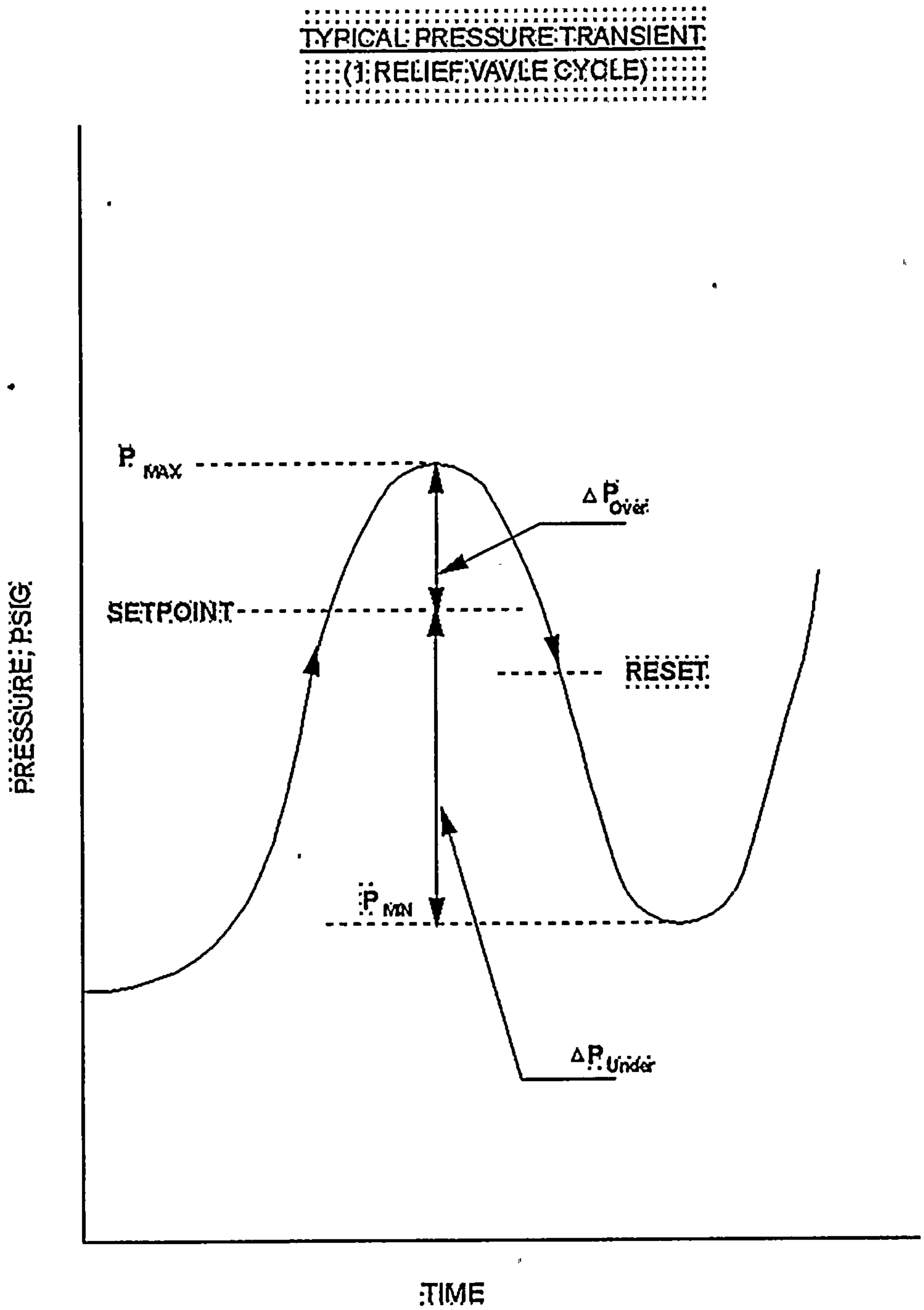
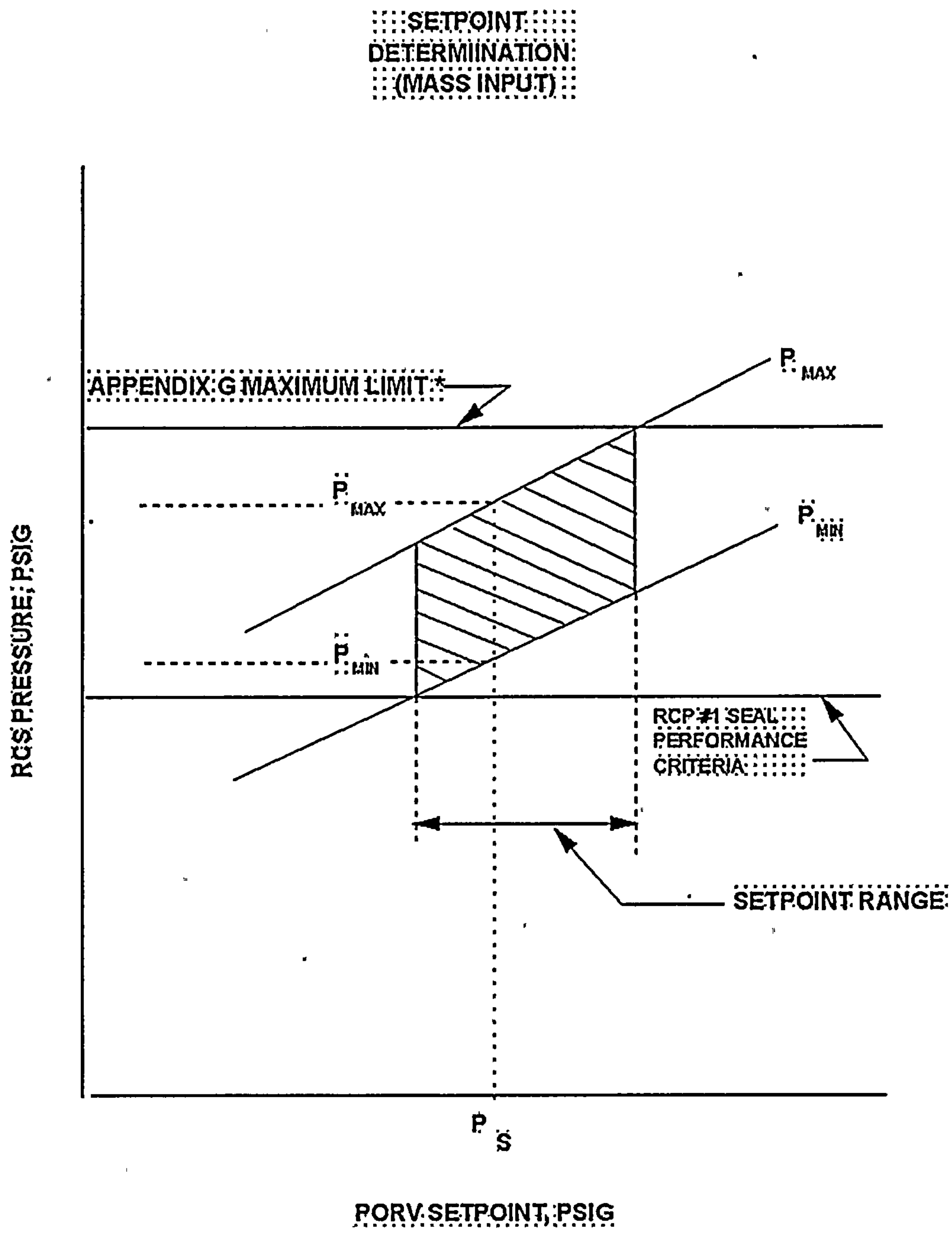
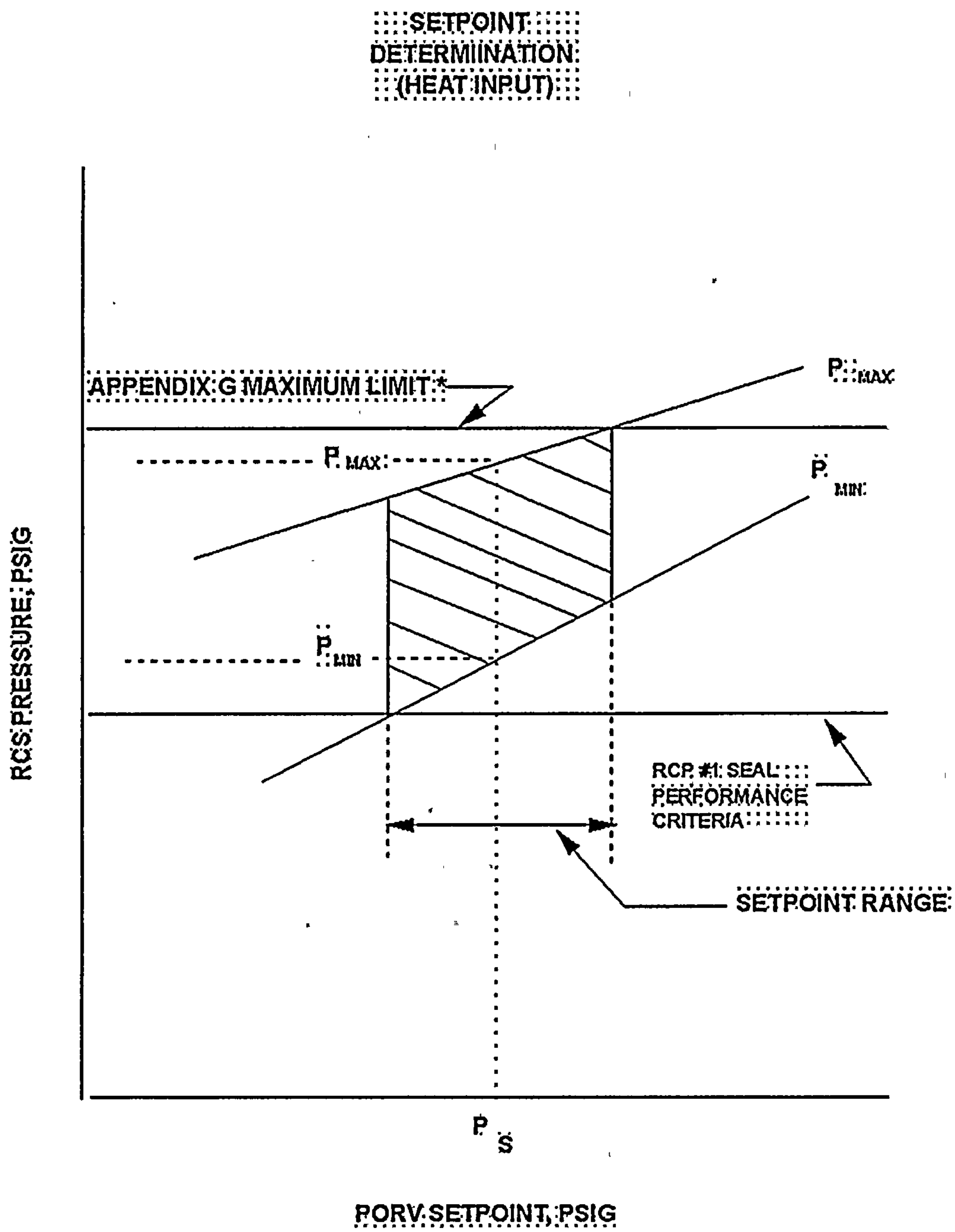

TYPICAL PRESSURE TRANSIENT
(1 RELIEF VAVLE CYCLE)
P MAX
Δ P Over
SETPOINT
RESET
PRESSURE, PSIG
P MIN
Δ P Under
TIME

Figure 3.3



* The maximum pressure limit is the minimum of the Appendix G limit, the PORV discharge piping structural analysis limit, or the RHR system limit

Figure 3.4



* The maximum pressure limit is the minimum of the Appendix G limit, the PORV discharge piping structural analysis limit, or the RHR system limit

4.0 REFERENCES

1. NUREG 1431, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors", Revision 0, September, 1992.

2. U.S. Nuclear Regulatory Commission, "Removal of Cycle-Specific Parameter Limits from Technical Specifications", Generic Letter 88-16, October, 1988.

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

4. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors", Appendix G, Fracture Toughness Requirements.

5. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G, Fracture Toughness Criteria For Protection Against Failure.

6. R. G. Soltesz, R. K. Disney, J. Jedruch, and S. L. Ziegler, Nuclear Rocket Shielding Methods, Modification, Updating and Input Data Preparation. Vol. 5--Two-Dimensional Discrete Ordinates Transport Technique, WANL-PR(LL)-034, Vol. 5, August 1970.

7. ORNL RSIC Data Library Collection DLC-76 SAILOR Coupled Self-Shielded, 47 Neutron, 20 Gamma-Ray, P3, Cross Section Library for Light Water Reactors.

8. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components", Division 1, Subsection NB: Class 1 Components.

9. Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements", NUREG-0800 Standard Review Plan 5.3.2, Pressure-Temperature Limits, July 1981, Rev. 1.

10. ASTM E-208, Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Standards, Section 3, American Society for Testing and Materials.

11. B&W Owners Group Report BAW-2202, "Fracture Toughness Characterization of WF-70 Weld Material", B&W Owners Group Materials Committee, September 1993.

12. Letter, Clyde Y. Shiraki, Nuclear Regulatory Commission, to D. L. Farrar, Commonwealth Edison Company, "Exemption from the Requirement to Determine the Unirradiated Reference Temperature in Accordance with the Method Specified in 10 CFR 50.61(b) (2) (i) (TAC NOS. M84546 and M84547)", Docket Nos. 50-295 and 50-304, February 22, 1994.

13. Code of Federal Regulations, Title 10, Part 50, "Fracture Toughness Requirements for Light-Water Nuclear Power Reactors", Appendix H, Reactor Vessel Material Surveillance Program Requirements.

14. Timoshenko, S. P. and Goodier, J. N., Theory of Elasticity, Third Edition, McGraw-Hill Book Co., New York, 1970.

15. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix A, Analysis of Flaws, Article A-3000, Method For K_I Determination.

16. WRC Bulletin No. 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials", Welding Research Council, New York, August 1972.

17. ASME Boiler and Pressure Vessel Code Case N-514, Section XI, Division 1, "Low Temperature Overpressure Protection", Approval date: February 12, 1992.

18. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures", NUREG-0800 Standard Review Plan 5.2.2, Overpressure Protection, November 1988, Rev. 2.

19. BWNT, "RELAPS/MOD2, An Advanced Computer Program for Light-Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164P-A.

20. Instrument of America (ISA) Standard 67.04-1994.

REDLINED VERSION

3.0 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM (LTOPS)

3.1 INTRODUCTION

The purpose of the LTOPS is to supplement the normal plant operational administrative controls to protect the reactor vessel from being exposed to conditions of fast propagating brittle fracture. The LTOPS also protects the Residual Heat Removal (RHR) System from overpressurization. This has been achieved by conservatively choosing an LTOPS setpoint which prevents the RCS from exceeding the pressure/temperature limits established by 10 CFR Part 50 Appendix G[4] requirements, and the RHR System from exceeding 110% of its design pressure. The LTOPS is designed to provide the capability, during relatively low temperature operation (typically less than 350°F), to automatically prevent the RCS pressure from exceeding the applicable limits. Once the system is enabled, no operator action is involved for the LTOPS to perform its intended pressure mitigation function. Thus, no operator action is modelled in the analyses supporting the setpoint selection, although operator action may be initiated to ultimately terminate the cause of the overpressure event.

The PORVs located near the top of the pressurizer, together with additional actuation logic from the low-range pressure channels, are utilized to mitigate potential RCS overpressure transients. The LTOPS provides the relief capacity for specific transients which would not be mitigated by the RHR System relief valve. In addition, a limit on the PORV piping is accommodated due to the potential for water hammer effects to be developed in the piping associated with these valves as a result of the cyclic opening and closing characteristics during mitigation of an overpressure transient. Thus, a pressure limit more restrictive than the 10CFR50, Appendix G[4] allowable is imposed above a certain temperature so that the loads on the piping from a LTOPS event would not affect the piping integrity.

Two specific transients have been defined, with the RCS in a water-solid condition, as the design basis for LTOPS. Each of these scenarios assumes no RHR System heat removal capability. The RHR System relief valve (203) does not actuate during the transients. The first transient consists of a heat injection scenario in which a