ATTACHMENT A

Beaver Valley power Station, Unit No. 1
Proposed Technical Specification Change No. 191

Revise the Technical Specifications as follows:

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B 3/4 4-1a	B 3/4 4-1a

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REACTOR COOLANT PUMP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 If both OPPS PORV's are not OPERABLE, an idle reactor coolant pump in a non-isolated loop shall not be started, unless:

1. The actual pressurizer water level is less than 60 percent (840 ft^3), ~~or~~ and
2. The secondary water temperature * of each steam generator is less than 25°F above each of the in-service RCS cold leg temperatures.

APPLICABILITY: When the temperature of one or more of the non-isolated loop cold legs is ≤ the enable temperature setforth in Specification 3.4.9.3

ACTION:

With the pressurizer water level greater than or equal to 60 percent or the temperature of the steam generator in the loop associated with the reactor coolant pump being started greater than or equal to 25° above the cold leg temperature of the other non-isolated loops, suspend the startup of the reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The pressurizer water volume ~~or~~ and the secondary water temperature of the non-isolated steam generators shall be determined within ten minutes prior to starting a reactor coolant pump.

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* The secondary water temperature is to be verified by direct measurement of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping.

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B6607-2

RT_{NDT} AFTER 9.5 EFPY: 1/4T, 202°F

3/4T, 176°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 9.5 EFPY.

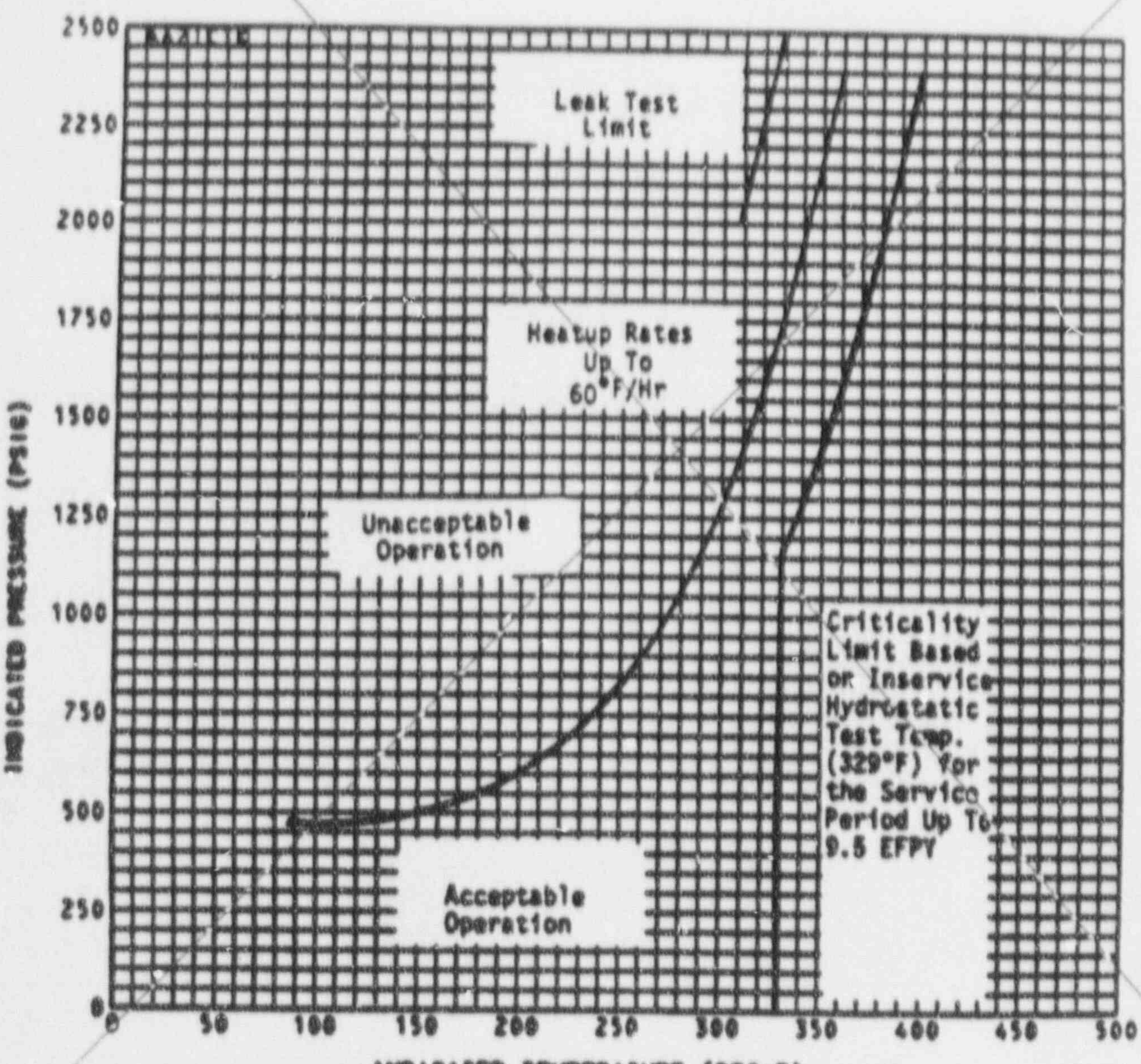

FIGURE 3.4-2 Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 9.5 EFPY

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B 6607-2

RT_{NDT} AFTER 9.5 EFPY: 1/4T, 202°F

3/4T, 176°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 9.5 EFPY.

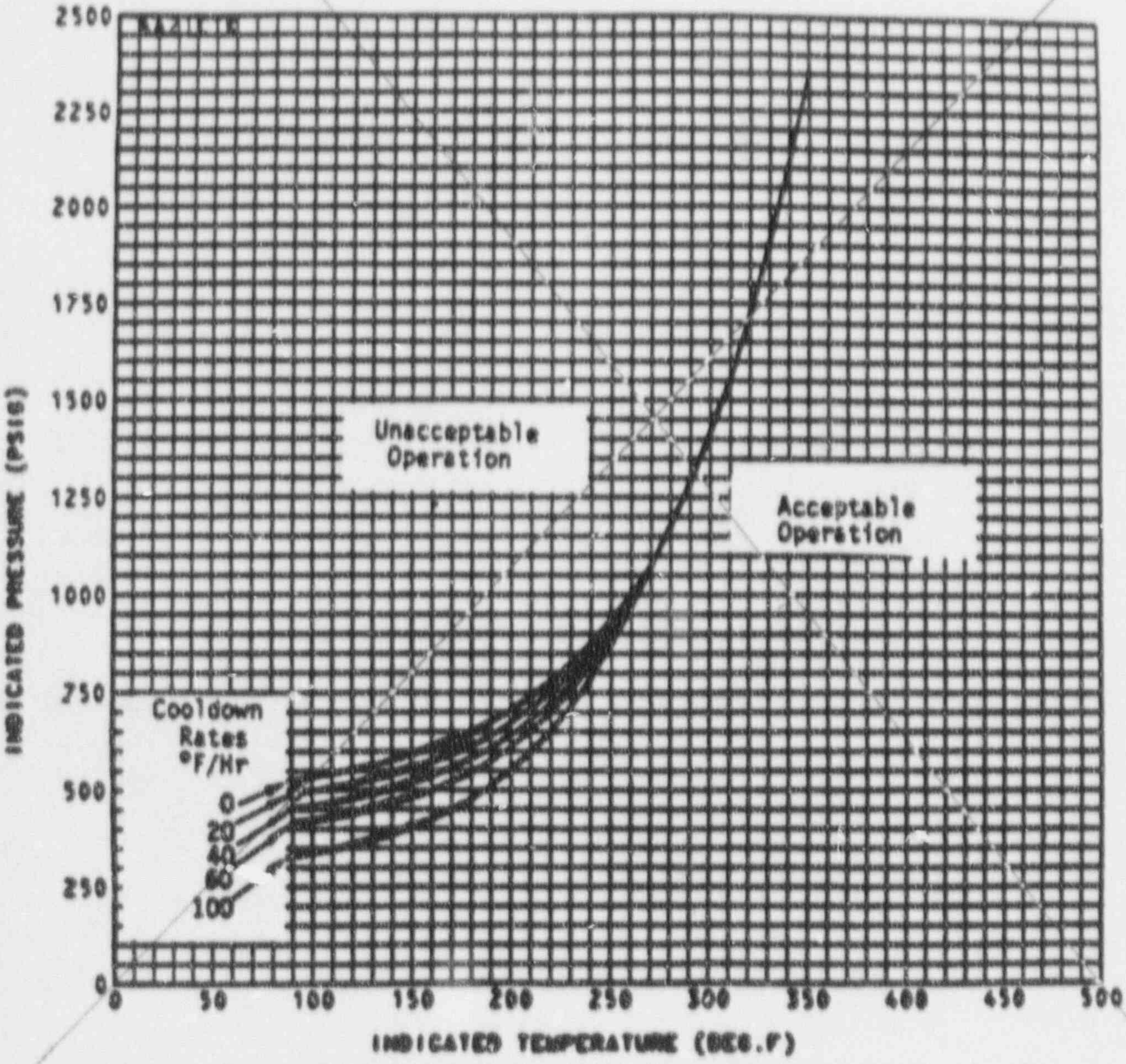

FIGURE 3.4-3 Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 9.5 EFPY

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a nominal trip setpoint of ≤ ~~444~~ ~~436~~ 432 psig, or

b. A reactor coolant system vent of ≥ 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the non-isolated RCS cold legs is ≤ an enable temperature of ~~292~~ 314°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status. Refer to Technical Specification 3.4.1.6 for further limitations.

b. With both PORV's inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENT

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE BY:

3/4.4.1 REACTOR COOLANT LOOPS, (continued)

of Appendix G by ~~either~~ (1) restricting the water level in the pressurizer and thereby providing a volume for the primary coolant to expand into ~~or~~ and (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 25°F above each of the RCS cold leg temperatures.

Power is removed from the isolated loop stop valves (hot leg and cold leg) to ensure that no reactivity addition to the core can occur while the loop is isolated due to inadvertent opening of the isolated loop stop valves. Isolated loop startup is limited to Modes 5 and 6 in accordance with the NRC SER on N-1 loop operation. Verification of the isolated loop boron concentration prior to opening the isolated loop stop valves provides a reassurance of the adequacy of the shutdown margin in the remainder of the system. Restoration of power to the hot leg stop valve allows opening this valve to complete the recirculation flowpath in conjunction with the relief line bypassing the cold leg stop valve and ensures adequate mixing in the isolated loop. This enables the temperature and boron concentration of the isolated loop to be brought to equilibrium with the remainder of the system. Limiting the temperature differential between the isolated loop and the remainder of the system prior to opening the cold leg stop valve prevents any significant reactivity effects due to cool water addition to the core.

Startup of an idle ~~lop~~ loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to

PROPOSED

ATTACHMENT B

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 191
Revision of technical Specification 3.4.9.1 and 3.4.9.3
HEATUP AND COOLDOWN CURVES AND OPPS SETPOINT

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would extend the applicability of the heatup and cooldown curves to 16 effective full power years (EFPY) and modify the overpressure protection system (OPPS) pressure setpoint and enable temperature to incorporate analysis results applicable to 16 EFPY. In addition, Specification 3.4.1.6.1, Surveillance Requirement 4.4.1.6.1, and Bases 3/4.4.1 have been revised by changing "or" to "and."

B. BACKGROUND

The proposed changes replace the current heatup and cooldown curves applicable to 9.5 EFPY, with curves applicable to 16 FFPY. The new curves were developed by Westinghouse in accordance with Attachment E, "Duquesne Light Company Beaver Valley Unit 1 Life Attainment Plan." The curves provided in this plan have been developed in accordance with Regulatory Guide 1.99, Revision 2.

Changes to the OPPS pressure setpoint and enable temperature were provided by our request for Technical Specification Change No. 176, Revision 1 (TAC 76889) and are applicable to 9.5 EFPY. The new OPPS setpoint and enable temperature are applicable to 16 EFPY. The new OPPS pressure setpoint was selected from the 16 EFPY curve developed by Westinghouse in accordance with Attachment F, "Beaver Valley Unit 1 Low Temperature Overpressure Protection System (LTOPS) Setpoint Analysis at 16, 24, 32 and 48 EFPY." The new enable temperature was determined in accordance with the criteria provided in Standard Review Plan (SRP) Section 5.2.2 Branch Technical Position RSB 5-2.

During the review of a previous technical specification change request, it was identified that a change to Specification 3.4.1.6 should be incorporated to ensure the requirements of both conditions in the limiting condition for operation are satisfied.

C. JUSTIFICATION

The heatup and cooldown curves (Figures 3.4-2 and 3.4-3) have been updated to extend the applicable requirements to 16 EFPY. The curves applicable to 16 EFPY were developed in accordance with the same methodology used to develop the curves presently used. This will extend the applicability of these curves beyond the schedule for removal of the next surveillance capsule at 15 EFPY.

Specification 3.4.9.3 has been updated by providing a new OPPS pressure setpoint and enable temperature. The new OPPS pressure setpoint of 436 psig was selected from Attachment F Table 2 at 16 EFPY and 85°F. This is the most conservative setpoint defined by the 16 EFPY curve provided on Attachment F Figure 3. This setpoint has been reduced by 4°F to 432°F to address a nonconservatism identified in the new steam generator tube plugging analysis. The new OPPS enable temperature of 314°F was determined in accordance with the relationship RT_{NDT} + 90°F provided in SRP section 5.2.2 Branch Technical Position RSB 5-2. As shown on the 16 EFPY heatup and cooldown curves the limiting RT_{NDT} is at the 1/4t location and is 224°F, therefore, the new enable temperature is 224°F +90°F = 314°F.

The intent of Specification 3.4.1.6 is to provide an alternate set of conditions in the event a reactor coolant pump must be started when both PORV's are not operable. Specification 3.4.1.6.1, Surveillance Requirement 4.4.1.6.1, and Bases 3/4.4.1 have been revised from "or" to "and" to ensure both conditions are in effect to mitigate the consequences of starting a reactor coolant pump. The action statement of specification 3.4.1.6 has been revised to state "greater than or equal to" for both the pressurizer level and temperature differential to address the full range of setpoint conditions.

D. SAFETY ANALYSIS

Attachment E was developed by Westinghouse to provide additional heatup and cooldown curves for vessel exposures extending to 48 EFPY. The current heatup and cooldown curves are applicable to 9.5. Plant operation will exceed 9.5 EFPY during Cycle 9, therefore, new curves applicable to 16 EFPY and determined in accordance with the same methodology used to produce the 9.5 EFPY curves were developed by Westinghouse. These curves will be used to address plant operation until new curves can be generated based on the examination of the next capsule removed.

The new OPPS pressure setpoint and enable temperature were determined using the same methodology used to determine the values provided in Technical Specification Change No. 176 Revision 1 (TAC 76889). Attachment F was developed by Westinghouse to provide additional OPPS setpoints for vessel exposures extending to 48 EFPY. These OPPS curves could be used in determining applicable setpoints based on current plant pressure and temperature conditions, however, the plant does not have an installed system to automatically perform this function. Technical Specification Change No. 176 Revision 1 incorporated a single pressure setpoint based on the lowest pressure on the 9.5 EFPY curve at 85°F. This setpoint has been updated by incorporating a single pressure setpoint based on the lowest pressure on the 16 EFPY curve at 85°F, as shown on Attachment F Table 2, this is 436 psig. A nonconservatism was identified during the new steam generator tube plugging analysis performed by Westinghouse, therefore, to eliminate this condition the setpoint has been reduced to 432 psig.

The OPPS enable temperature is based on the limiting reactor vessel material RT_{NDT}. As shown on the new heatup and cooldown curves the RT_{NDT} is 224°F, this is different from the 202°F value on the 9.5 EFPY, therefore, the OPPS enable temperature has been updated to incorporate a new value applicable to 16 EFPY.

The consequences of a heat input transient caused by the addition of energy from the secondary system when starting a reactor coolant pump can be mitigated without using the OPPS. The reactor coolant system will be protected against overpressure transients and will not exceed the limits of 10 CFR 50 Appendix G by (1) restricting the pressurizer water level to provide a volume for the primary coolant to expand into, and (2) restricting the temperature differential between the reactor coolant system and the steam generator to limit the addition of energy transferred to the reactor coolant system. Changing the Limiting Condition for Operation, the surveillance requirement and the Bases from "or" to "and" ensures the required conditions are satisfied in accordance with the accident analyses. The action statement conditions of specification 3.4.1.6 have been changed from "greater than" to "greater than or equal to." Specification 3.4.1.6 requires pressurizer level less than 60 percent and temperature differential less than 25°F, therefore, the action statement has been revised to ensure the specification addresses the full range of setpoint conditions including these at, above and below the specified setpoint.

Based on the above considerations, these changes reflect the application of methodologies recognized by the NRC and the industry as providing a sufficient margin of safety. The fracture toughness requirements of 10 CFR 50 Appendix G are satisfied and conservative operating restrictions are applied in the proposed heatup and cooldown curves, therefore, these changes are considered to be safe and will not reduce the safety of the plant.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated, focusing on the three standards set forth in 10 CFR 50.92(c) as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

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

The heatup and cooldown curves have been revised to extend their applicability from 9.5 EFPY to 16 EFPY. The new heatup and cooldown curves were developed in accordance with the methodology provided in Regulatory Guide 1.99 Revision 2. This is consistent with the methodology used to determine the current curves.

The OPPS setpoint and enable temperature have also been updated to extend their applicability to 16 EFPY. An analysis was performed by Westinghouse to provide additional OPPS setpoints for vessel exposures extending to 48 EFPY. These OPPS curves could be used in determining applicable setpoints based on current plant pressure and temperature conditions, however, the plant does not have an installed system to automatically perform this function. Technical Specification Change No. 176 Revision 1 incorporated a single pressure setpoint based on the lowest pressure on the 9.5 EFPY curve at 85°F. This setpoint has been updated by incorporating a single pressure setpoint based on the lowest pressure on the 16 EFPY curve at 85°F, this results in a setpoint of 436 psig. A nonconservatism in the OPPS setpoint was identified during the new steam generator tube plugging analysis, therefore, Westinghouse has determined that reducing the setpoint by 4°F to 432°F will eliminate this condition.

The OPPS enable temperature is based on the limiting reactor vessel material RT_{NDT}. The new heatup and cooldown curves specify a RT_{NDT} of 224°F versus 202°F on the current curves. Section 5.2.2 of the Standard Review Plan includes Branch Technical Position RSB 5-2 which provides guidance for determining the OPPS enable temperature. This method was used and results in an enable temperature of 224°F + 90°F = 314°F.

The consequences of a heat input transient caused by the addition of energy from the secondary system when starting a reactor coolant pump can be mitigated without using the OPPS. The reactor coolant system will be protected against overpressure transients and will not exceed the limits of 10 CFR 50 Appendix G by (1) restricting the pressurizer water level to provide a volume for the primary coolant to expand into, and (2) restricting the temperature differential between the reactor coolant system and the steam generator to limit the addition of energy transferred to the reactor coolant system. Changing the Limiting Condition for Operation, the surveillance requirement and the Bases from "or" to "and" ensures the required conditions are satisfied in accordance with the accident analyses. The action

statement conditions of Specification 3.4.1.6 have been changed from "greater than" to "greater than or equal to." Specification 3.4.1.6 requires pressurizer level less than 60 percent and temperature differential less than 25°F, therefore, the action statement has been revised to ensure the specification addresses the full range of setpoint conditions including those at, above and below the specified setpoint.

These changes were determined in accordance with the methodolo set forth in the regulations to provide an adequate rgin of safety to ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events. Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

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

The new heatup and cooldown curves were developed in accordance with the methodology used to determine the current curves and are consistent with the methodology set forth in the regulations. The new OPPS pressure setpoint was selected from the most conservative position on the 16 EFPY low temperature overpressure protection setpoint curve to ensure sufficient margin is available to prevent violation of the pressure-temperature limits due to anticipated mass and heat input transients. The new enable temperature provides a wider range over which the OPPS is active and was determined in accordance with the regulations.

The intent of Specification 3.4.1.6 is to provide an alternate set of conditions in the event a reactor coolant pump must be started when both PORV's are not operable. Specification 3.4.1.6.1, Surveillance Requirement 4.4.1.6.1, and Bases 3/4.4.1 have been revised from "or" to "and" to ensure both conditions are in effect to mitigate the consequences of starting a reactor coolant pump.

These changes are consistent with the regulations and will not affect the reliability of the reactor vessel or the plant heatup and cooldown procedures. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The revised heatup and cooldown curves, OPPS pressure setpoint, enable temperature, and change to Specification 3.4.1.6 will continue to ensure the reactor coolant system will be protected from pressure transients at low temperatures. The proposed changes will not reduce the reliability of the OPPS, nor will they increase the likelihood of vessel damage or failure in the event of an overpressure transient. These changes are established in accordance with current regulations and the latest regulatory guidance. Plant operation will be maintained within required limits, therefore, the reactor vessel materials will behave in a non-brittle manner to remain within the plant design basis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfies the no significant hazards consideration standards of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL EVALUATION

The proposed changes have been evaluated and it has been determined that the changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed changes meet the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed changes is not required.

H. UFSAR CHANGES

Reference to Attachment F has been added to UFSAR section 4.2.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Change No. 191

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REACTOR COOLANT SYSTEM

REACTOR COOLANT PUMP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.6 If both OPPS PORV's are not OPERABLE, an idle reactor coolant pump in a non-isolated loop shall not be started, unless:

1. The actual pressurizer water level is less than 60 percent (840 ft^3), and
2. The secondary water temperature* of each steam generator is less than 25°F above each of the in-service RCS cold leg temperatures.

APPLICABILITY: When tne temperature of one or more of the non-isolated loop cold legs is ≤ the enable temperature setforth in Specification 3.4.9.3.

ACTION:

With the pressurizer water level greater than or equal to 60 percent or the temperature of the steam generator in the loop associated with the reaccor coolant pump being started greater than or equal to 25° above the cold leg temperature of the other non-isolated loops, suspend the startup of the reactor coolant pump.

SURVEILLANCE REQUIREMENTS

4.4.1.6.1 The pressurizer water volume and the secondary water temperature of the non-isolated steam generators shall be determined within ten minutes prior to starting a reactor coolant pump.

* The secondary water temperature is to be verified by direct measurement of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping.

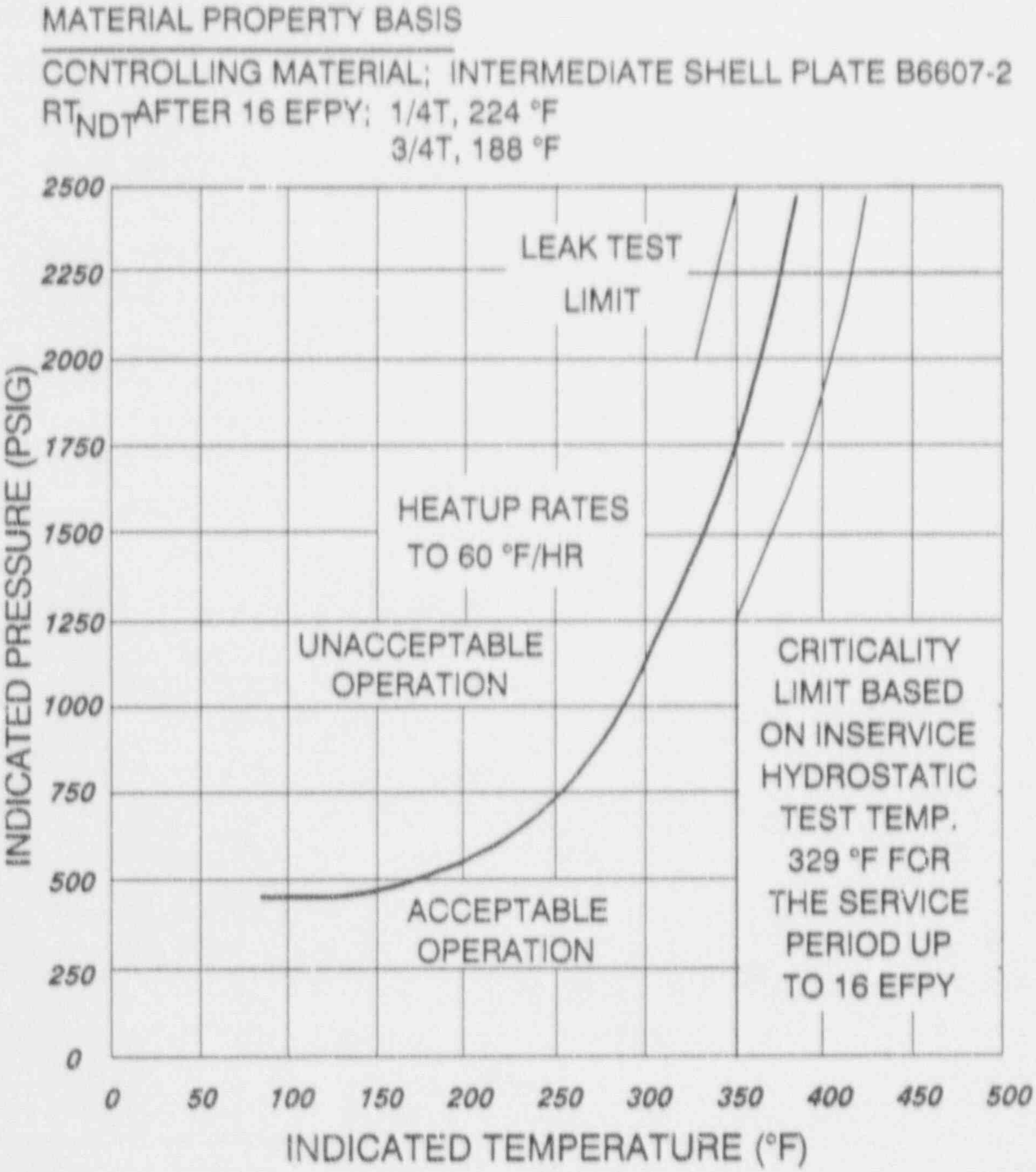

FIGURE 3.4-2

Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First 16 EFPY

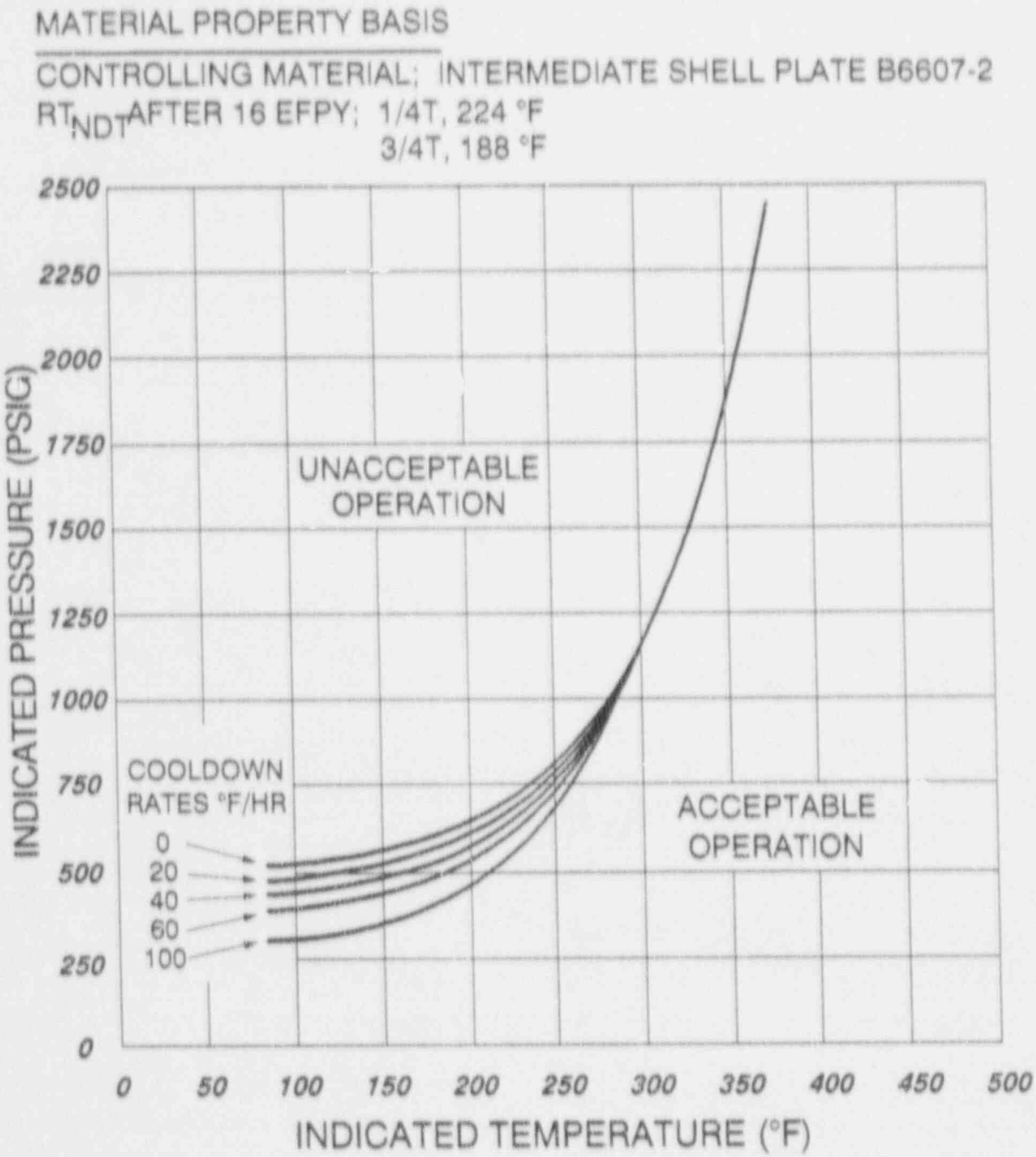

FIGURE 3.4-3

Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 16 EFPY

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. Two power operated relief valves (PORVs) with a nominal trip setpoint of ≤ 432 psig, or

b. A reactor coolant system vent of ≥ 3.14 square inches.

APPLICABILITY: When the temperature of one or more of the non-isolated RCS cold legs is ≤ an enable temperature of 314°F.

ACTION:

a. With one PORV inoperable, either restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch vent(s) within the next 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status. Refer to Technical Specification 3.4.1.6 for further limitations.

b. With both PORV's inoperable, depressurize and vent the RCS through a 3.14 square inch vent(s) within 12 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENT

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE BY:

3.4.4.1 REACTOR COOLANT LOOPS, (continued)

of Appendix G by (1) restricting the water level in the pressurizer and thereby providing a volume for the primary coolant to expand into and (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 25°F above each of the RCS cold leg temperatures.

Power is removed from the isolated loop stop valves (hot leg and cold leg) to ensure that no reactivity addition to the core can occur while the loop is isolated due to inadvertent opening of the isolated loop stop valves. Isolated loop startup is limited to Modes 5 and 6 in accordance with the NRC SER on N-1 loop operation. Verification of the isolated loop boron concentration prior to opening the isolated loop stop valves provides a reassurance of the adequacy of the shutdown margin in the remainder of the system. Restoration of power to the hot leg stop valve allows opening this valve to complete the recirculation flowpath in conjunction with the relief line bypassing the cold leg stop valve and ensures adequate mixing in the isolated loop. This enables the temperature and boron concentration of the isolated loop to be brought to equilibrium with the remainder of the system. Limiting the temperature differential between the isolated loop and the remainder of the system prior to opening the cold leg stop valve prevents any significant reactivity effects due to cool water addition to the core.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point. The relief capacity of a single safety valve is adequate to

ATTACHMENT D

Beaver Valley Power Station, Unit No.1

Proposed Technical Specification Change No. 191

UFSAR CHANGES

References for Section 4.2

1. Ernest L. Robinson, "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the ASME, (July, 1944).

2. D. H. Winne, B. M. Wundt, "Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies", ASME (December 1, 1957).

3. J. W. Murdock, "Performance Characteristics of Elbow Flowmeteres," Transactions of the ASME, (September, 1964).

4. J. J. Szyslowski, R. Salvatori, "Determination of Design Pipe Breaks for Westinghouse Reactor Coolant Systems" WCAP-7503 Revision 1, Westinghouse Electric Corporation (February, 1972).

5. Duquese Light Company to NRC submittal concerning NUREG-0737, Item II.D.1 Pressurizer Safety and Relief Line Piping and Support Evaluation dated June 24, 1983.

6. Duquesne Light Company to NF submittal concerning NUREG-0737, Item II.D.1 Plant Specific Report dated July 1, 1982.

7. Beaver Valley Unit 1 Low Temperature Overpressure Protection System (LTOPS) Setpoint Analysis at 16, 24, 32 and 48 EFPY, Westinghouse Letter DLW-90-528, January 23, 1990.

ATTACHMENT E

Beaver Valley Power Station, Unit No.1

Proposed Technical Specification Change No. 191

Duquesne Light Company

Beaver Valley Unit 1

Life Attainment Plan

DUQUESNE LIGHT COMPANY
BEAVER VALLEY UNIT 1
LIFE ATTAINMENT PLAN

N. K. Ray
J. M. Chicots

February 1990

Approved by: [signature]
T. A. Meyer, Manager
Structural Materials & Reliability Technology

Work Performed For Duquesne Light Company

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear and Advanced Technology Division
P.O. Box 2728
Pittsburgh, Pennsylvania 15230-2728

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1.0 INTRODUCTION

Neutron embrittlement represents the most significant damage mechanism that could potentially limit the lifetime of the reactor vessel. This report will examine the various issues which are affected by irradiation damage, and examine the potential need for flux reductions to demonstrate reactor vessel integrity for different time spans, including life extension of the Beaver Valley Unit 1 Reactor vessel.

RT_{PTS} values were calculated using PTS rule [1] and Regulatory Guide 1.99, Revision 2 [2] using material chemistry of Table 1-1. Table 4-1, indicates that the lower plate B6903-1 is the most limiting material for the Pressurized Thermal shock evaluation. RT_{PTS} Vs fluence plots are developed using PTS rule and Reg. guide 1.99, Rev. 2 methodology and are shown in Figures 4-2 and 4-3. These figures are applicable as long as (1) material chemistry remains unchanged and (2) the rule for calculation of PTS remains the same.

The calculation for RT_{NDT} for the beltline region of the Beaver Valley Unit 1 reactor vessel indicates that the limiting material for developing heatup and cooldown curves, are the lower plate B6903-1 and intermediate plate B6607-2 for 16, 24, 32 and 48 Effective Full Power Years (EFPY). RT_{NDT} values for the limiting material is reported in Table 4-2.

The flux reduction goals were carried out to insure 1) the RT_{PTS} values for the Beaver Valley Unit 1 reactor vessel materials remain below the screening criteria for thermal shock and 2) comfortable margin (identified by the Beaver Valley Unit 1 personnel) remains up to the end of 48 EFPY.

Table 4-1 indicates that the RT_{PTS} value for B6903-1 for 48 EFPY is above the PTS screening criteria using PTS rule. For this, flux reduction is needed only for the lower plate B6903-1 using PTS rule. Using Reg. Guide 1.99, Rev. 2, it has been shown in Table 4-1 that the PTS values for the Beaver Valley Unit 1 remain below the PTS screening criteria up to 48 EFPY.

Also, it has been revealed by the Duquesne Light personnel that by adjusting the low temperature overpressurization (LTOP) system set point, they will have comfortable margin to operate tne reactor vessel up to the end of 48 EFPY, assuming i) no increase in flux and ii) no significant increase in material chemistry of the belt line region.

TABLE 1-1

BEAVER VALLEY UNIT 1 REACTOR VESSEL BELTLINE REGION MATERIAL PROPERTIES

	Cu (Wt.%)	Ni (Wt.%)	CF	I[(a)] (°F)	M[(b)] (°F)
Int. Plate, B6607-1	.14	.62	100.50	43	34
Int. Plate, B6607-2	.14	.62	100.50	73	34
Lower Plate, B6903-1	.20	.54	141.80 (167.9)[c]	27	34 17[(d)]
Lower Plate, B7203-2	.14	.57	98.65	20	34
Long. Weld, 305424	.28	.63	191.65 (191.4)[c]	-56	65.5 44.05[(d)]
Long. Weld, 305414	.34	.07	210.45	-56	65.5
Circ. Weld, 90136	.29	.61	132.90	-56	65.5

(a) The initial RT_{NDT} (I) values for the forgings are measured and initial RT_{NDT} value for the weld is generic.

(b) Margin (M) as per Reg. Guide 1.99, rev. 2; the standard deviation for the initial RT_{NDT} margin term is assumed to be zero since the initial RT_{NDT} values were obtained from conservative (i.e., "upper bound") test results.

(c) Numbers in ( ).corresponds to surveillance capsule data.

(d) σ_Δ is cut into half, when Surveillance Capsule Data is used.

2.0 DESCRIPTION OF KEY ISSUES

The reactor vessel issues that were addressed are pressurized thermal shock (PTS), low upper shelf fracture toughness, and operating limitations (i.e. heatup and cooldown pressure-temperature limits). All of these issues are associated with irradiation embrittlement of the reactor vessel materials.

Figure 2-1 depicts the effects of irradiation on the charpy V-notch fracture toughness of reactor vessel steels. One effect is the shift in transition temperature from ductile (high toughness) to non-ductile behavior. The other effect is the drop in fracture toughness at higher temperatures (i.e., the upper shelf). A significant increase in transition temperature raises a PTS concern and causes a potential restriction in pressure-temperature limitations during heatup and cooldown of the vessel and plant. A low upper shelf impact energy raises concerns relative to both normal and abnormal conditions, including PTS events. Large transition temperature shifts are generally associated with weldments containing high copper content while low shelf behavior is also generally associated with high copper weldments, but only for specific types of welds.

The PTS Rule [1] outlines regulations to address the potential for PTS events on pressurized water reactor vessels. PTS events are transients that result in a rapid and severe cooldown in the primary system coincident with a high or increasing system pressure. The PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a large transition temperature shift exists because of neutron irradiation. Such an event may produce the propagation of flaws postulated to exist near the inner surface of the reactor vessel, thereby potentially affecting the integrity of the vessel. Screening criteria for shifts in transition temperature for reactor vessel materials have been prescribed in the PTS rule such that operation above those criteria requires extensive analyses to defend safe continued operation.

10CFR50 Appendix G [3] contains a minimum upper shelf impact-energy requirement of 50 ft-lb. Non-compliance with this criterion requires a volumetric examination of the beltline region, additional evidence of material toughness from supplemental fracture toughness tests, and a safety analysis demonstrating continued safe operation. The Beaver Valley Unit 1 reactor vessel surveillance capsule exhibits a more than adequate shelf level for continued safe plant operation (WCAP-12005) [5].

Figure 2-2 gives an example of heatup and cooldown operational limit curves that were generated in accordance with the ASME Code [4], which is made mandatory by 10CFR50 Appendix G [3]. This figure also exemplifies how the pressure-temperature limits become more restrictive with increasing neutron embrittlement of the reactor vessel; i.e., the operating window between competing limits can become very small at the lower temperatures.

Another underlying concern regarding the shift in transition temperature is the uncertainty in defining the material chemistry that is used in predicting future transition temperature shifts and in the irradiation damage prediction formulations themselves. The following sections discuss the uncertainties that were taken into account in the definition of neutron flux reduction goals.

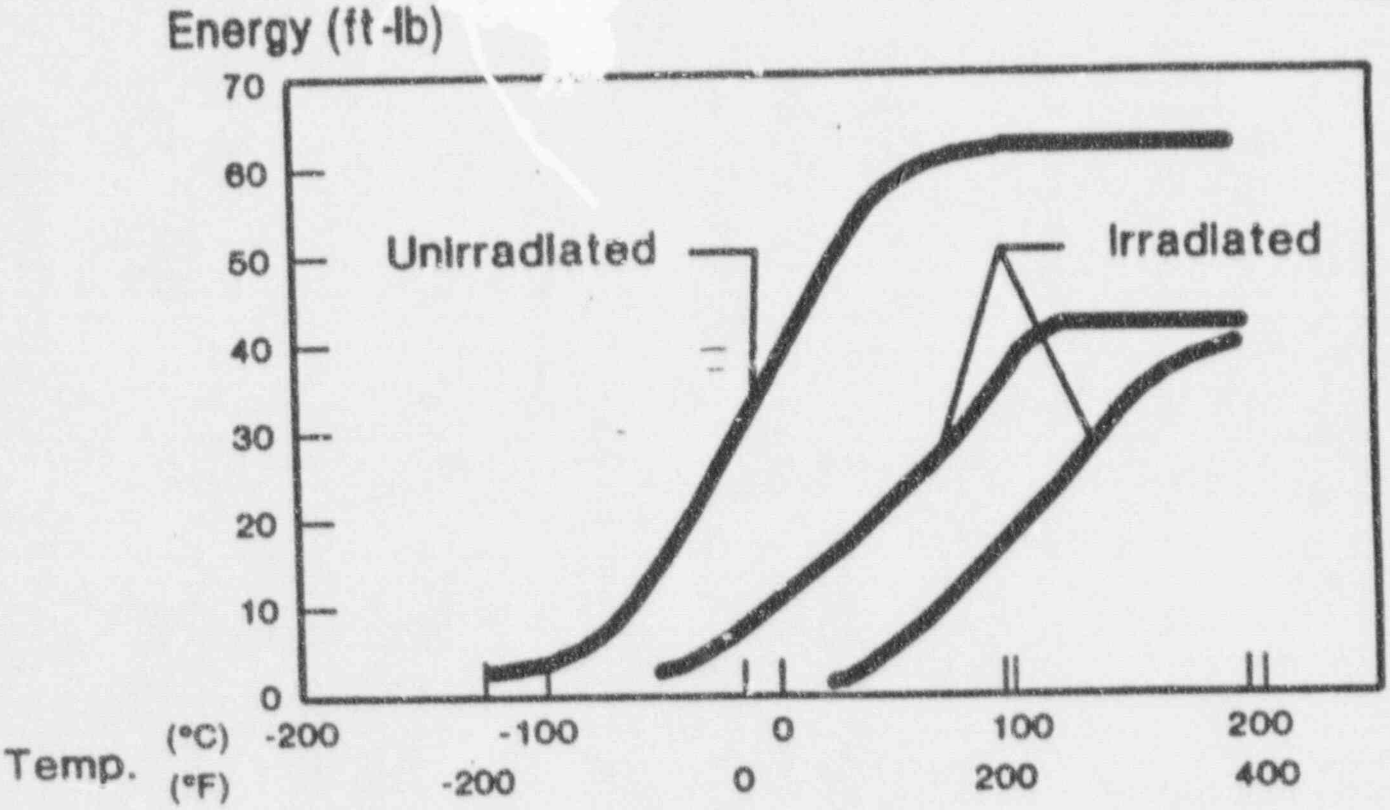

Figure 2-1. Effect of Irradiation on the Charpy V-Notch Toughness (Sample)

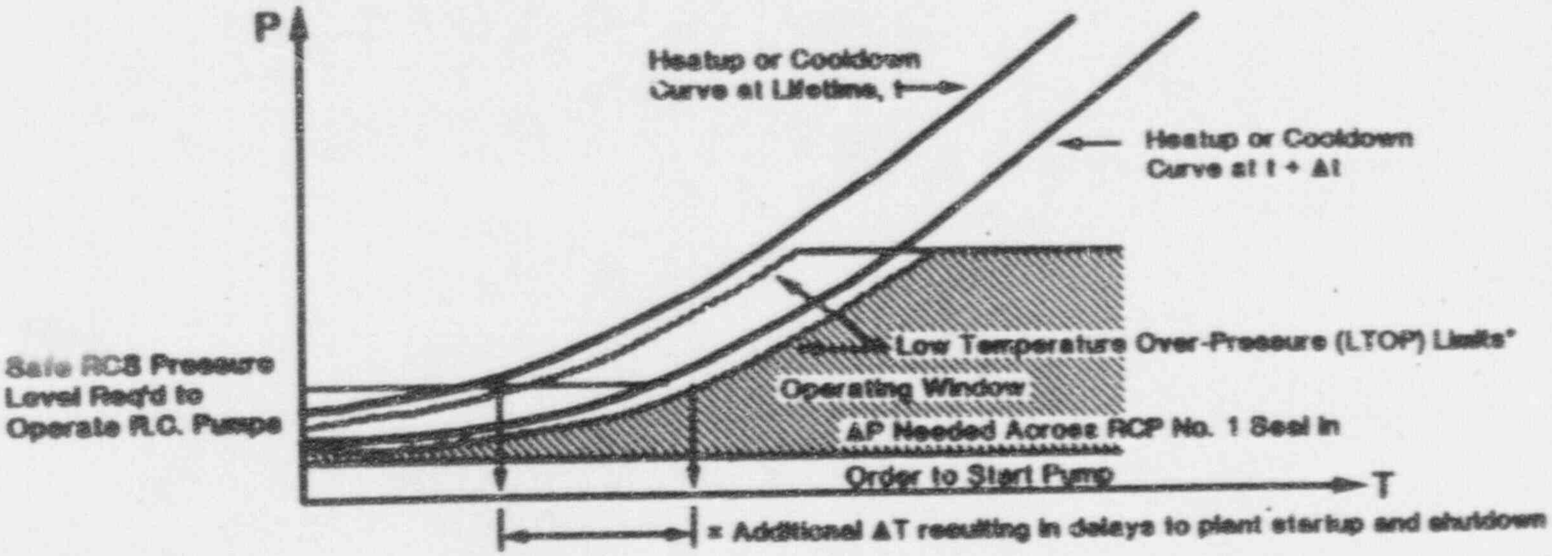

Figure 2-2. Pressure-Temperature Heatup and Cooldown Operational Limit Curves (sample)

3.0 IRRADIATION EMBRITTLEMENT PREDICTIONS

Neutron irradiation has been shown to produce embrittlement which reduces the toughness properties of reactor vessel steels. The decrease in the toughness properties can be assessed by determining the increase to higher temperatures of the reactor vessel material reference nil-ductility transition temperature (RT_{NDT}). Because the chemistry (especially copper and nickel content) of reactor vessel steel has been identified as a major contributor to radiation embrittlement, methods have been developed to relate the magnitude of the increase in RT_{NDT} to the amount of neutron fluence. Based on the initial RT_{NDT} value and the material chemistry of the reactor vessel limiting core region materials, the post irradiation RT_{NDT} values are determined.

Westinghouse, other NSSS vendors, the U.S. Nuclear Regulatory Commission and others have developed trend curves and methods for predicting adjustment of RT_{NDT} as a function of neutron fluence and copper, and nickel content. The two prediction methods of most importance to the Beaver Valley Unit 1 reactor vessel are the methods used in 1) the Pressurized Thermal Shock Rule [1] and 2) Regulatory Guide 1.99 Revision 2 [2]. Currently the method identified in the PTS rule is required to be used for the evaluation of reactor vessels against the prescribed PTS screening criteria. However, a more recent method has been developed and is identified in Regulatory Guide 1.99 Revision 2. This method is expected at some time in the future to be required to be used in the PTS Rule. Therefore, both of these prediction methods have been considered in the neutron flux reduction evaluation.

3.1 Pressurized Thermal Shock Methodology

In the PTS Rule, the NRC Staff has selected a conservative and uniform method for determining plant-specific values of reference temperature for PTS (RT_{PTS}) at a given time.

The prescribed equations in the PTS rule for calculating RT_{PTS} are actually one of several ways to determine RT_{NDT}. For the purpose of comparison with the screening criteria, which are discussed later, the value of RT_{PTS} for

the reactor vessel must be calculated for each weld and plate, or forging in the beltline region as given below. For each material, RT_{PTS} is the lower of the results given by Equation 1 and 2.

Equation 1:

$$RT_{PTS} = I + M + [-10 + 470(Cu) + 350(Cu)(Ni)]\ f^{0.270}$$

Equation 2:

$$RT_{PTS} = I + M + 283\ f^{0.194}$$

where

I = the initial reference transition temperature of the unirradiated material measured as defined in the ASME Section III Code, NB-2331. If a measured value is not available, the following generic mean values must be used: 0°F for welds made with Linde 80 flux, and -56°F for welds made with Linde 0091, 1092 and 124 and ARCOS B-5 weld fluxes.

M = the margin to be added to cover uncertainties in the values of initial RT_{NDT}, copper and nickel content, fluence, and calculation procedures. In Equation 1, M=48°F if a measured value of I was used, and M=59°F if the generic mean value of I was used. In Equation 2, M=0°F if a measured value of I was used, and M=34°F if the generic mean value of I was used.

Cu and Ni = the best estimate weight percent of copper and nickel in the material.

f = the maximum neutron fluence, in units of $10^{19} n/cm^2$ (Energies greater than or equal to 1 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest neutron fluence for the period of service in question.

Note that the chemistry values given in equation 1 and 2 are best estimate mean values. The margin, M, produces upper bound RT_{PTS} predictions. Thus, the mean material chemistry values are to be used when available so as not to compound conservatism.

3.2 Regulatory Guide 1.99 Revision 2

The Nuclear Regulatory Commission (NRC) has developed a method for predicting radiation embrittlement of reactor vessel material which is published in Regulatory Guide 1.99. Regulatory Guide 1.99 was originally published in July 1975 with a Revision 1 being issued in April 1977 and a current Revision 2 [2] issued in May 1988. The Adjusted Reference Temperature (ART), based on the methods of Regulatory Guide 1.99 Revision 2, can be compactly described by the sequence of equations listed below:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

$$\Delta RT_{NDT} = [CF]f^{(.28 - 0.10 \text{ Log } f)} \text{ where}$$

f = Neutron fluence, n/cm^2 (E 1 MeV), divided by 10^{19}

CF = Chemistry factor from tables for welds and for base metal (plates and forgings) (if no data use 0.35% Cu and 1.0% Ni)

The neutron fluence at any depth in the vessel wall is determined as follows:

$$f = f_{surf.} (e^{-0.24X})$$

x = depth into vessel wall from inner (wetted) surface

$$\text{Margin} = 2 [\sigma_I^2 + \sigma_\Delta^2]^{0.5}$$

where

σ_I = standard deviation for the initial RT_{NDT}. If the initial RT_{NDT} is measured, σ_I is considered to be 0°F. If initial RT_{NDT} is not measured, σ_I is taken to be 17°F for welds.

σ_Δ = standard deviation of ΔRT_{NDT}; 28°F for welds and 17°F for base metal except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}.

4.0 PLANT SPECIFIC EVALUATION

In the assessment of reactor vessel material conditions in accordance with the PTS rule [1] and Regulatory Guide 1.99, Revision 2 [2], an area of uncertainty is establishing the best estimate chemical content of critical reactor vessel welds, particularly the copper content. Best estimate values of copper and nickel content are needed for use in the equations to project RT_{NDT} values for comparison with the screening criteria for PTS and also to develop heatup and cooldown curves.

Figure 4-1 identifies and indicates the location of all beltline region materials for the Beaver Valley Unit 1 reactor vessel. The materials of importance for the beltline region are the intermediate and lower shell plates and the associated longitudinal and circumferential welds.

4.1 Pressurized Thermal Shock Evaluation

Pressurized thermal shock evaluations were performed using (1) PTS rule and (2) Regulatory Guide 1.99, Revision 2. These calculations were carried out for the entire beltline region materials for the Beaver Valley Unit 1 Reactor Vessel. These results are reported in Table 4-1. Also, plots were developed, RT_{PTS} or RT_{NDT} Vs. fluence and are shown in figures 4-2 and 4-3. These figures are applicable as long as i) the material chemistry of the beltline region remain unchanged, ii) the flux remain unchanged.

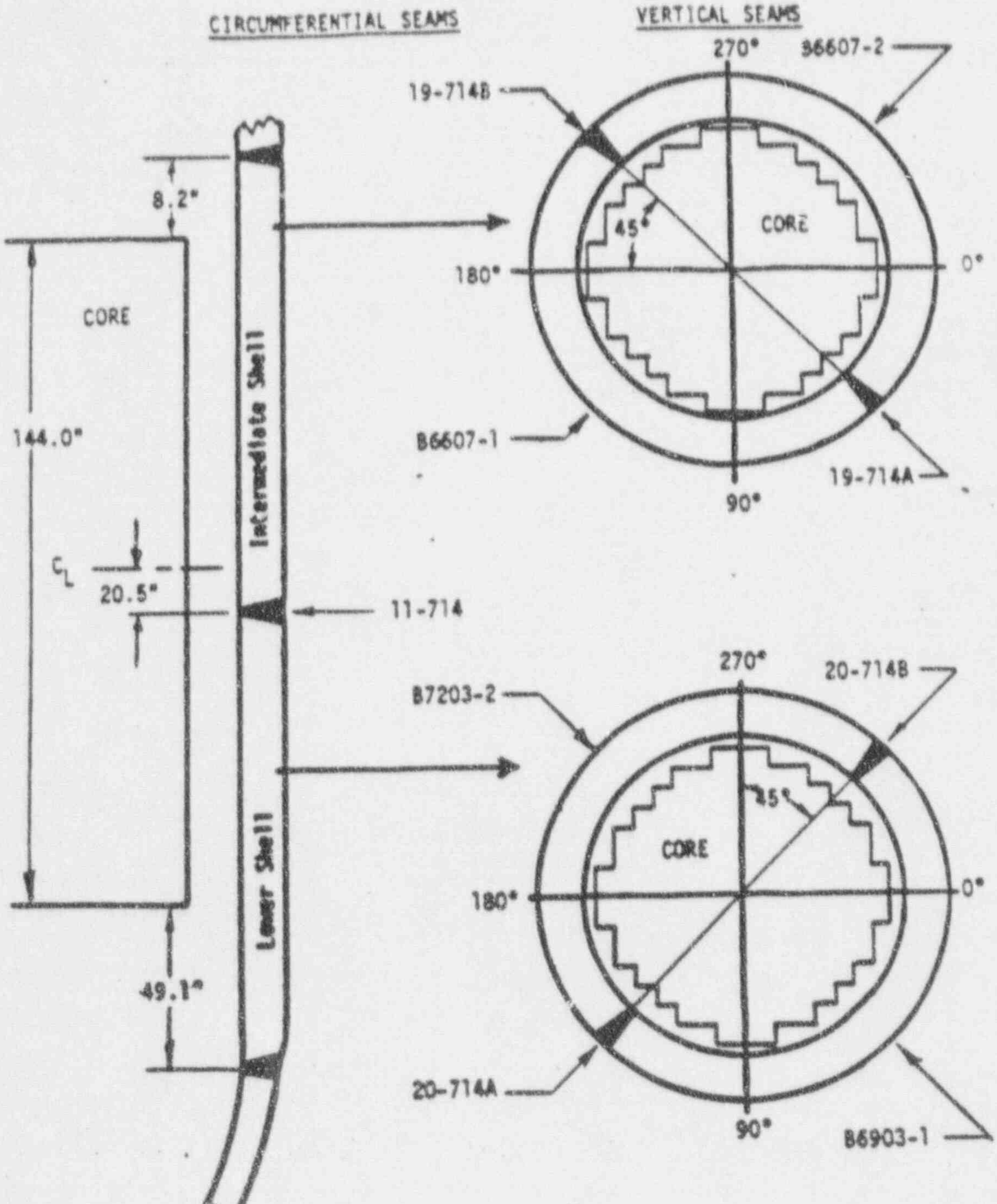

Figure 4-1. Identification and Location of Beltline Region Material for the Beaver Valley Unit 1 Reactor Vessel

TABLE 4-1

PROJECTED RT_{PTS} FOR THE BEAVER VALLEY UNIT 1

REACTOR VESSEL BELTLINE MATERIALS

Component	EFPY	Projected Fluence (1,2)	RT_{PTS} (PTS Rule) (°F)	RT_{PTS} (Reg. Guide 1.99, Rev. 2) (°F)	PTS Screening Criteria
Int. Plt, B6607-1	32	4.07	217	214	270
Int. Plt, B6607-2	32	4.07	247	244	270
Lower Plt, B6903-1	32	4.07	253	254	270
Lower Plt, B7203-2	32	4.07	190	188	270
Long. Weld, 305424	32	0.75	173	186	270
Long. Weld, 305414	32	0.75	209	203	270
Circ. Weld, 90136	32	4.07	198	190	300
Int. Plt, B6607-1	48	6.11	231	222	270
Int. Plt, B6607-2	48	6.11	261	252	270
Lower Plt, B6903-1	48	6.11	274	265	270
Lower Plt, B7203-2	48	6.11	204	196	270
Long. Weld, 305424	48	1.13	192	208	270
Long. Weld, 305414	48	1.13	233	227	270
Circ. Weld, 90136	48	6.11	220	201	300

(1) Fluence is in 10^{19} n/cm^2.

(2) Projected fluence from Reference [5].

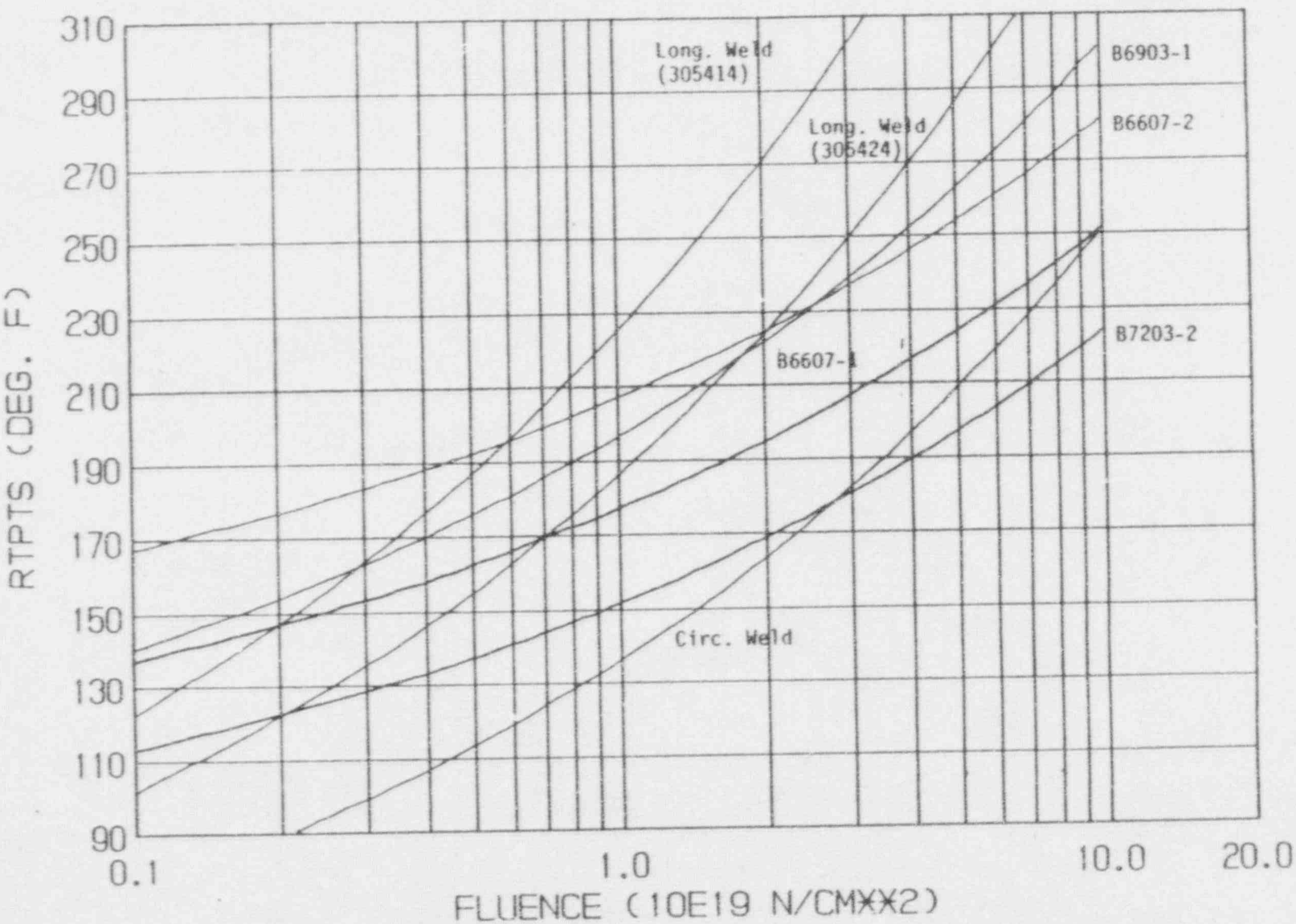

Figure 4-2. Fluence Vs. RT_{PTS} for Beltline Region Materials (PTS Rule)

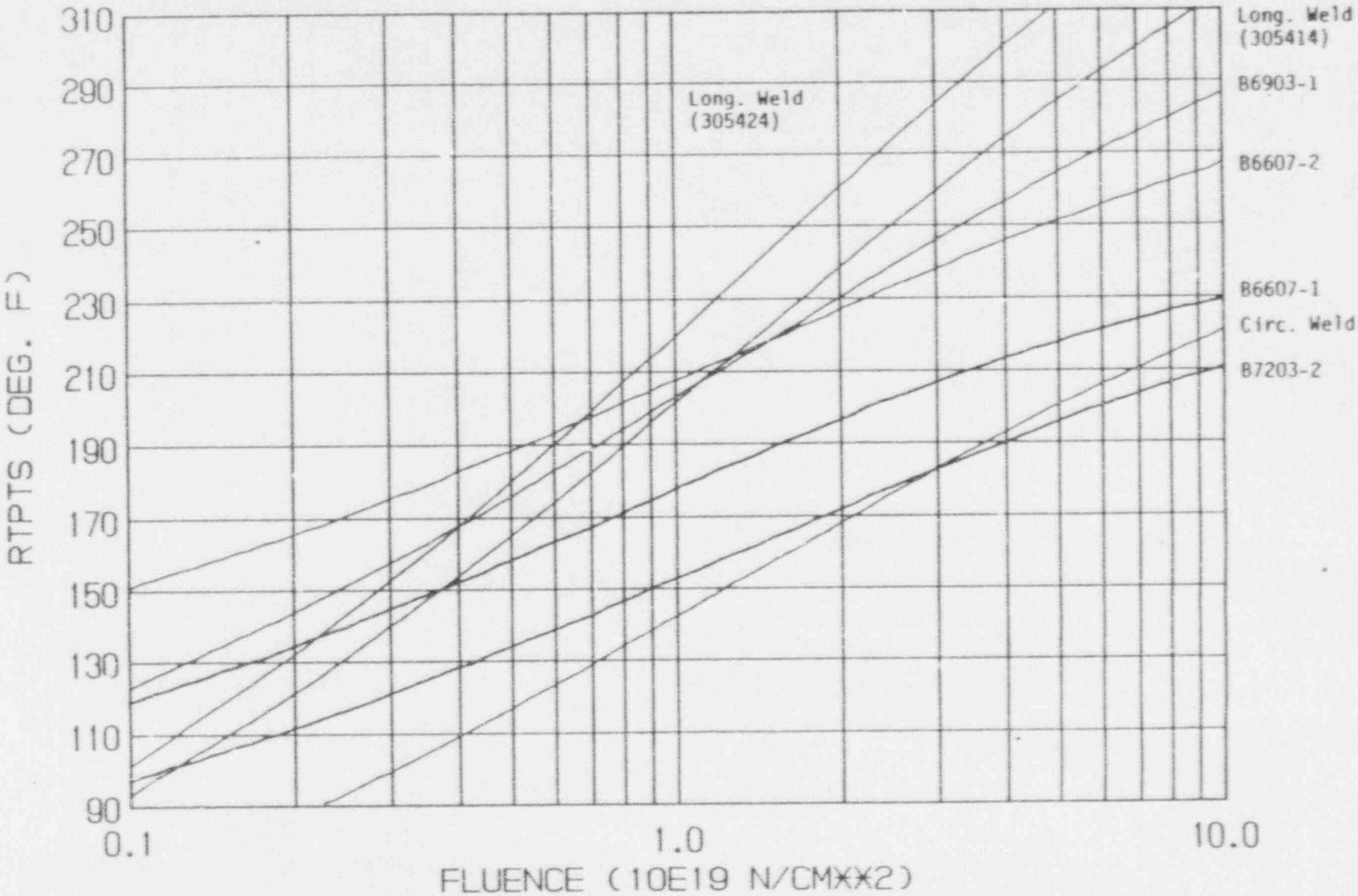

Figure 4-3. Fluence Vs. RT_{PTS} for Beltline Region Materials (Reg. Guide 1.99, Rev. 2)

4.2 Heatup and Cooldown Curves

Heatup and cooldown curves are developed for the Beaver Valley Unit 1 vessel based on the most limiting beltline region material. RT_{NDT} calculations at 1/4T (T being the thickness of vessel at beltline region) and 3/4T locations were done for all the beltline regions. It was found that the lower plate B6903-1 is the most limiting material for the development of heatup and cooldown curves. RT_{NDT} values at 1/4T and 3/4T locations for 16, 24, 32 and 48EFPYS are provided in Table 4-2.

Figures 4-4 and 4-5 show the present heatup/cooldown curves which are applicable for 9.5 EFPY. Westinghouse generated heatup/cooldown curves for 16, 24, 32 and 48 EFPY using Reg. Guide 1.99, Rev. 2. The inner surface peak fluences projected for 16, 24, 32 and 48 EFPYs are $2.112 \times 10^{19} n/cm^2$, $3.168 \times 10^{19} n/cm^2$, and $4.07 \times 10^{19} n/cm^2$ and $6.105 \times 10^{19} n/cm^2$ respectively [5]. The heatup and cooldown curves for 16, 24, 32 and 48 EFPY are shown in figures 4-6 through 4-13.

TABLE 4-2

RT_{NDT} VALUES AT 1/4T AND 3/4T LOCATIONS FOR 16,24, 32, AND 48 EFPY
FOR THE LIMITING PLATE USING REG. GUIDE 1.99, REV. 2

16 EFPY		24 EFPY		32 EFPY		48 EFPY	
1/4T (°F)	3/4T	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)
215*	188*	229**	199*	238**	206*	252**	218*
(224)	(179)	(243)	(198)	(254)	(210)	(270)	(229)

Projected neutron fluence (E > 1 MeV) at inner surface are 2.112×10^{19} n/cm^2, 3.168×10^{19} n/cm^2, 4.07×10^{19} n/cm^2 and 6.11×10^{19} n/cm^2 for 16, 24, 32, and 48 EFPYs, respectively [5].

Number in ( ) represents RT_{NDT} values using chemistry factors based on surveillance capsule data.

* For intermediate plate, B6607-2
** For lower plate, B6903-1

TABLE 4-3

RT_{NDT} VALUES USED AT 1/4T AND 3/4T LOCATIONS FOR
DEVELOPMENT OF HEATUP AND COOLDOWN CURVES

16 EFPY		24 EFPY		32 EFPY		48 EFPY	
1/4T (°F)	3/4T	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)	1/4T (°F)	3/4T (°F)
224	188	243	199	254	210	270	229

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B 6607-2

RT_{NDT} AFTER 9.5 EFPY: 1/4T, 202°F

3/4T, 176°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 9.5 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

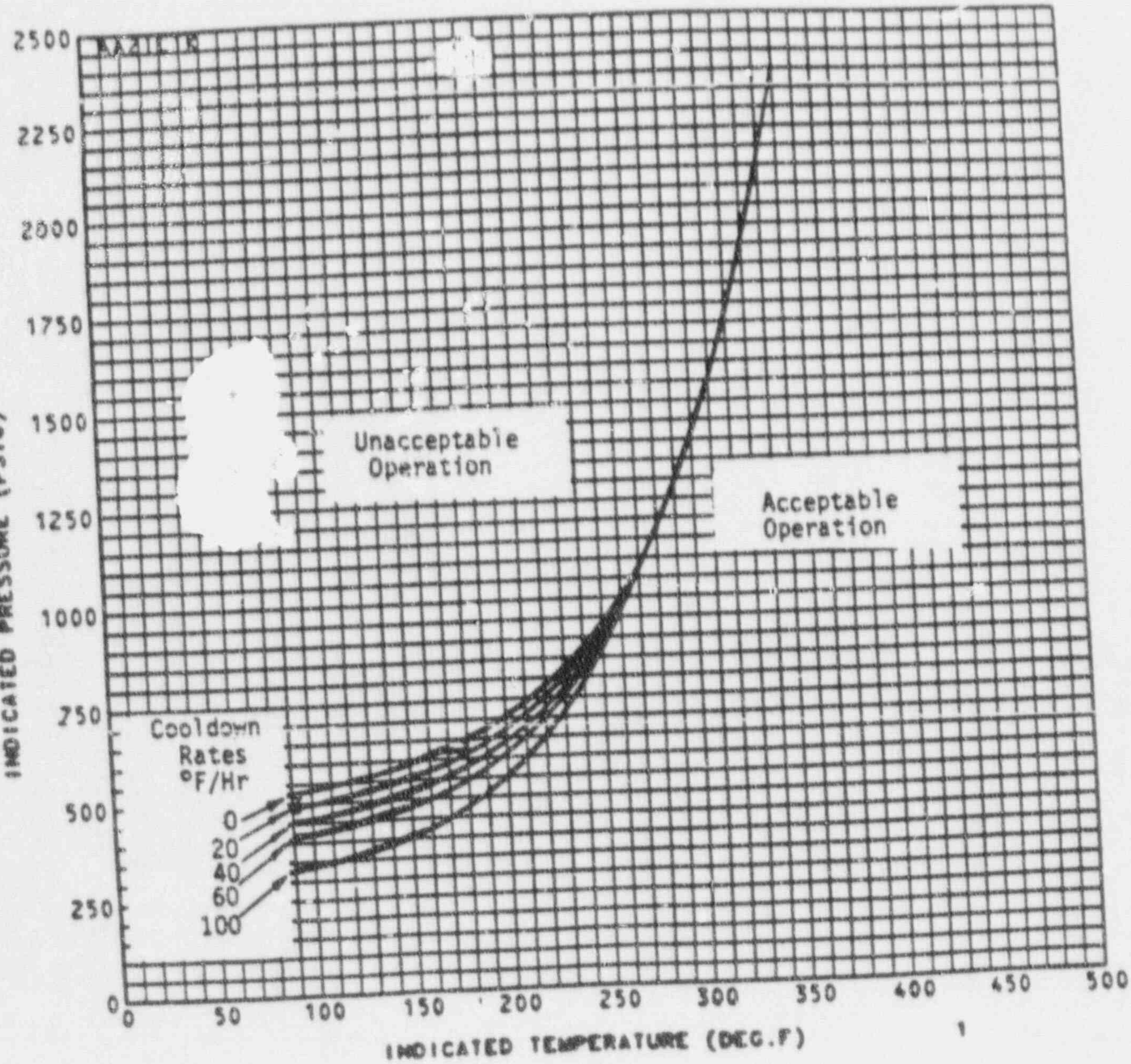

Figure 4-4. Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 9.5 EFPY, using Reg. Guide 1.99, Rev. 2 (Current Curves) (No Margin for Instrument Error)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B6607-2

RT_{NDT} AFTER 9.5 EFPY: 1/4T, 202°F

3/4T, 176°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 9.5 EFPY. DOES NOT CONTAIN MARGIN FOR POSSIBLE INSTRUMENT ERRORS.

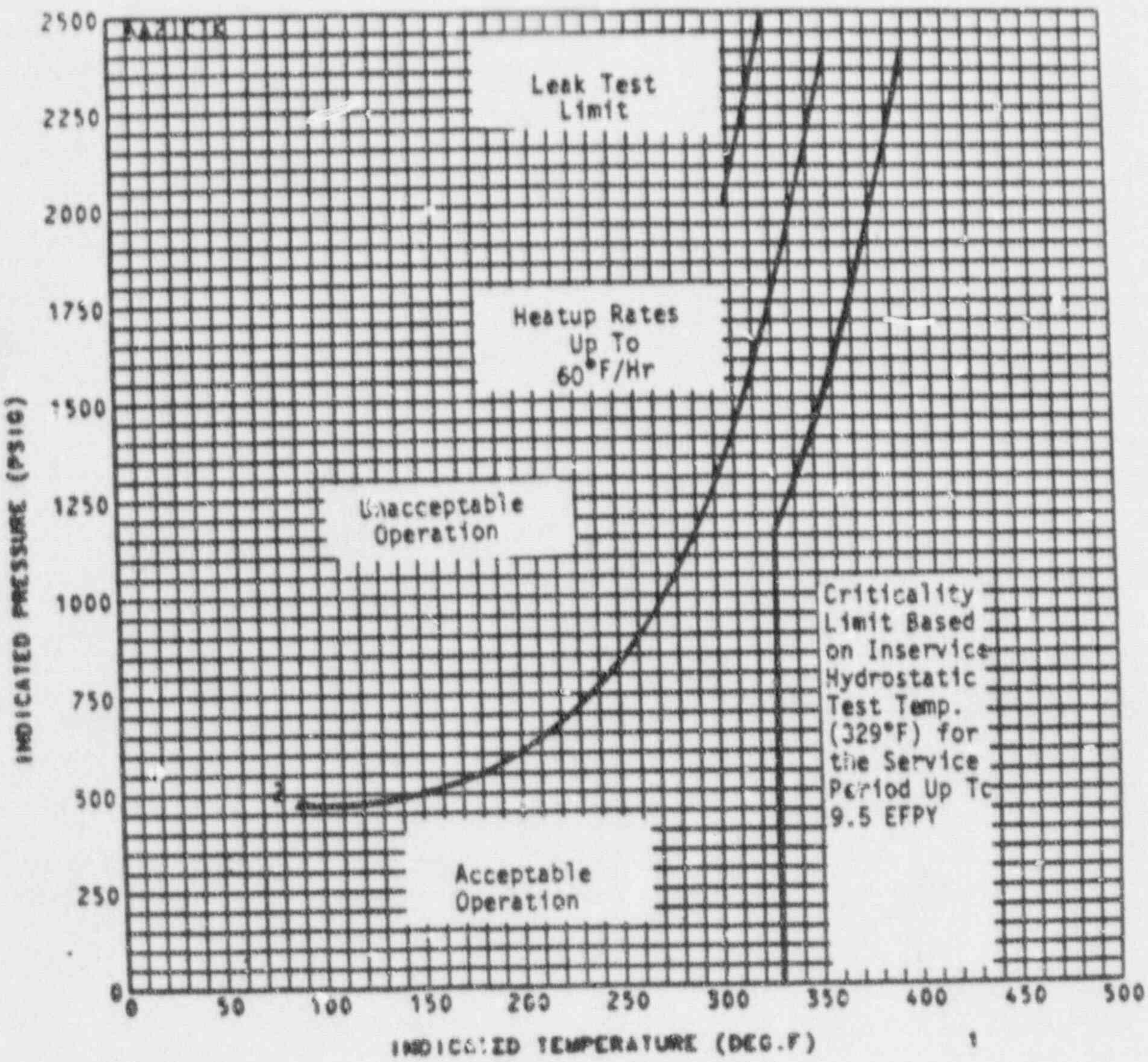

Figure 4-5. Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 9.5 EFPY Using Reg. Guide 1.99, Rev. 2 (Current Curves) (No Margin for Instrument Error)

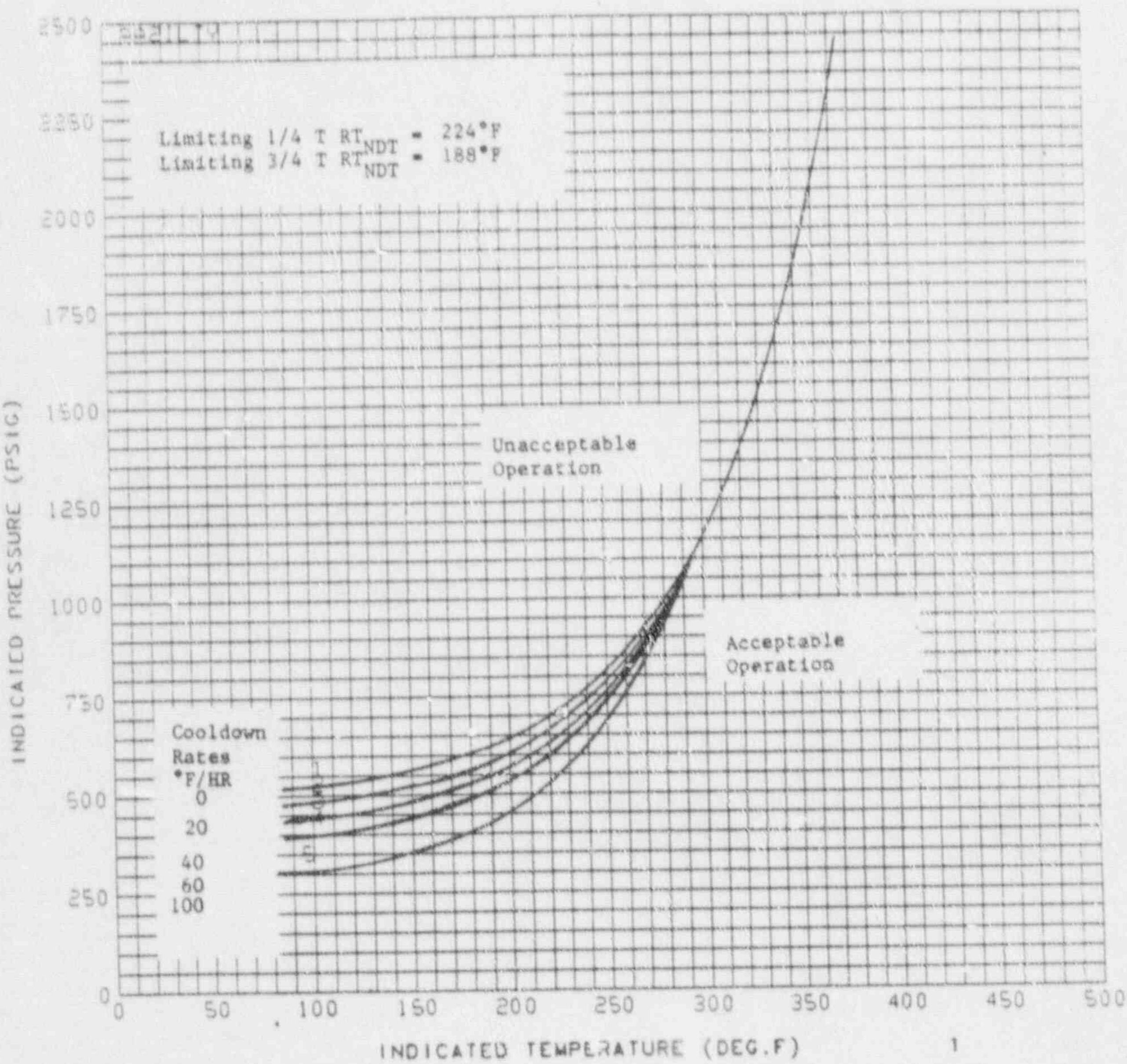

Figure 4-6. Beaver Valley Unit 1 Reactor Vessel Cooldown Limitations Applicable for the First 16 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

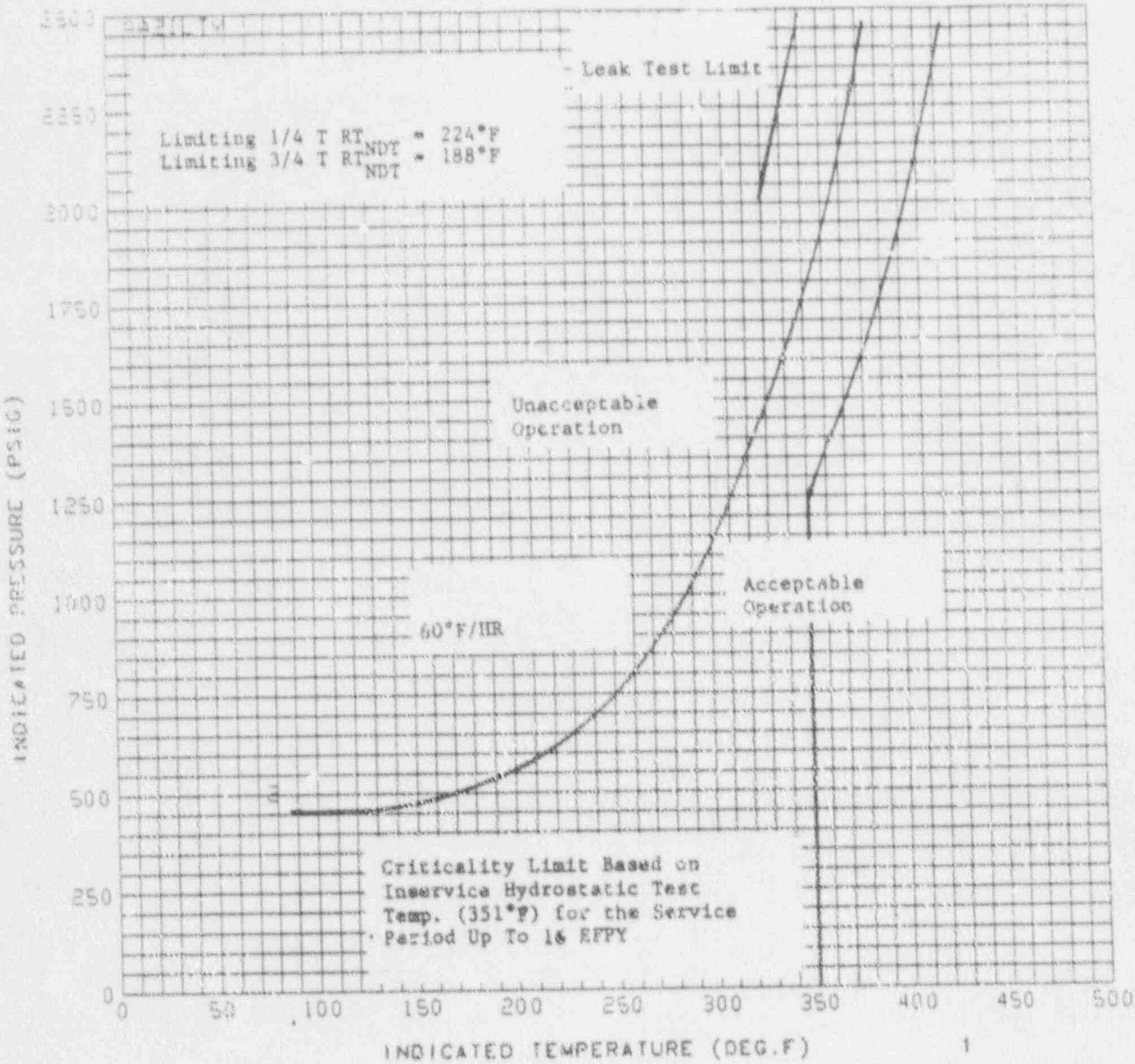

Figure 4-7. Beaver Valley Unit 1 Reactor Vessel Heatup Limitations Applicable for the First 16 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

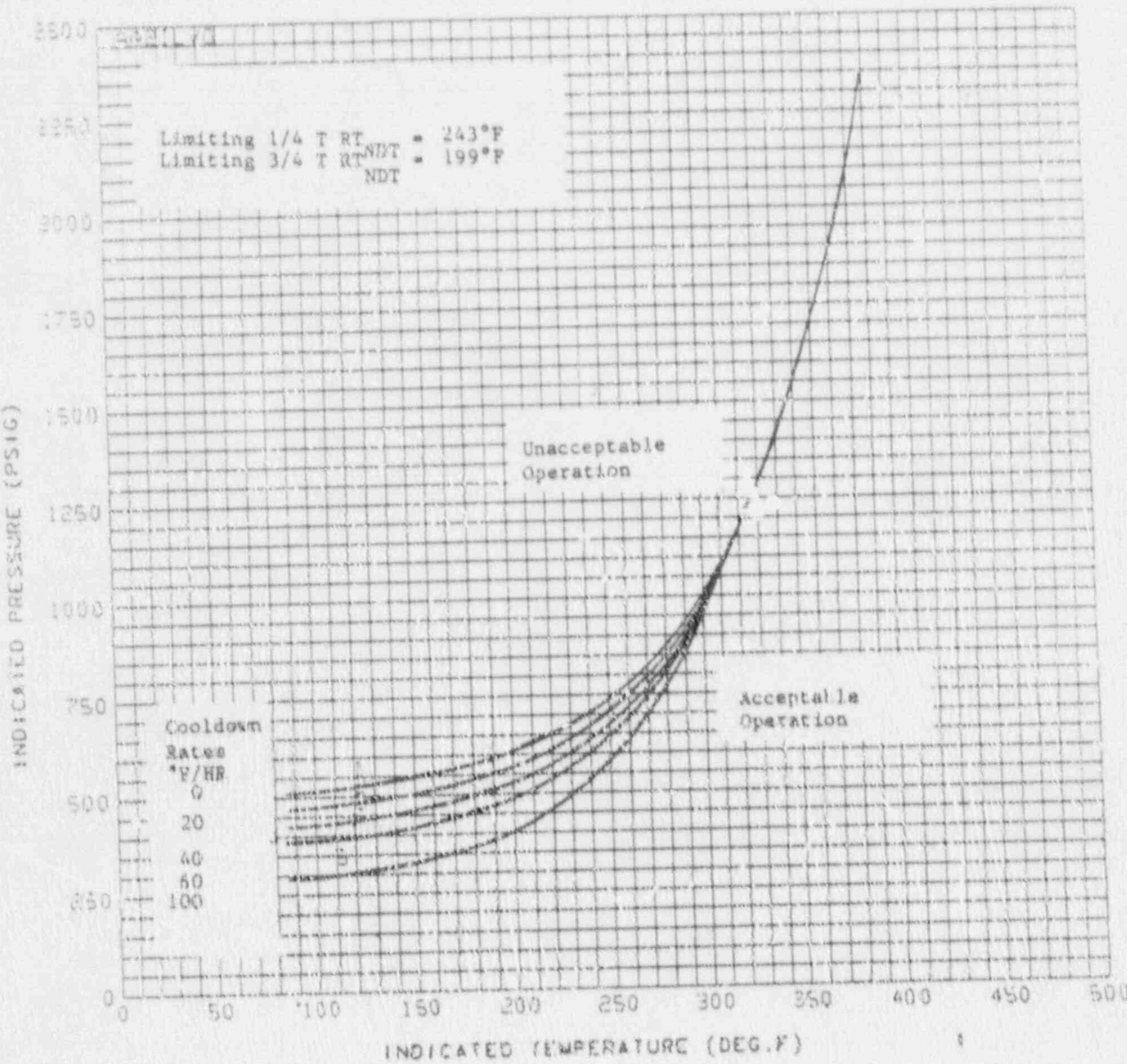

Figure 4-8. Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for The First 24 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

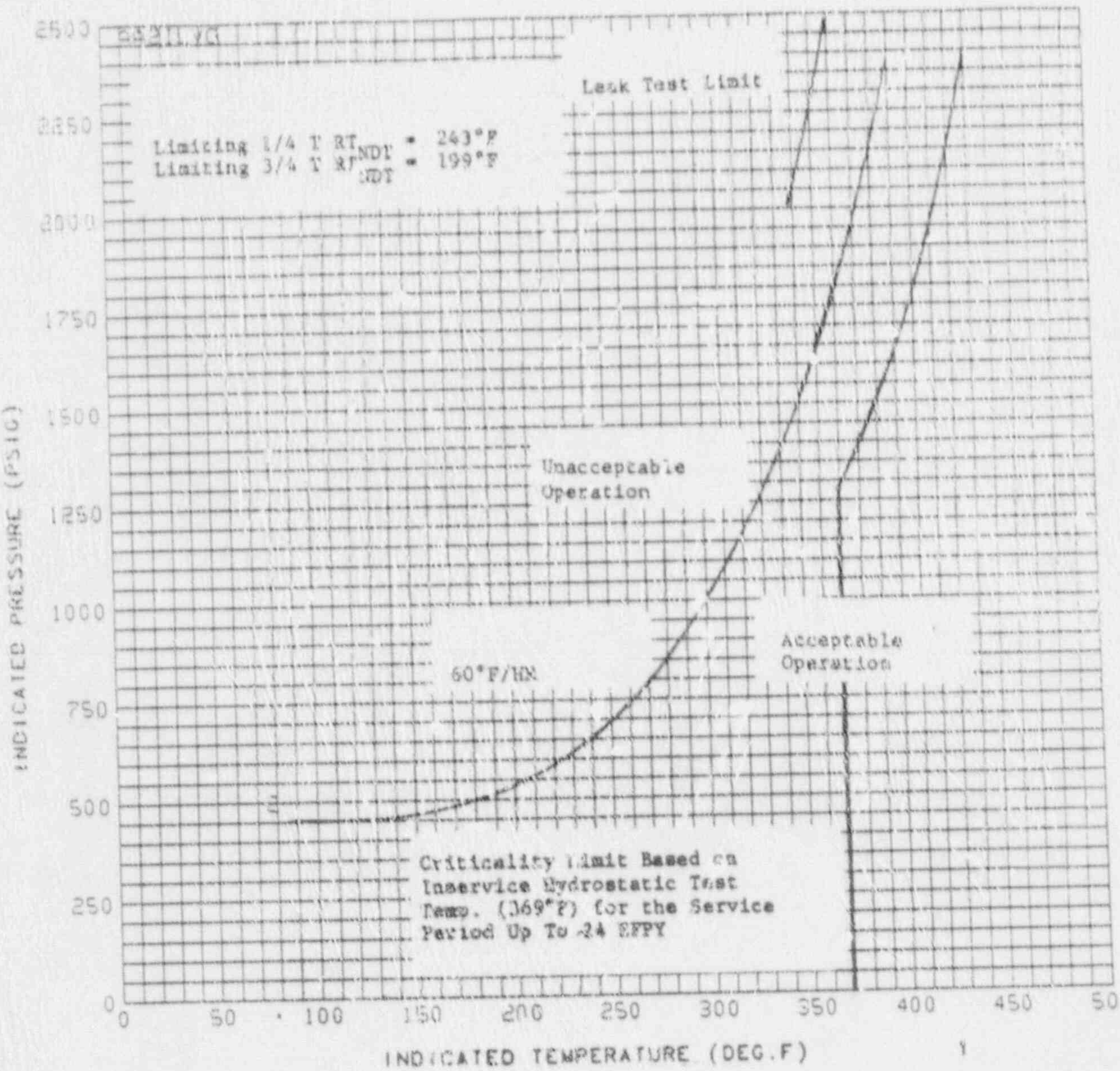

Figure 4-9. Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 24 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

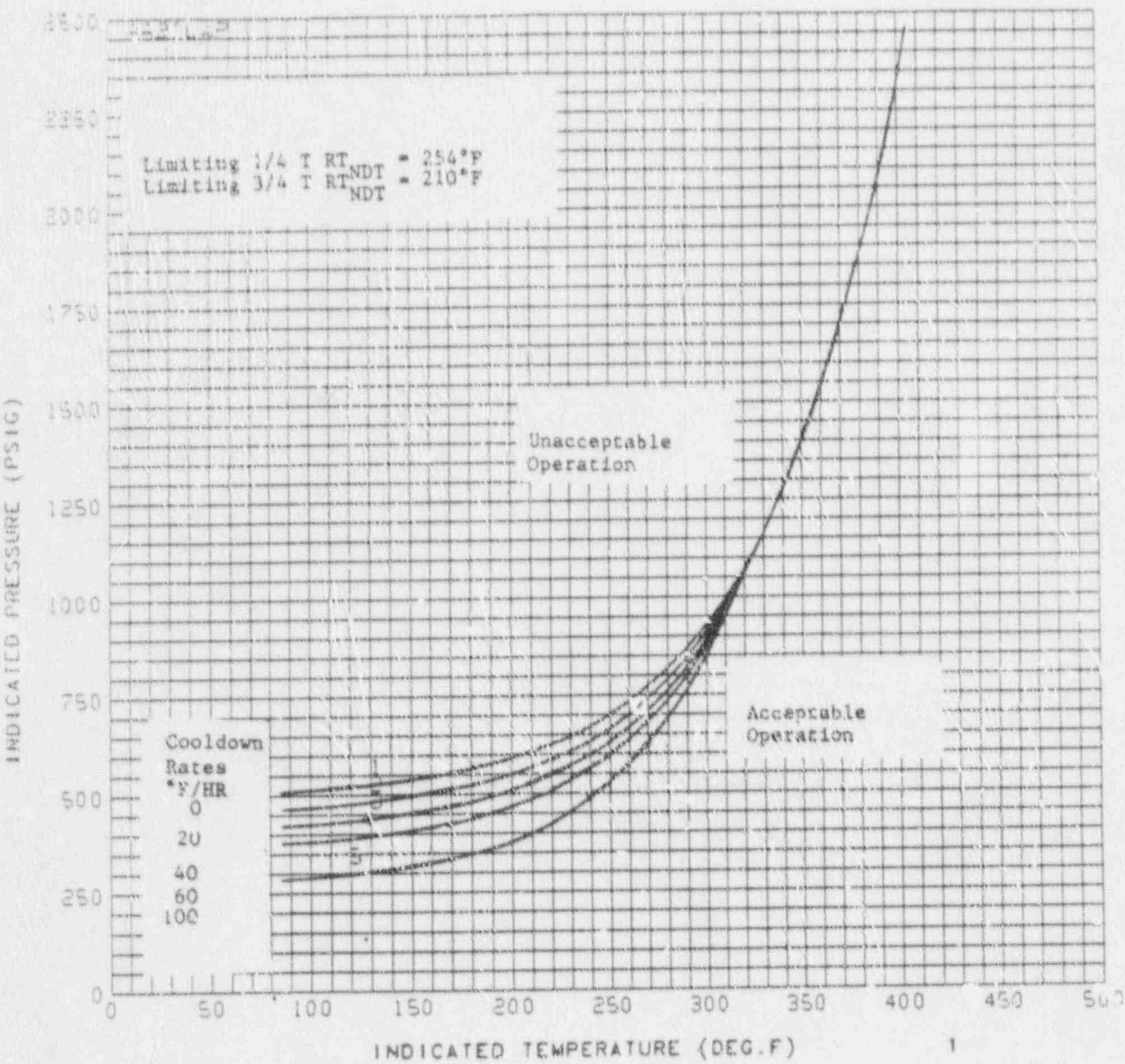

Figure 4-10. Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 32 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

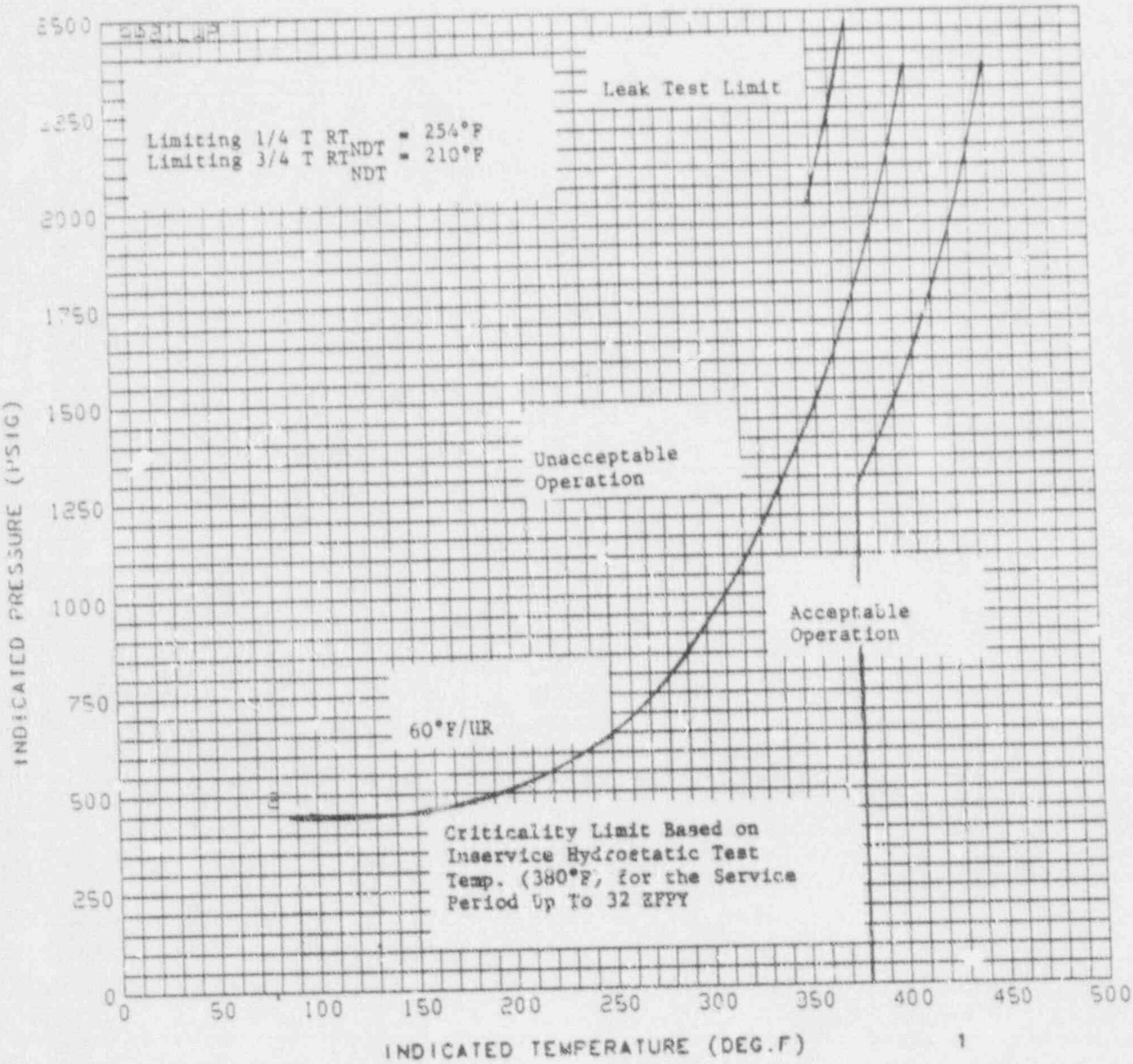

Figure 4-11. Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 32 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

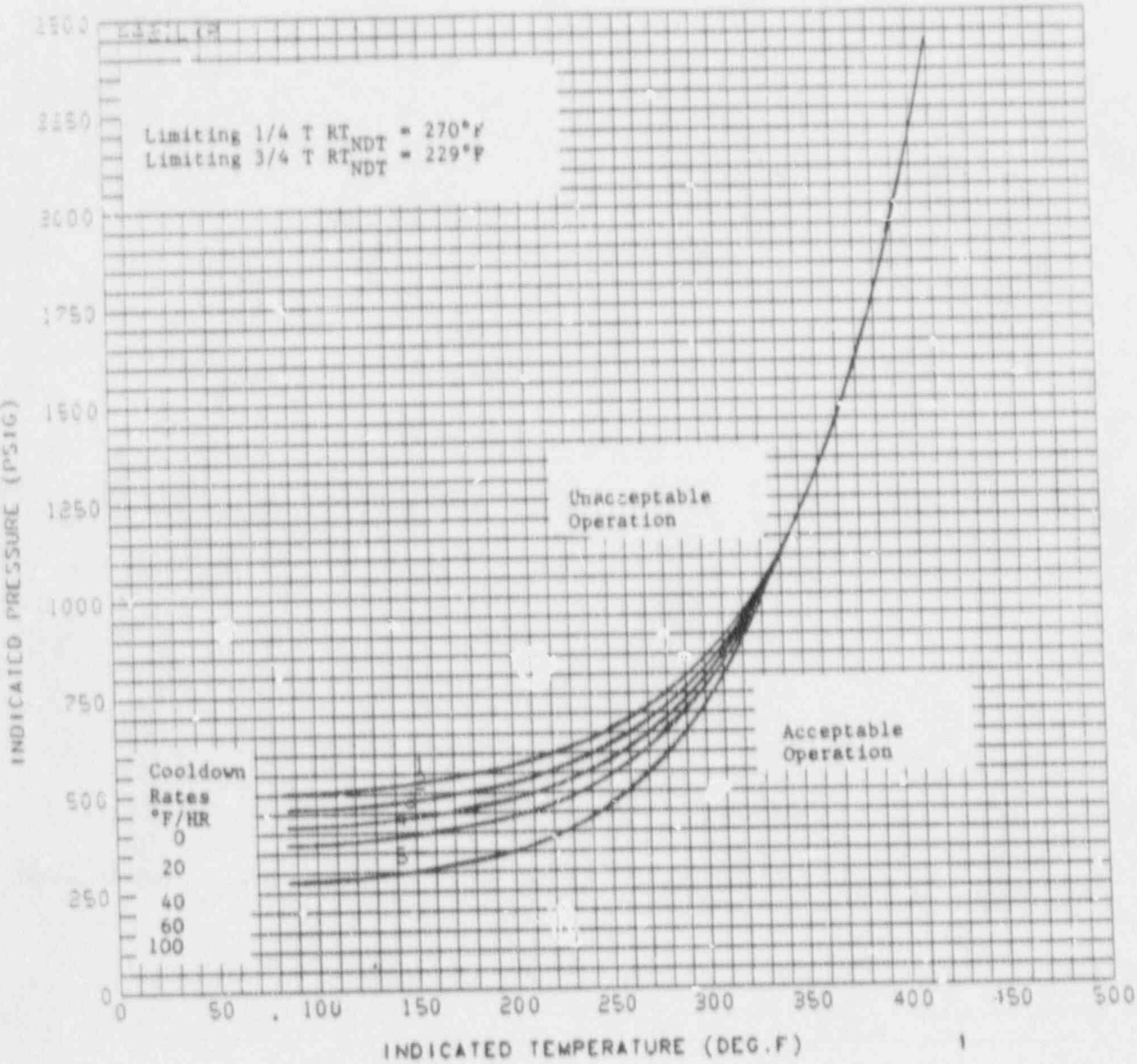

Figure 4-12. Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable For The First 48 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

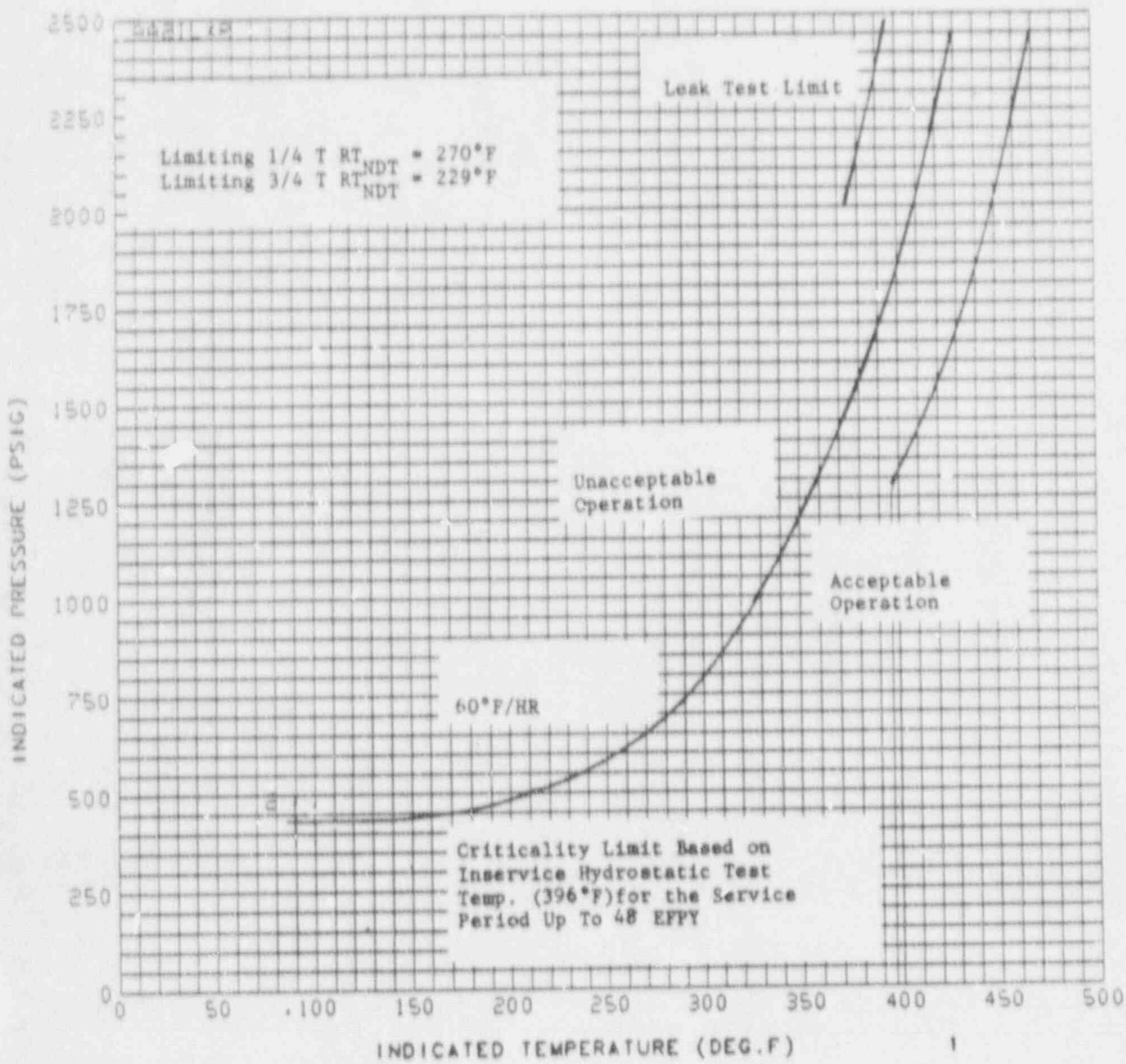

Figure 4-13. Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable For The First 48 EFPY Using Reg. Guide 1.99, Rev. 2. (No margin for instrument error)

5.0 FLUX REDUCTION GOAL

Neutron flux reduction goals for the Beaver Valley Unit 1 reactor vessel were established for the licensed life and an additional 20 calendar year life considered for life extension period for the key issues related to neutron embrittlement. The issues considered in this report are i) Pressurized thermal shock, ii) Operating limits and iii) Emergency response guidelines. Flux reduction goals were established based on these three issues.

5.1 Fluence Limits

In order to set neutron flux reduction goals, a target end-of-life neutron fluence must first be established. In setting the target end-of-life fluence, a variety of key issues were taken into consideration to ensure setting the correct-target.

Neutron fluence limits were established for both end-of-design life and life extension. The end-of-design life was assumed to be 32 effective full power years (EFPY). The life extension was assumed to be 48 EFPY.

5.1.1 Pressurized Thermal Shock Criteria

The consideration in setting the neutron flux reduction goals was pressurized thermal shock. PTS was assessed using the current PTS rule and the Regulatory Guide 1.99 Revision 2 methodology. Emergency Response Guideline limits [6] that have been established for operator guidance during PTS events were also evaluated.

PTS Rule Methodology

The neutron fluence at which the PTS screening criteria are reached was calculated by solving both Equations 1 and 2, as described in Section 3.1, for f.

that is,

$$f_1 = [\frac{RT_{PTS} - I - M}{-10 + 470\ Cu + 350\ Cu\ Ni}]^{1/0.27}$$

$$f_2 = [\frac{RT_{PTS} - I - M}{283}]^{1/0.194}$$

where RT_{PTS} is defined to be the appropriate screening criterion. The screen- ing criterion as specified by the PTS rule, is 270°F for longitudinal welds and base material. For circumferential welds, the screening criterion is 300°F.

By using the material properties for the beltline region and the appropriate margin terms, two neutron fluences were calculated. The lowest value of the two fluences is the applicable fluence. From Table 4-1, it can be seen that the limiting material for the PTS, is the lower plate B6903-1. All beltline region materials are within the PTS Screening criteria (See Table 4-1) except the lower plate B6903-1. Flux reduction is required only for the plate B6903-1 to maintain RT_{PTS} values within the screening criteria. The above methodology is used to develop flux reduction factor required for B6903-1 and is shown in figure 5-1.

Regulatory Guide 1.99 Revision 2 Methodology

Regulatory Guide 1.99 Revision 2 presents the latest method that has been developed for predicting radiation embrittlement which may in the future be required to be used in the PTS rule. To account for this possibility, target neutron fluences at which the current PTS rule screening criteria will be reached were established in a similar fashion to the PTS Rule Methodology of the previous section.

For this particular case, the limiting PTS material is the lower plate, B6903-1. For the purpose of PTS evaluation, surveillance capsule data are not used. And, since surveillance capsule data is not used, the σ_Δ term is not cut in half.

$$ART = \text{Initial } RT_{NDT} + \text{Margin} + CF \times \text{Fluence Factor}$$

Rearranging those equations,

$$\text{fluence factor} = f^{(.28 - .10 \log f)} = \frac{ART - \text{initial } RT_{NDT} - \text{Margin}}{CF}$$

By using the appropriate chemistry, margins, initial RT_{NDT}, and the PTS screening criteria values, a fluence factor can be determined. The neutron fluence may then be determined from a plot of the fluence expression, which is presented in the Regulatory Guide 1.99, Rev. 2. From Table 4-1, it can be seen, that RT_{PTS} values for all the beltline region materials are within PTS Screening Criteria using Regulatory Guide 1.99, Revision 2. Hence, there is no need to develop any flux reduction factor curve using Regulatory Guide 1.99, Revision 2.

5.1.2 Heatup and Cooldown Curves

Heatup and cooldown curves were generated for Beaver Valley Unit 1 Reactor Vessel for 16, 24, 32 and 48 EFPYS. These operating limits are shown in figures 4-5 through 4-13. Using these data Low Temperature Over Pressurization (LTOP) system set points were developed [7].

Detail discussions were held with Duquesne Light Company personnel to investigate the consequences of these operating limits on the operation of the plant. The LTOP system set points were thoroughly scrutinized and it was agreed upon that the operators will have sufficient margin for the safe startup and shutdown of the plant. No flux reduction is required for the operating limits.

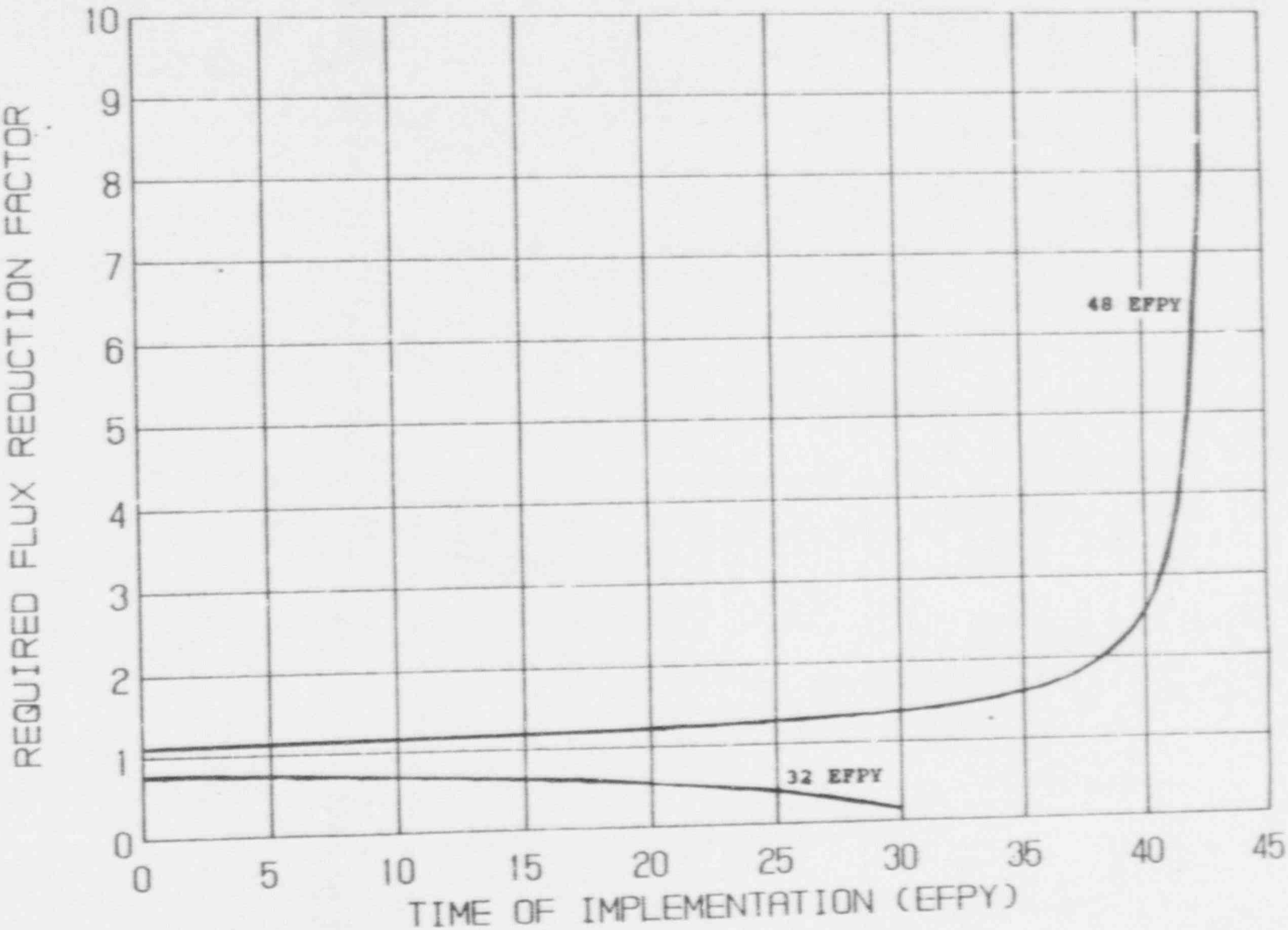

Figure 5-1. Required Flux Reduction Factor Using PTS Rule

Figure 5-2 indicates that at 180°F (48EFPY), the LTOP set pressure is 470 psig, based on a steady state heatup and cooldown allowable pressure of 548 psig (See figure 4-12). To maintain the Reactor Coolant Pump (RCP) Seal, the Reactor Coolant System (RCS) pressure must be maintained at above 325 psig before starting the RCP (for Beaver Valley Unit 1). Assuming an administrative margin of 50 psig, the LTOP system must be set at a Reactor Coolant System pressure of 375 psig for protecting the RCP seal.

Another point of interest is, how long the vessel needs to be radiated, so that the LTOP set point drops to 375 psig at 180°F. It has been calculated that the steady state heatup and cooldown allowable pressure at 180°F will be 455 psig to have a corresponding LTOP set point of 375 psig.

It has been observed that the Beaver Valley Unit 1 reactor vessel needs to be radiated for a long period (beyond 48 EFPY) to have a steady state heatup cooldown curves of allowable pressure of 455 psig at 180°F. This is because i) the radiation damage becomes saturated beyond certain accumulated fluence (see Trend Curve of Ref. 2) and ii) the reference fracture toughness, KIR, see Ref. 4) does not change significantly beyond certain RT_{NDT} level.

5.1.3 Emergency Response Guideline Limits

Emergency Response Guideline (ERG) pressure/temperature limits [6] were developed in order to establish guidance for operator action in the event of an emergency situation, such as a PTS event. Generic categories of limits were developed for the guidelines based on the limiting inside surface RT_{NDT}. The following table presents the generic categories, which were conservatively generated for the Westinghouse Owners Group so that they would be applicable to all Westinghouse plants:

ERG PRESSURE-TEMPERATURE LIMITS

ERG Pressure-Temperature Limit	Applicable RT_{NDT} Value	Applicable Weld
CATEGORY I	$RT_{NDT} < 200°F$	Longitudinal & Circumferential
CATEGORY II	$200°F < RT_{NDT} < 250°F$	Longitudinal & Circumferential
CATEGORY IIIb	$250°F < RT_{NDT} < 300°F$	Circumferential Only

The highest RT_{NDT} for which the generic category ERG pressure/temperature limits were developed was 250°F for plate, forging and longitudinal weld and 300°F for a circumferential weld. Thus, if the limiting vessel material exceeds 250°F RT_{NDT} for a plate, or 300°F RT_{NDT} for a circumferential weld, plant-specific ERG pressure-temperature limits must be developed. For the Beaver Valley Unit 1 vessel, the limiting material is the lower plate, B6903-1, so the screening criteria for the ERG is 250°F.

RT_{NDT} at the inner surface for all beltline region materials are shown in Table 4-1. It indicates that Beaver Valley Unit 1 Vessel, plate B6903-1, will exceed the screening criteria of the ERG for plates at 32 EFPY. There, it is suggested that a new ERG for Beaver Valley Unit 1 should be developed before the vessel reaches 32 effective full power years.

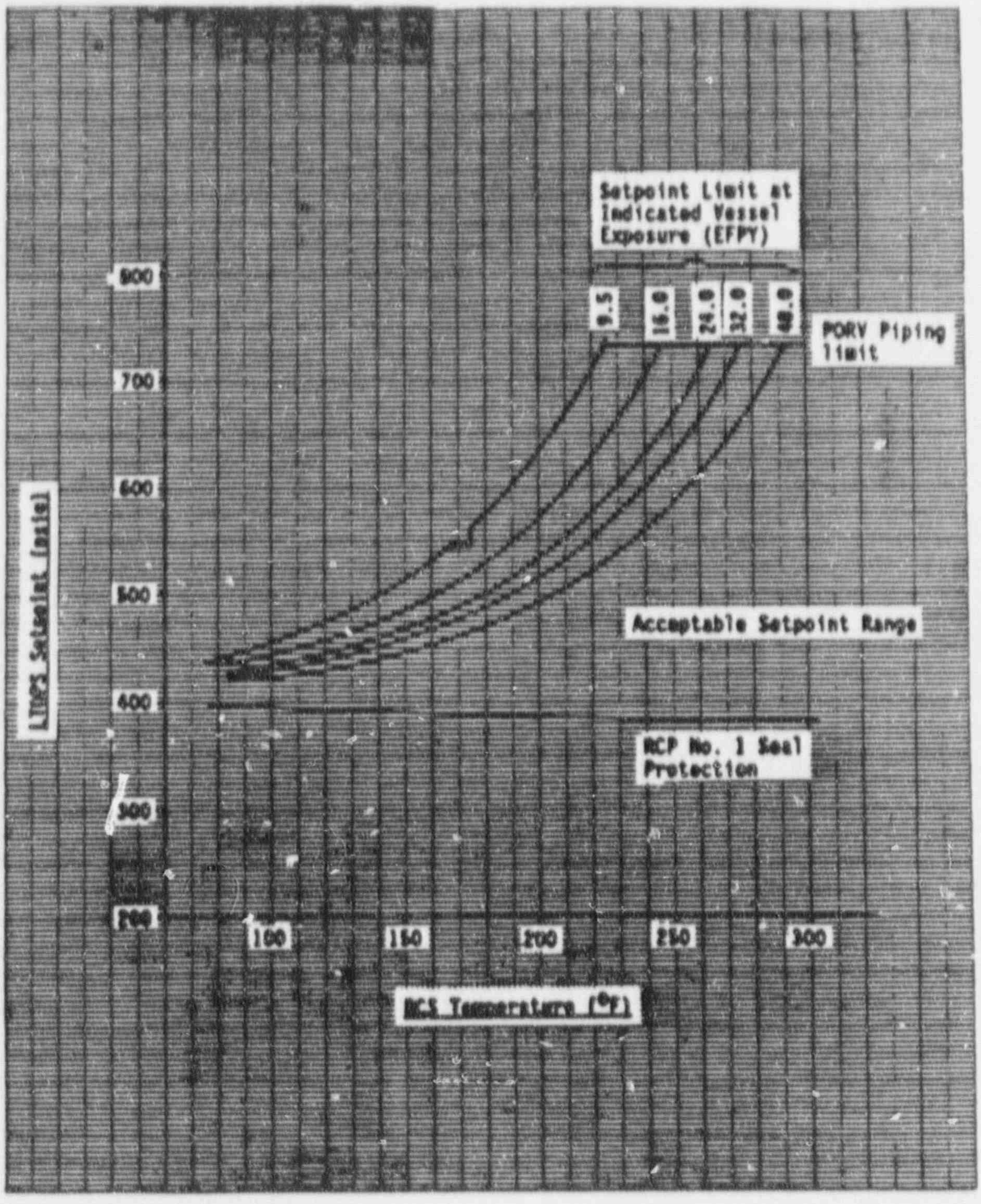

Figure 5-2. Acceptable LTOPS Set Point Range Vs. RCS Temperature

6.0 CONCLUSION

Pressurized Thermal Shock Criteria

From Table 4-1, it can be seen that no flux reduction is needed for vessel life of 32 and 48 EFPYs using Regulatory Guide 1.99, Revision 2. However, flux reduction is needed for a life attainment of 48 EFPY using PTS rule. The flux reduction factor curve is developed and is shown in figure 5-1 using PTS rule. Fuel management options may be considered to reduce the cumulative fluence for the vessel life up to 48 EFPY, to maintain RT_{PTS} values for the Beaver Valley Unit 1, below the PTS screening criteria.

However, it is to be noted that the NRC recently issued a proposed methodology[8] to calculate the RT_{PTS} values. This methodology is essentially the same as the methods used in Regulatory Guide 1.99, Revision 2. This report identifies that the Beaver Valley Unit 1 vessel will not exceed screening criteria using Regulatory Guide 1.99, Revision 2.

Heatup and Cooldown Curves

It has been concluded in section 5.1.2, that the Beaver Valley Unit 1 has acceptable margin of operation up to 48 EFPY provided i) the Reactor Vessel Core maintains the current level or less flux ii) no significant change in material chemistry in the beltline region and iii) no significant change in the regulatory arena.

Emergency Response Guidelines

The limiting material for Beaver Valley Unit 1 Vessel, has been found to be the lower plate B6903-1. The screening criteria for plate is 250°F for Emergency Response Guidelines. From Table 4-1, it can be seen that, the RT_{NDT} at inner surface up to 32 EFPY is 254°F and 265° up to 48 EFPY (using Reg. Guide 1.99, Rev. 2). It is suggested, that a new Emergency Response Guidelines should be developed for Beaver Valley Unit 1 before the vessel reaches 32 EFPY.

7.0 REFERENCES

1. U. S. Nuclear Regulatory Commission, 10CFR50, "Analysis of Potential Pressurized Thermal Shock Events," Federal Register, Vol. 50, No. 141, July 23, 1985.

2. Revision 2 to Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," USNRC, May 1988.

3. U. S. Nuclear Regulatory Commission, 10CFR50, Appendix G, "Fracture Toughness Requirements," Vol. 48, No. 104, May 1983.

4. ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure," 1972.

5. Yanichko, S.E. et. al., "Analysis of Capsule Westinghouse from the Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," Westinghouse Electric Corp. Report WCAP-12005, November 1988.

6. Emergency Response Guidelines - Revision 1, Westinghouse Owners Group, September 1, 1983.

7. Beaver Valley Unit 1 Low-Temperature Overpressure Prevention System (LTOPS) Setpoint Analysis at 16, 24, 32 and 48 EFPY, Letter Report DLW-90-528, January 23, 1990.

8. U. S. Nuclear Regulatory Commission, Proposed Rule, 10CFR50," Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Vol. 54, No. 246, December 26, 1989.

ATTACHMENT F

Beaver Valley Power Station, Unit No.1

Proposed Technical Specification Change No. 191

Beaver Valley Unit 1

Low Temperature Overpressure Protection System (LTOPS)

Setpoint Analysis at 16, 24, 32, and 48 EFPY

DLW-90-528

TABLE OF CONTENTS

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References

1. Westinghouse Project Letter DLW-89-519, "LTOPS Setpoint Design Changes", 01/31/89

DLW-90-528

Introduction

The Low Temperature Overpressure Protection System (LTOPS) limits pressure transients during cold shutdown, heatup, and cooldown operations in order to minimize the potential for imparing reactor vessel integrity when operating at or near vessel ductility limits. The imposition of this constraint, together with the requirements for reactor coolant pump operation create a set of both high and low temperature dependent pressure bounds. It is a regulatory requirement that the upper bound not be exceeded. Violating the lower bound is not a regulatory concern, but does result in damage to the reactor coolant pump number 1 seal. The goal of setpoint selection should be to prevent either bound from being exceeded below the LTOPS enable temperature (275 ^{O}F for Beaver Valley Unit 1).

Westinghouse Electric Corporation was advised by Duquesne Light Company that the 9.5 Effective Full Power Year (EFPY) pressure vessel temperature limit (Appendix G curve) applicable to the Beaver Valley Unit 1 reactor vessel, and the minimum pressure required to start the reactor coolant pumps, has created an operational constraint which is unduly limiting the rate at which the plant is able to heatup from cold shutdown. A meeting (May 11, 1988) was subsequently held at the Beaver Valley site with DLCo personnel in order to provide background and to explore possible solutions to the problem. As a result of that meeting, and a second meeting (July 14, 1988) held at the Westinghouse Energy Center, instruction was given to Westinghouse by DLCo which defined the kind of analysis that would best meet the requirements of Beaver Valley Unit 1. The results of that analysis were provided to DLCo by reference 1.

Subsequently, DLCo expressed an interest in obtaining LTOPS setpoints for several additional vessel exposures; extending to 48 EFPY. These additional setpoints were generated from steady-state pressure-temperature limits based on revision 2 of NRC Regulatory Guide 1.99, as applied to the calculation documented by reference 1.

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Summary of Results

The result of the analysis is summarized by Figure 1, illustrating the range, as a function of RCS temperature, of acceptable LTOPS setpoints for steady-state pressure-temperature limits based on revision 2 of NRC Reg. Guide 1.99. The figure also shows the reduction in setpoint range as a function of increasing reactor vessel exposure.

The result of the analysis is that, at low temperatures, the reactor coolant pump number 1 seal will not be protected. The calculation requires the assumption that one of the pressurizer power operated relief valves (PORV) dedicated to the LTOPS function will fail, resulting in a relatively higher overpressure. Consequently, the analysis indicates that no combination of setpoints exist that will preclude opening both PORV's. The result of both valves opening is a pressure undershoot much larger than that experienced for just a single valve opening. It should be noted, however, that the algorithm currently employed by Westinghouse is quite conservative, in that the maximum possible charging flow (typically, in the 400 gpm range) is assumed at the LTOPS setpoint selected for the parameter study. In reality, the charging flow probably will not exceed 150 gpm, considerably reducing the resulting overpressure. At low temperatures, it is recommended that DLCo take advantage of this conservatism and select the setpoints such that the differences between the setpoint pressures are maximized. At a vessel exposure of 9.5 EFPY with a reactor coolant system temperature of 80 ^{O}F, for example, select the first opening valve setpoint at 400 psig, and the second at 440 psig.

As the reactor coolant system temperature increases, the margin between the Appendix G limit and the reactor coolant pump no. 1 seal limit also increases. Eventually, a temperature is reached that comfortably allows the selection of staggered LTOPS setpoints which will preclude opening both PORV's; even with the high mass injection rates assumed by the analysis. From Figure 2.13 of reference 1, at a reactor coolant system temperature of 120 ^{O}F (vessel exposure of 9.5 EFPY), the first opening valve can be set at 395 psig, and the second opening valve at 475 psig. As vessel exposure increases, the temperature which allows these staggered setpoints also increases. At 48 EFPY, for example, the minimum temperature at which staggered setpoints can be obtained is about 180 ^{O}F. Table 1 summarizes, as a function of vessel exposure, a set of recommended setpoints which preclude multiple valve opening, and the estimated minimum temperatures at which these setpoints can be implemented.

Table 1 LTOPS Setpoint Summary

Reactor Vessel Exposure (EFPY)	Reactor Coolant Temperature (^{O}F)	Setpoint (psig)	
		Valve No. 1	Valve No. 2
9.5	120.0	395.0	475.0
16.0	136.0	390.0	470.0
24.0	152.0	390.0	470.0
32.0	162.0	390.0	470.0
48.0	180.0	390.0	470.0

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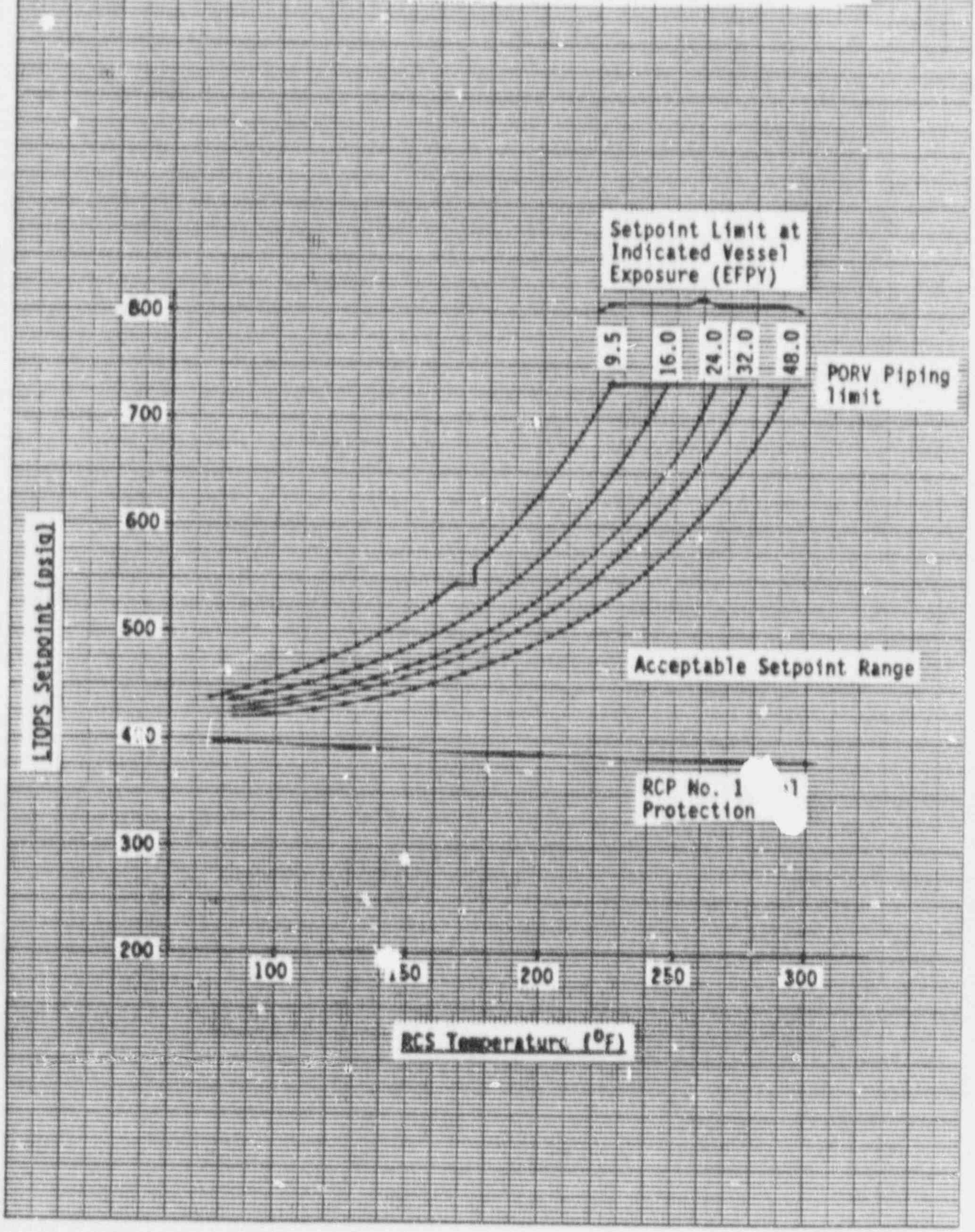
Figure 1 Acceptable LTOPS Setpoint Range vs. RCS Temperature

Beaver Valley Unit 1



DLW-90-528

Appendix G Limits

The steady-state cooldown Appendix G limits, without instrumentation uncertainty, were developed by the Materials Technology group within Westinghouse (reference Attachement 1), and form the basis for the LTOPS setpoint selection. The limits were generated for reactor vessel exposures of 16, 24, 32, and 48 EFPY.

LTOPS Setpoints Generation

The reactor coolant system pressure extrema, shown by Figure 2, is reproduced from Table 2.7 of reference 1. The reference 1 calculation considered mass injection events only. The setpoint evaluation for these additional exposures is similarly based. The correlation between the Appendix G limit and the maximum setpoint pressure was obtained by assuming a linear relationship between the maximum overpressure and the setpoint pressure:

$$\text{Setpoint Pressure} = a_0 + a_1(\text{Max. Overpressure})$$

By the methoo of least squares (reference Attachment 2):

$a_0 = -105.133$

$a_1 = 1.04821$

The maximum LTOPS setpoints, as a function of reactor coolant system temperature, were generated by substituting the appropriate Appendix G limit for the "Max. Overpressure" in the above equation. These setpoints are provided by Table 2:

Table 2 DLW Maximum LTOPS Setpoints for the Indicated Appendix G Limit[1]

	Vessel Exposure (EFPY)							
	16.0		24.0		32.0		48.0	
Temp (^{o}F)	Limit	Setpt	Limit	Setpt	Limit	Setpt	Limit	Setpt
85.0	516.6	436.4	509.6	429.0	506.3	425.6	502.2	421.3
100.0	523.8	443.9	515.1	434.8	511.1	430.6	506.0	425.3
140.0	553.1	474.6	537.6	458.4	530.2	450.6	521.1	441.1
180.0	605.3	529.3	577.5	500.2	564.3	486.4	548.3	469.6
220.0	698.3	626.8	648.9	575.0	625.4	550.4	596.6	520.2
247.1	800.0	733.4	-----	-----	-----	-----	-----	-----
260.0			776.0	708.3	734.2	664.5	682.9	610.7
265.5			800.0	733.4	-----	-----	-----	-----
276.3					800.0	733.4	-----	-----
280.0							748.7	679.7
292.3							800.0	733.4

1. Indicated pressure in units of psig.

DLW-90-528

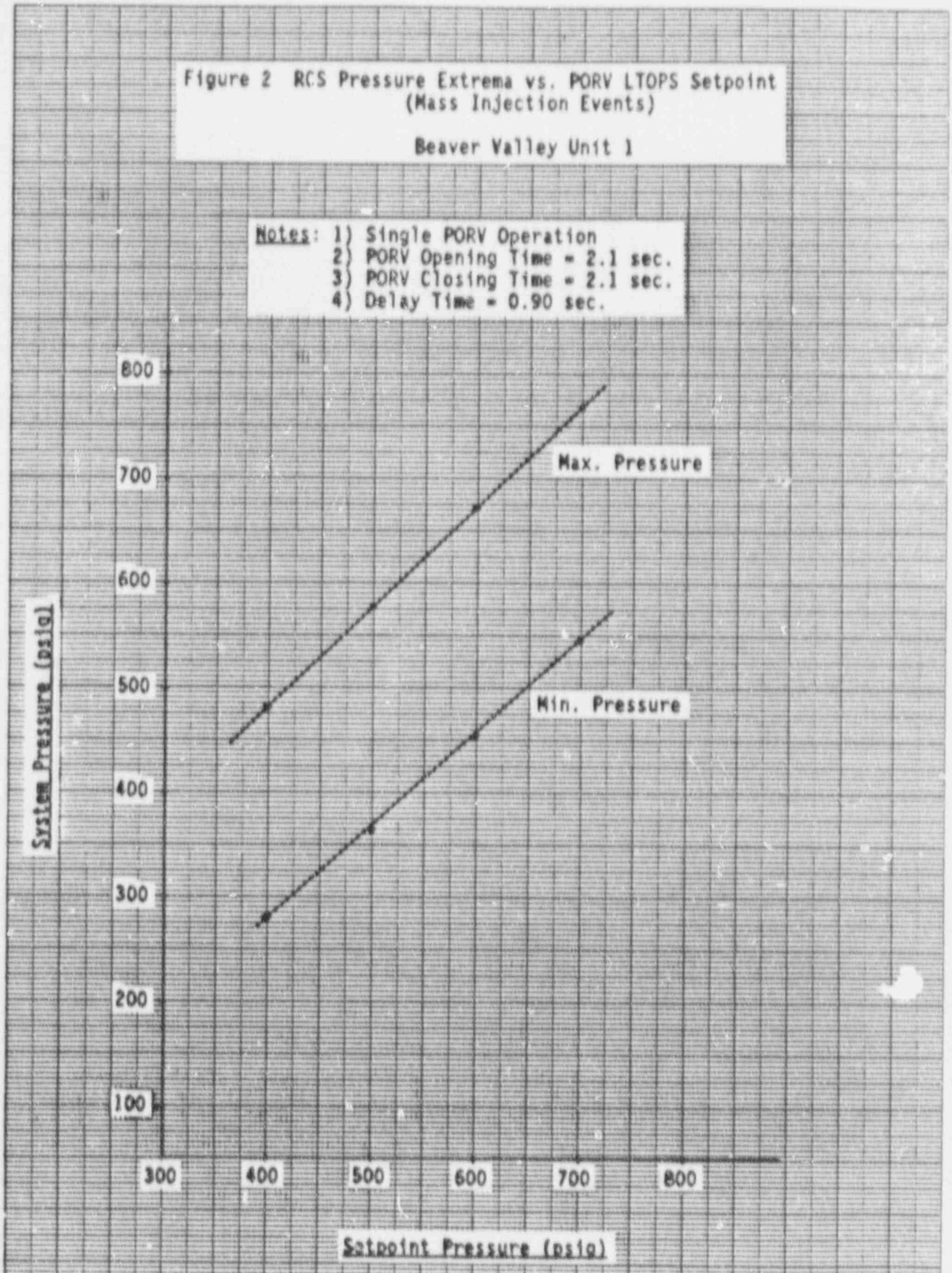


Figure 2 RCS Pressure Extrema vs. PORV LTOPS Setpoint
(Mass Injection Events)
Beaver Valley Unit 1
Notes: 1) Single PORV Operation
2) PORV Opening Time = 2.1 sec.
3) PORV Closing Time = 2.1 sec.
4) Delay Time = 0.90 sec.
Max. Pressure
Min. Pressure
System Pressure (psig)
800
700
600
500
400
300
200
100
300
400
500
600
700
800
Setpoint Pressure (psig)

DLW-90-508

The setpoints from Table 2 are reproduced, with roundoff, on Table 3. In addition to the maximum allowed LTOPS setpoint, the table provides the minimum allowed setpoint, for the RCP No. 1 seal protection, and the difference between the maximum and minimum setpoints. This difference provides the range, as a function of reactor coolant temperature, from which the LTOPS setpoint(s) can be selected.

The maximum LTOPS setpoints, from Table 2, are graphically shown as a function of reactor coolant system temperature, parametric with vessel exposure, on Figure 3. The acceptable setpoint range is indicated to the right of the bounding curves. The 9.5 EFPY limit has been copied directly from Figure 2.16 of reference 1. Note that the "notch" accounting for the reactor vessel flange ligaments has essentially disappeared for vessel exposures in excess of 16 EFPY. The minimum system pressure required to protect the reactor coolant pump number 1 seal is unaffected by neutron fluence, and thus remains unchanged from reference 1.

Table 3 DLW Maximum LTOPS Setpoint Spread at the Indicated Appendix G Limit

		Vessel Exposure (EFPY)									
		9.5		16.0		24.0		32.0		48.0	
RCS Temp (Deg F)	Min Setpt	Max Setpt	Delta	Max Setpt	Delta	Max Setpt	Delta	Max Setpt	Delta	Max Setpt	Delta
85.0	396.0	444.0	48.0	436.0	40.0	429.0	33.0	426.0	30.0	421.0	25.0
100.0	394.0	454.0	60.0	444.0	50.0	435.0	41.0	431.0	37.0	425.0	31.0
120.0	392.0	472.0	80.0	457.0	65.0	445.0	53.0	439.0	47.0	432.0	40.0
140.0	389.0	496.0	107.0	475.0	86.0	458.0	69.0	451.0	62.0	441.0	52.0
160.0	388.0	529.0	141.0	498.0	110.0	476.0	88.0	466.0	78.0	453.0	65.0
180.0	386.0	572.0	186.0	529.0	143.0	500.0	114.0	486.0	100.0	470.0	84.0
200.0	385.0	629.0	244.0	571.0	186.0	532.0	147.0	514.0	129.0	491.0	106.0

Notes: Min Setpt prevents RCP No. 1 seal closure

Max Setpt prevents exceeding the Appendix G limit

Setpoint pressures in units of psig

DLW-90-508

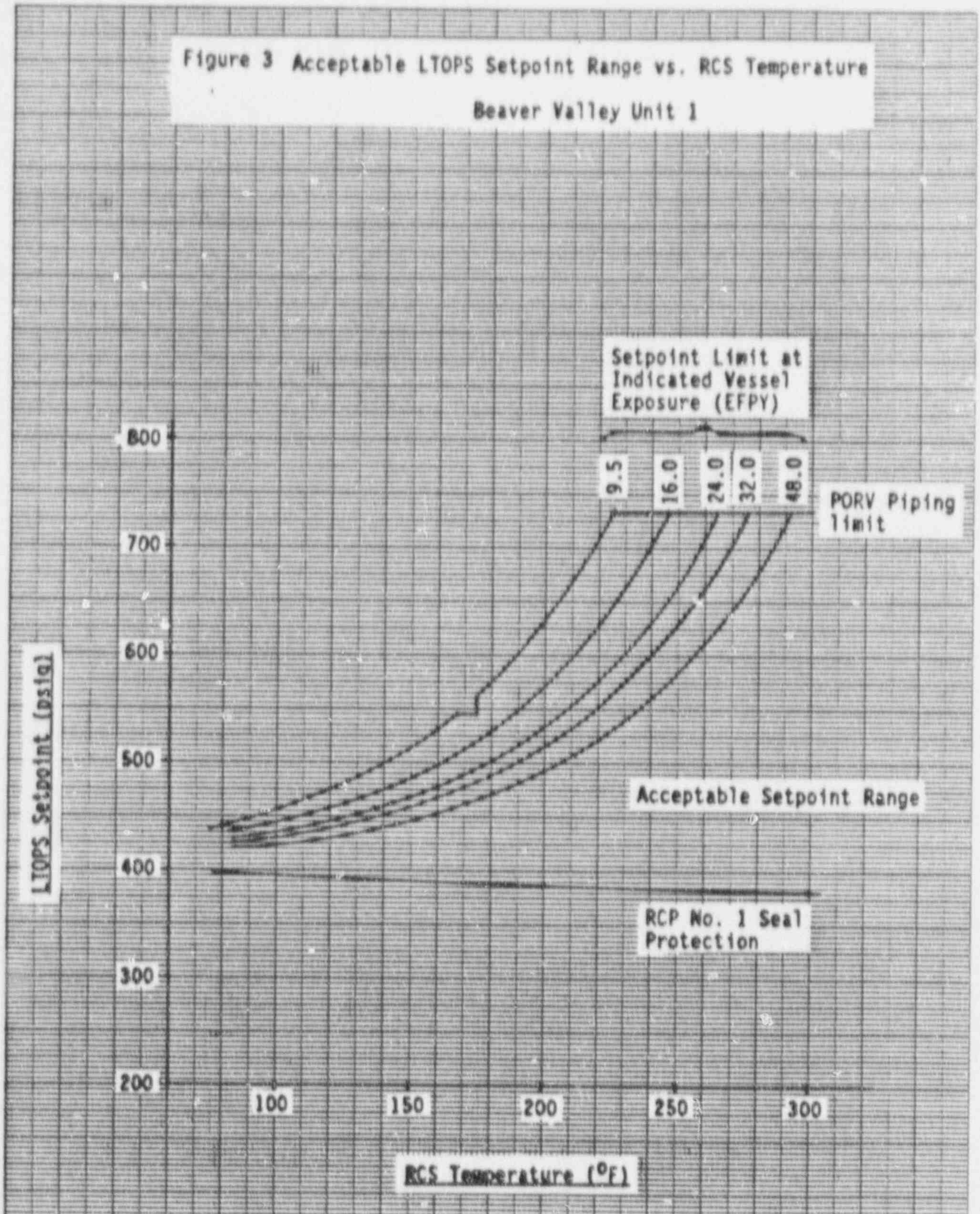


Figure 3 Acceptable LTOPS Setpoint Range vs. RCS Temperature
Beaver Valley Unit 1
Setpoint Limit at Indicated Vessel Exposure (EFPY)
9.5
16.0
24.0
32.0
48.0
PORV Piping limit
Acceptable Setpoint Range
RCP No. 1 Seal Protection
LTOPS Setpoint (psig)
800
700
600
500
400
300
200
100
150
200
250
300
RCS Temperature (°F)

DLW-90-502 8

Attachment 1

Appendix G Limits

DLW-90-528

MT-SMART-177-(89)

From MATERIALS TECHNOLOGY
WIN 236-6465
Date November 3, 1989
Subject Data Points for Developing LTOP Set Points for Beaver Valley Unit 1 Vessel

To J. P. Kutz -- EC/EAST 425C

cc: N. P. Mueller
D. C. Adamonis
T. A. Meyer
S. S. Palusamy
T. R. Mager
R. D. Rishel

As a part of Reactor Vessel Operating limit study for Beaver Valley Unit 1, Structural Materials and Reliability Technology has completed the generation of heatup and cooldown curves. The following information are attached for your use in the development of LTOP setpoints.

- O Steady State Cooldown for 16, 24, 32 and 48 EFPYs
- O No instrumentation margins for possible error in pressure and temperature.

Please call us, if you need any additional infr mation.

N. K. Ray
Structural Materials & Reliability Technology

DLW-90-528

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 16.000 EFP YEARS
FLAW DEPTH = AOWIN :

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	516.56	21	185.000	614.08	41	285.000	1028.03
2	90.000	518.81	22	190.000	623.68	42	290.000	1068.43
3	95.000	521.24	23	195.000	634.00	43	295.000	1111.83
4	100.000	523.84	24	200.000	645.09	44	300.000	1158.48
5	105.000	526.64	25	205.000	657.00	45	305.000	1208.61
6	110.000	529.66	26	210.000	668.69	46	310.000	1262.47
7	115.000	532.89	27	215.000	683.47	47	315.000	1320.28
8	120.000	536.37	28	220.000	698.30	48	320.000	1382.35
9	125.000	540.1[illegible]	29	225.000	714.09	49	325.000	1448.64
10	130.000	544.[illegible]4	30	230.000	731.18	50	330.000	1520.22
11	135.000	54[illegible].47	31	235.000	748.58	51	335.000	1596.56
12	140.000	[illegible]53.12	32	240.000	769.20	52	340.000	1678.81
13	145.000	[illegible].12	33	245.000	790.48	53	345.000	1766.78
14	150.000	[illegible]63.[illegible]9	34	250.000	813.13	54	350.000	1861.06
15	155.000	56[illegible].[illegible]7	35	255.000	837.73	55	355.000	1961.84
16	160.000	575.39	36	260.000	863.91	56	360.000	2069.82
17	165.000	582.07	37	265.000	892.31	57	365.000	2185.28
18	170.000	589.25	38	270.000	922.60	58	370.000	2308.81
19	175.000	596.98	39	275.000	955.32	59	375.000	2440.59
20	180.000	605.28	40	280.000	990.36			

DLW COOLDOWN CURVES REG. GUIDE 1.99,REV.2 10/28/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 24.000 EFP YEARS
FLAW DEPTH = 4OWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	509.64	22	190.000	591.71	43	295.000	966.35
2	90.000	511.37	23	195.000	599.62	44	300.000	1002.36
3	95.000	513.13	24	200.000	608.11	45	305.000	1040.92
4	100.000	515.13	25	205.000	617.13	46	310.000	1082.32
5	105.000	517.28	26	210.000	626.96	47	315.000	1126.81
6	110.000	519.58	27	215.000	637.52	48	320.000	1174.58
7	115.000	522.07	28	220.000	648.88	49	325.000	1225.86
8	120.000	524.73	29	225.000	660.94	50	330.000	1280.96
9	125.000	527.60	30	230.000	674.07	51	335.000	1340.09
10	130.000	530.69	31	235.000	688.19	52	340.000	1403.34
11	135.000	534.00	32	240.000	703.35	53	345.000	1471.56
12	140.000	537.57	33	245.000	719.50	54	350.000	1544.59
13	145.000	541.40	34	250.000	737.04	55	355.000	1622.91
14	150.000	545.52	35	255.000	755.86	56	360.000	1706.90
15	155.000	549.95	36	260.000	775.97	57	365.000	1796.70
16	160.000	554.71	37	265.000	797.74	58	370.000	1893.03
17	165.000	559.72	38	270.000	820.96	59	375.000	1996.36
18	170.000	565.23	39	275.000	846.11	60	380.000	2106.72
19	175.000	571.15	40	280.000	872.95	61	385.000	2224.65
20	180.000	577.51	41	285.000	901.97	62	390.000	2350.78
21	185.000	584.35	42	290.000	933.05			

DLW-90-522

DLW COOLDOWN CURVES REG. GUIDE 1.99,REV.2 10/25/89

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIATION PERIOD = 32.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	506.32	23	195.000	583.25	45	305.000	960.95
2	90.000	507.79	24	200.000	590.53	46	310.000	996.58
3	95.000	509.38	25	205.000	598.35	47	315.000	1034.71
4	100.000	511.09	26	210.000	606.74	48	320.000	1075.65
5	105.000	512.93	27	215.000	615.66	49	325.000	1119.60
6	110.000	514.81	28	220.000	625.38	50	330.000	1166.53
7	115.000	516.93	29	225.000	635.83	51	335.000	1217.58
8	120.000	519.21	30	230.000	647.06	52	340.000	1272.12
9	125.000	521.67	31	235.000	659.10	53	345.000	1330.60
10	130.000	524.30	32	240.000	671.96	54	350.000	1393.35
11	135.000	527.14	33	245.000	685.92	55	355.000	1460.60
12	140.000	530.19	34	250.000	700.91	56	360.000	1537.91
13	145.000	533.47	35	255.000	716.88	57	365.000	1610.37
14	150.000	536.99	36	260.000	734.22	58	370.000	1693.13
15	155.000	540.78	37	265.000	752.83	59	375.000	1782.37
16	160.000	544.86	38	270.000	772.71	60	380.000	1877.66
17	165.000	549.24	39	275.000	794.25	61	385.000	1978.75
18	170.000	553.95	40	280.000	817.19	62	390.000	2088.90
19	175.000	559.01	41	285.000	842.06	63	395.000	2205.73
20	180.000	564.34	42	290.000	868.60	64	400.000	2330.52
21	185.000	570.20	43	295.000	897.32	65	405.000	2463.93
22	190.000	576.49	44	300.000	928.02			

DLW-90-528

DLW-90-528

THE FOLLOWING DATA WERE PLOTTED FOR COOLDOWN PROFILE 1 (STEADY-STATE COOLDOWN)

IRRADIAT..ON PERIOD = 48.000 EFP YEARS
FLAW DEPTH = AOWIN T

	INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)		INDICATED TEMPERATURE (DEG.F)	INDICATED PRESSURE (PSI)
1	85.000	502.23	24	200.000	568.81	47	315.000	921.21
2	90.000	503.40	25	205.000	575.10	48	320.000	953.82
3	95.000	504.68	26	210.000	581.76	49	325.000	988.78
4	100.000	506.01	27	215.000	588.92	50	330.000	1026.29
5	105.000	507.47	28	220.000	596.62	51	335.000	1066.56
6	110.000	509.03	29	225.000	604.90	52	340.000	1109.83
7	115.000	510.72	30	230.000	613.67	53	345.000	1156.23
8	120.000	512.52	31	235.000	623.24	54	350.000	1206.30
9	125.000	514.37	32	240.000	633.53	55	355.000	1259.98
10	130.000	516.46	33	245.000	644.58	56	360.000	1317.69
11	135.000	518.71	34	250.000	656.46	57	365.000	1378.49
12	140.000	521.13	35	255.000	669.10	58	370.000	1445.86
13	145.000	523.72	36	260.000	682.85	59	375.000	1517.00
14	150.000	526.52	37	265.000	697.62	60	380.000	1593.29
15	155.000	529.52	38	270.000	713.33	61	385.000	1675.10
16	160.000	532.75	39	275.000	730.40	62	390.000	1762.60
17	165.000	536.22	40	280.000	748.74	63	395.000	1856.78
18	170.000	539.95	41	285.000	768.30	64	400.000	1957.37
19	175.000	543.96	42	290.000	789.51	65	405.000	2064.91
20	180.000	548.27	43	295.000	812.09	66	410.000	2179.85
21	185.000	552.91	44	300.000	836.61	67	415.000	2303.18
22	190.000	557.90	45	305.000	862.71	68	420.000	2434.66
23	195.000	563.14	46	310.000	891.01			

DLW-90-5028

Attachment 2

Overpressure-Setpoint Correlation

DLW-90-528

<u>Overpressure-Setpoint Correlation</u>

The analytical correlation between reactor coolant system overpressure and LTOPS setpoint pressure is derived from a PC based fitting routine, with the assumption of a linear relationship:

$$\text{Setpoint Pressure} = a_0 + a_1(\text{Max. Overpressure})$$

The output from the fitting routine is shown below:

TERM	COEFFICIENT
0	-1.051334000E+02
1	1.048214000E+00

WHAT NEXT ? 2

X-ACTUAL	Y-ACTUAL	Y-CALC	DIFFERENCE	PCT DIFF.
4.8200E+02	4.0000E+02	4.0011E+02	-1.0547E-01	-2.6360E-02
5.7700E+02	5.0000E+02	4.9969E+02	3.1427E-01	6.2894E-02
6.7300E+02	6.0000E+02	6.0031E+02	-3.1427E-01	-5.2351E-02
7.6800E+02	7.0000E+02	6.9989E+02	1.0547E-01	1.5069E-02

STD ERROR OF ESTIMATE FOR Y = .3314956

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Energy Systems

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DLW-91-163
June 18, 1991

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PS-DLW-0207
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Ref: 1) DLW-91-155, 6/06/91
2) DLW-90-528, 1/23/90

Dear Mr. Halliday:

DUQUESNE LIGHT COMPANY
BEAVER VALLEY POWER STATION UNITS 1 AND 2
Steam Generator Tube Plugging Analysis Program
LTOPS Setpoint

Section 4.2 of WCAP-12966 (Duquesne Light Company Beaver Valley Power Station Units 1 and 2, 20 Percent Steam Generator Tube Plugging Analysis Program Engineering and Licensing Report), transmitted via Reference 1, includes an evaluation of the effect of 20% steam generator tube plugging on the Cold Overpressure Transients and the Low Temperature Overpressure Prevention System (LTOPS) setpoints. The conclusion of the evaluation is that the current setpoints are nonconservative and that the required setpoint change to remove the nonconservatism is a less than 4 psi reduction in the setpoint. The evaluation also concludes that the nonconservatism is small and exceeding the Appendix G limit by this amount will have no impact on the probability of brittle vessel fracture.

At the Offsite Review Committee meeting on June 11, 1991, Duquesne Light Company (DLCo) informed Westinghouse that it was their intention to reduce the LTOPS setpoint for Unit 1 by 4 psi to accommodate the nonconservatism identified in WCAP-12966. DLCo stated that the present Unit 1 LTOPS Technical Specification 3.4.9.3 requires a setpoint of 350 psig and is applicable for up to 9.5 effective full power years (EFPYs). Via Reference 2, Westinghouse provided revised LTOPS setpoints for Units 1 and 2 based on setpoint calculation methodology that does not include instrument uncertainties. Based on the information provided in Reference 2, DLCo has prepared and submitted to the NRC for approval a technical specification revision to increase the Unit 1 LTOPS setpoint to 444 psig for up to 9.5 EFPYs. Also, DLCo is presently preparing an additional revision to the subject Unit 1 technical specification to incorporate new Appendix G limit curves for up to 16 EFPYs and to revise the LTOPS setpoint to 436 psig consistent with the new Appendix G limits. Based on the results contained in WCAP-12966, DLCo has stated that they will reduce the 444 psig and 436 psig setpoints in the two proposed technical specification revisions by 4 psi to accommodate the Cold Overpressure

Transients associated with 20% tube plugging. Once these changes are approved by the NRC and implemented by DLCo, the technical specification setpoint will prevent low temperature overpressure for the 20% tube plugging condition. Until NRC approval and DLCo implementation of the revisions to the subject technical specification, the installed LTOPS setpoint will not include the 4 psi reduction but will continue to include the margin associated with the old calculation methodology which includes instrument uncertainties. Based on this LTOPS setpoint implementation status for Unit 1, DLCo requested Westinghouse to review the results contained in WCAP-12966 and provide a recommendation on whether the present setpoint of 350 psi should be reduced by 4 psi.

Westinghouse has reviewed this item as requested and has confirmed that the present LTOPS setpoint of 350 psig does not need to be reduced by 4 psi. The technical basis for this conclusion is that the margin in the present setpoint of 350 psi is greater than the 4 psi nonconservatism that results from the change in cold overpressure transients due to 20% tube plugging. Consequently, the existing Unit 1 LTOPS setpoint is conservative for tube plugging levels up to the 20% limit. With respect to the proposed technical specification revisions, it is acceptable to reduce the setpoints provided in Reference 2 by 4 psi to accommodate the 20% tube plugging condition, resulting in Unit 1 setpoints of 440 psig for up to 9.5 EFPYs and 432 psig for up to 16 EFPYs.

Please advise if there are any questions or comments on this information or if we may be of additional assistance.

Very truly yours,

J. N. Steinmetz, Manager
Central Region
Customer Projects Department

cc: NERU Records
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