FENOC

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October 31, 2001
L-01-135

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

Subject: Beaver Valley Power Station, Unit No. 1 and No. 2
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
License Amendment Request Nos. 295 and 167

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above licenses in the form of changes to the Technical Specifications. The proposed changes will create a Pressure and Temperature Limits Report (PTLR) for each unit based on the guidance provided by Generic Letter 96-03 "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits."

The proposed Technical Specification changes for Unit No. 1 and Unit No. 2 are presented in Attachments A-1 and A-2, respectively. The safety analysis and no significant hazard evaluation is presented in Attachment B. The proposed PTLRs for Unit No. 1 and Unit No. 2 are presented in Attachments C-1 and C-2, respectively. The changes proposed to the Technical Specification Bases are provided for information only in Attachments D-1 and D-2 for Unit No. 1 and Unit No. 2, respectively. The Beaver Valley Power Station (BVPS) Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. Attachment E is provided as an aid to the review of the changes proposed in this License Amendment Request. The attachment contains a duplication of a table appearing in Generic Letter 96-03, with additional columns that identify the location of each requirement in the BVPS Unit 1 and Unit 2 PTLRs.

This change has been reviewed by the Beaver Valley review committees. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazard evaluation.

A001

FENOC requests NRC approval of License Amendment Request 167 for Beaver Valley Unit 2 by February 1, 2002, to support refueling outage 2R09. An implementation period of up to 60 days is requested following the effective date of this amendment.

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs at 724-682-5203.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 31, 2001.

Sincerely,

Lew W. Myers

Lew W. Myers

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

ATTACHMENT A-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 295

The following is a list of the affected TS pages:

Affected Pages	Pending LARs
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DEFINITIONS

SAFETY LIMITS

BASES

ADMINISTRATIVE CONTROLS

This page contains changes proposed by LAR 292.

Figure Index

Moved to PTLR.

2) Major changes in the design of radwaste treatment systems (liquid, gaseous and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions.)

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

CORE OPERATING LIMITS REPORT

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these limits is addressed in individual specifications.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.38 The PTLR is the unit specific document that provides the reactor vessel pressure and temperature (P/T) limits, including heatup and cooldown rates and the Overpressure Protection System setpoint and enable temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.6. Plant operation within these operating limits is addressed in Specification 3.4.9.1, "Reactor Coolant System Pressure/Temperature Limits", and Specification 3.4.9.3, "Reactor Coolant System Overpressure Protection System."

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4[(1)].

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 Each charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

4.1.2.4.2 All charging pumps, except the above required OPERABLE pump, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the inservice RCS cold legs is ≤ the enable temperature ~~set forth in Specification 3.4.9.3~~specified in the PTLR by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged.

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(1) A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the non-isolated RCS cold legs is ≤ the enable temperature specified in the PTLR.~~set forth in Specification 3.4.9.3.~~

This page contains changes proposed by LAR 292.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,#
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,#
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,#
4. Residual Heat Removal Pump (A) and a heat exchanger,**
5. Residual Heat Removal Pump (B) and a second heat exchanger.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: Modes 4 AND 5.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and Residual Heat Removal pumps may be de-energized for up to 1 hour provided: 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration and 2) core outlet temperature is maintained at least 10°F below saturation temperature. For purposes of this specification, the addition of borated water to the RCS does not constitute dilution of the RCS boron concentration provided the boron concentration of the borated water being added is greater than the minimum required to satisfy the requirements of Specification 3.1.1.1 for Mode 4; or Specification 3.1.1.2 for Mode 5.

The first reactor coolant pump in a non-isolated loop shall not be started with one or more non-isolated RCS cold leg temperatures less than or equal to the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~, unless the secondary side water temperature of each steam generator in a non-isolated loop is less than 50°F above each of the non-isolated RCS cold leg temperatures.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

Moved to PTLR.

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure ~~shall be limited in accordance~~ with the limits ~~lines shown on Figure 3.4-2 and Figure 3.4-3~~ that are specified in the PTLR during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. ~~with:~~

~~a. A maximum heatup of 100°F in any one hour period,~~

~~b. A maximum cooldown of 100°F in any one hour period, and~~

~~c. A maximum temperature change of less than or equal to 5°F in any one hour period, during hydrostatic testing operations above system design pressure.~~

APPLICABILITY: MODES 1, 2[(1)], 3, 4 and 5.

Moved to PTLR.

ACTION:

With any of the above limits specified in the PTLR exceeded:

a. Restore the temperature and/or pressure to within the limit within 30 minutes, and

b. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System within 72 hours, and

c. Determine, from Action b above, that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

(1) See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits specified in the PTLR at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line specified in the PTLR within 15 minutes prior to achieving reactor criticality.

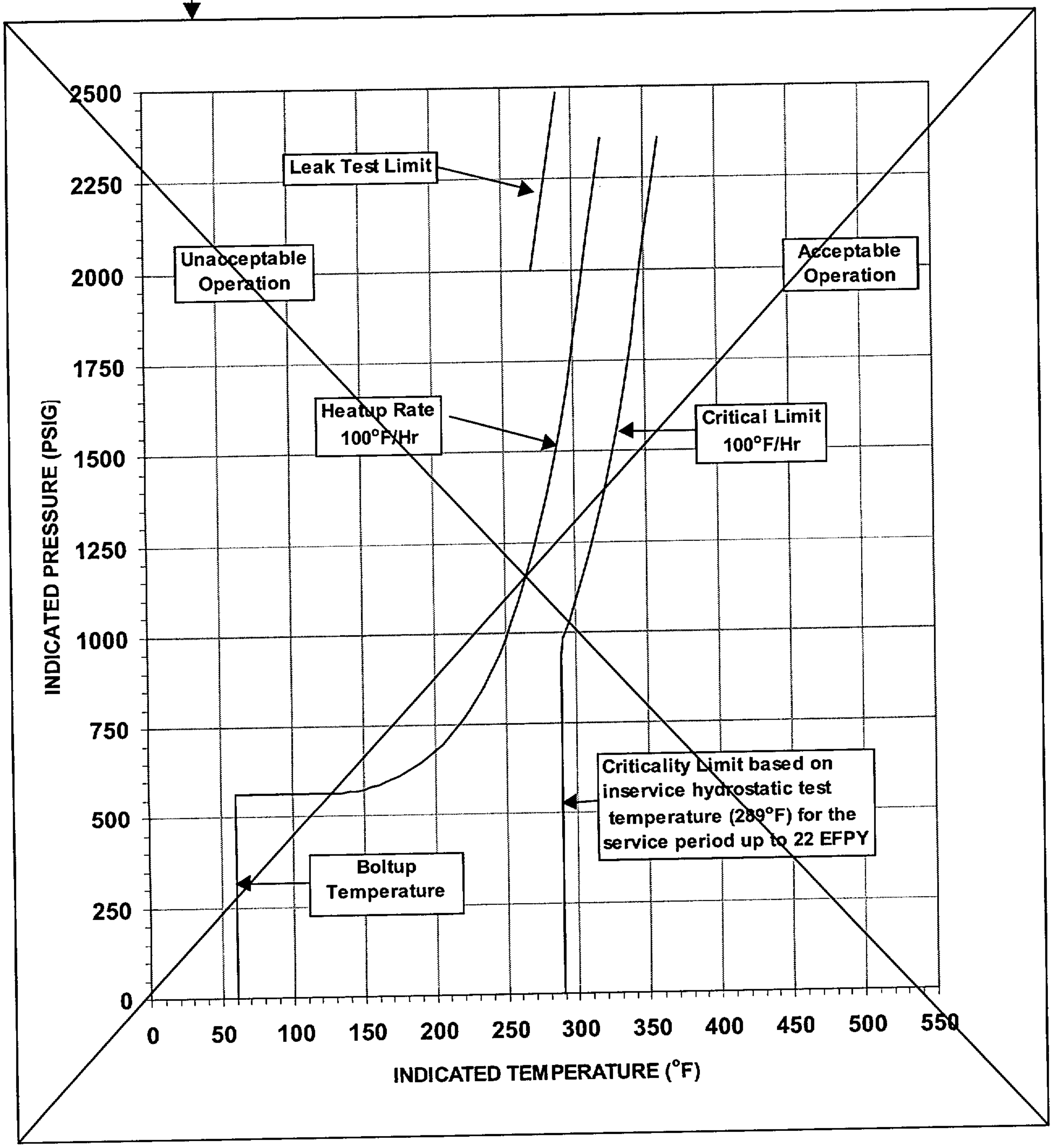
c. The reactor vessel material irradiation surveillance specimens shall be removed and examined in accordance with 10CFR50, Appendix H, to determine changes in material properties. The results of these examinations shall be used to update the PTLR~~Figures 3.4-2 and 3.4-3~~.

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~LIMITING MATERIAL:~~ ~~INTERMEDIATE & LOWER SHELL PLATE~~
~~LIMITING ART VALUES AT 22 EFPY:~~ ~~1/4T, 233°F~~
~~3/4T, 196°F~~

This page contains changes proposed by LAR 292.

Moved to PTLR.



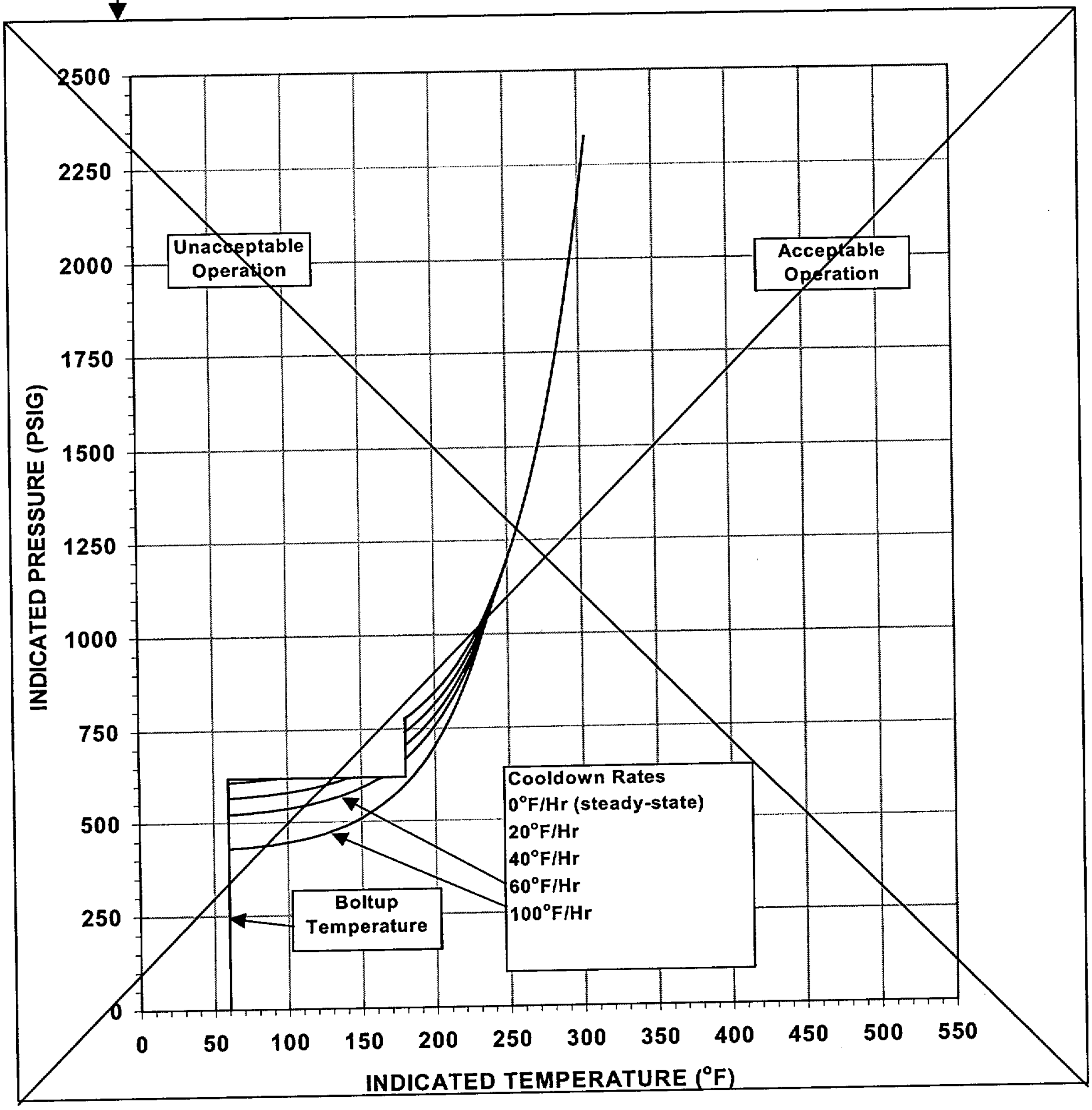
~~FIGURE 3.4-2~~
~~Beaver Valley Unit 1 Reactor Coolant System Heatup~~
~~Limitations Applicable for the First 22 EFPY~~

(Proposed Wording)

~~MATERIAL PROPERTY BASIS~~
~~LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE~~
~~LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F~~
~~3/4T, 196°F~~

Moved to PTLR.

This page contains changes proposed by LAR 292.



~~FIGURE 3.4-3~~

~~Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First 22 EFPY~~

~~(NEXT PAGE IS 3/4 4-27)~~

(Proposed Wording)

This page contains changes proposed by LAR 292.

3.4.9.3 An overpressure protection system shall be OPERABLE with a maximum of one charging pump[1] capable of injecting into the RCS and the accumulators isolated[2] and either a or b below:

a. Two power operated relief valves (PORVs) with a nominal maximum lift setting ~~less than or equal to 403 psig~~ within limits specified in the PTLR, or

b. The RCS depressurized and an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: Mode 4 when any RCS cold leg temperature is less than or equal to an enable temperature ~~of~~ specified in the PTLR ~~343°F~~,
Mode 5,
Mode 6 when the reactor vessel head is on.

Moved to PTLR.

ACTION:

a. With two or more charging pumps capable of injecting into the RCS, immediately initiate action to verify a maximum of one charging pump is capable of injecting into the RCS or depressurize and vent the RCS through a 2.07 square inch or larger vent within 12 hours.

b. With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves in the PTLR, isolate the affected accumulator within 1 hour or increase the RCS cold leg temperature above the enable temperature specified in the PTLR within the next 12 hours or depressurize the affected accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed by the heatup and cooldown curves in the PTLR within the next 12 hours.

c. With one PORV inoperable in MODE 4 (when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 2.07 square inch or larger vent within the next 12 hours.

(1) Two charging pumps may be capable of injecting into the RCS for pump swap operation for less than or equal to 1 hour.

(2) Accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves provided in the PTLR.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or depressurize and vent the RCS through a 2.07 square inch or larger vent within the next 12 hours.

e. With two PORVs inoperable, depressurize and vent the RCS through a 2.07 square inch or larger vent within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Verify at least once per 12 hours that:

a. A maximum of one charging pump is capable of injecting into the RCS, and

b. Each accumulator is isolated; however, with the accumulator pressure less than the low temperature overpressure protection setpoint, the accumulator discharge isolation valves may be opened to perform accumulator discharge check valve testing.

4.4.9.3.2 When PORVs are being used for overpressure protection, demonstrate each PORV is OPERABLE by:

a. Verifying each PORV block valve is open for each required PORV at least once per 72 hours, and

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and placed in operation after decreasing the RCS cold leg temperature to less than or equal to the enable temperature <u>specified in the PTLR</u> and at least once per 31 days, and

c. Performance of a CHANNEL CALIBRATION on each required PORV actuation channel at least once per 18 months.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE Low Head Safety Injection Pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted in accordance with 10 CFR 50.4 within 30 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps except the above required OPERABLE pumps, shall be demonstrated inoperable at least once per 12 hours whenever the temperature of one or more of the non-isolated RCS cold legs is ≤ the enable temperature specified in the PTLR ~~set forth in Specification 3.4.9.3~~ by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged.

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the non-isolated RCS cold legs is ≤ the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 BORON INJECTION SYSTEM

BORON INJECTION TANK < 350°F

LIMITING CONDITION FOR OPERATION

3.5.4.1.2 The boron injection tank flow path shall be isolated and power removed from the inlet or outlet valves.

APPLICABILITY: When the temperature of one or more of the non-isolated RCS cold legs is ≤ the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~.

ACTION:

With the boron injection tank not isolated, isolate the tank flow path and remove power from the inlet or outlet valves.

SURVEILLANCE REQUIREMENTS

4.5.4.1.2 The boron injection tank flow path shall be verified isolated by verifying at least once per 7 days that the Boron Injection Tank inlet or outlet valves are closed and de-energized except for purposes of flow testing or valve stroke testing.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. Deleted, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of ~~operation shown~~ ~~in the~~ P/T limit curves of the PTLR~~on Figures 3.4-2 and 3.4-3~~.

Moved to PTLR.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the unacceptable ~~region of operation~~ on the P/T limit curves of the PTLR~~Figures 3.4-2 and 3.4-3~~, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

Moved to PTLR.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable ~~region for operation as~~ shown on the P/T limit curves of the PTLR ~~of Figures 3.4-2 and 3.4-3~~ at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour.

Moved to PTLR.

CORE OPERATING LIMITS REPORT (Continued)

NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April 1995 (Westinghouse Proprietary).

As described in reference documents listed above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.6% of rated thermal power may be used when input for reactor thermal power measurement of feedwater flow is by the leading edge flow meter (LEFM).

Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997.

Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ System" Revision 0, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.6 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. Reactor Coolant System pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, hydrostatic testing, Overpressure Protection System (OPPS) enable temperature, and Power Operated Relief Valve (PORV) lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

1. Specification 3.4.9.1, "Reactor Coolant System Pressure/Temperature Limits", and

2. Specification 3.4.9.3, "Reactor Coolant System Overpressure Protection Systems".

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed

and approved by the NRC, specifically those described in the following documents:

1. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with WCAP-14040-NP-A, Rev. 2, and

2. the OPPS limits, i.e., PORV pressure relief setpoint and OPPS enable temperature, were developed in accordance with WCAP-14040-NP-A, Rev. 2.

The methodology listed in WCAP-14040-NP-A was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

ATTACHMENT A-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 167

The following is a list of the affected TS pages:

Affected Pages	Pending LARs
II	
XIV	
1-7	
3/4 1-8	
3/4 1-11	
3/4 4-3	157
3/4 4-30	
3/4 4-30a	
3/4 4-31	
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DEFINITIONS

SAFETY LIMITS

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ADMINISTRATIVE CONTROLS

MAJOR CHANGES

1.35 MAJOR CHANGES to radioactive waste systems (liquid, gaseous and solid), as addressed in the PROCESS CONTROL PROGRAM, shall include the following:

1) MAJOR CHANGES in process equipment, components, structures, and effluent monitoring instrumentation from those described in the Final Safety Analysis Report (FSAR) or the Hazards Summary Report and evaluated in the staff's Safety Evaluation Report (SER) (e.g., deletion of evaporators and installation of demineralizers; use of fluidized bed calciner/incineration in place of cement solidification systems);

2) MAJOR CHANGES in the design of radwaste treatment systems (liquid, gaseous, and solid) that could significantly increase the quantities or activity of effluents released or volumes of solid waste stored or shipped offsite from those previously considered in the FSAR and SER (e.g., use of asphalt system in place of cement);

3) Changes in system design which may invalidate the accident analysis as described in the SER (e.g., changes in tank capacity that would alter the curies released); and

4) Changes in system design that could potentially result in a significant increase in occupational exposure of operating personnel (e.g., use of temporary equipment without adequate shielding provisions).

MEMBER(S) OF THE PUBLIC

1.36 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

CORE OPERATING LIMITS REPORT

1.37 The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.5. Plant operation within these limits is addressed in individual specifications.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.38 The PTLR is the unit specific document that provides the reactor vessel pressure and temperature (P/T) limits, including heatup and cooldown rates and the Overpressure Protection System setpoint and enable temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification

6.9.6. Plant operation within these operating limits is addressed in Specification 3.4.9.1, "Reactor Coolant System Pressure/Temperature Limits", and Specification 3.4.9.3, "Reactor Coolant System Overpressure Protection System."

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 Each of the following boron injection flow paths shall be OPERABLE:

a. The flow path from the boric acid tanks via a boric acid transfer pump and one charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via one charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With the flow path from the boric acid tanks inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

b. With the flow path from the refueling water storage tank inoperable, restore the flow path to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

2. Verifying that the temperature of the heat traced portion of the flow path from the boric acid tanks is ≥ 65°F when the ambient air temperature of the Auxiliary Building is < 65°F.

b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

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*The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 4.1.2.3.2 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding the enable temperature specified in the PTLR plus 25°F~~375°F~~, whichever comes first.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3[(1)].

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 1 percent $\Delta k/k$ at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 Each charging pump shall be demonstrated OPERABLE pursuant to Specification 4.5.2.b.1.

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(1) The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pump declared inoperable pursuant to Specification 3.4.9.3 provided that the centrifugal charging pump is restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding the enable temperature specified in the PTLR plus 25°F~~375°F~~, whichever comes first.

This page contains changes proposed by LAR 157.

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE.

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,#
2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,#
3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,#
4. Residual Heat Removal Pump (A) and the (A) RHR heat exchanger,**
5. Residual Heat Removal Pump (B) and the (B) RHR heat exchanger.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: MODES 4 and 5.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and Residual Heat Removal pumps may be deenergized for up to 1 hour provided: 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

No reactor coolant pump in a non-isolated loop shall be started with one or more non-isolated RCS cold leg temperatures less than or equal to the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~, unless the secondary side water temperature of each steam generator in a non-isolated loop is less than 50°F above each of the non-isolated RCS cold leg temperatures.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

Moved to PTLR.

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure ~~shall be limited in~~ accordance with the limits ~~lines shown on Figures 3.4-2 and 3.4-3~~ that are specified in the PTLR during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. ~~with:~~

~~a. A maximum heatup of 60°F in any one hour period,~~

~~b. A maximum cooldown of 100°F in any one hour period, and~~

~~c. A maximum temperature change of less than or equal to 5°F in any one hour period during hydrostatic testing operations above system design pressure.~~

Moved to PTLR.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

With any of the above limits specified in the PTLR exceeded:

a. Restore the temperature and/or pressure to within the limit within 30 minutes, and

b. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System within 72 hours, and

c. Determine, from Action b above, that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. The Reactor Coolant System temperature and pressure shall be determined to be within the limits specified in the PTLR at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line specified in the PTLR within 15 minutes prior to achieving reactor criticality.

SURVEILLANCE REQUIREMENTS (Continued)

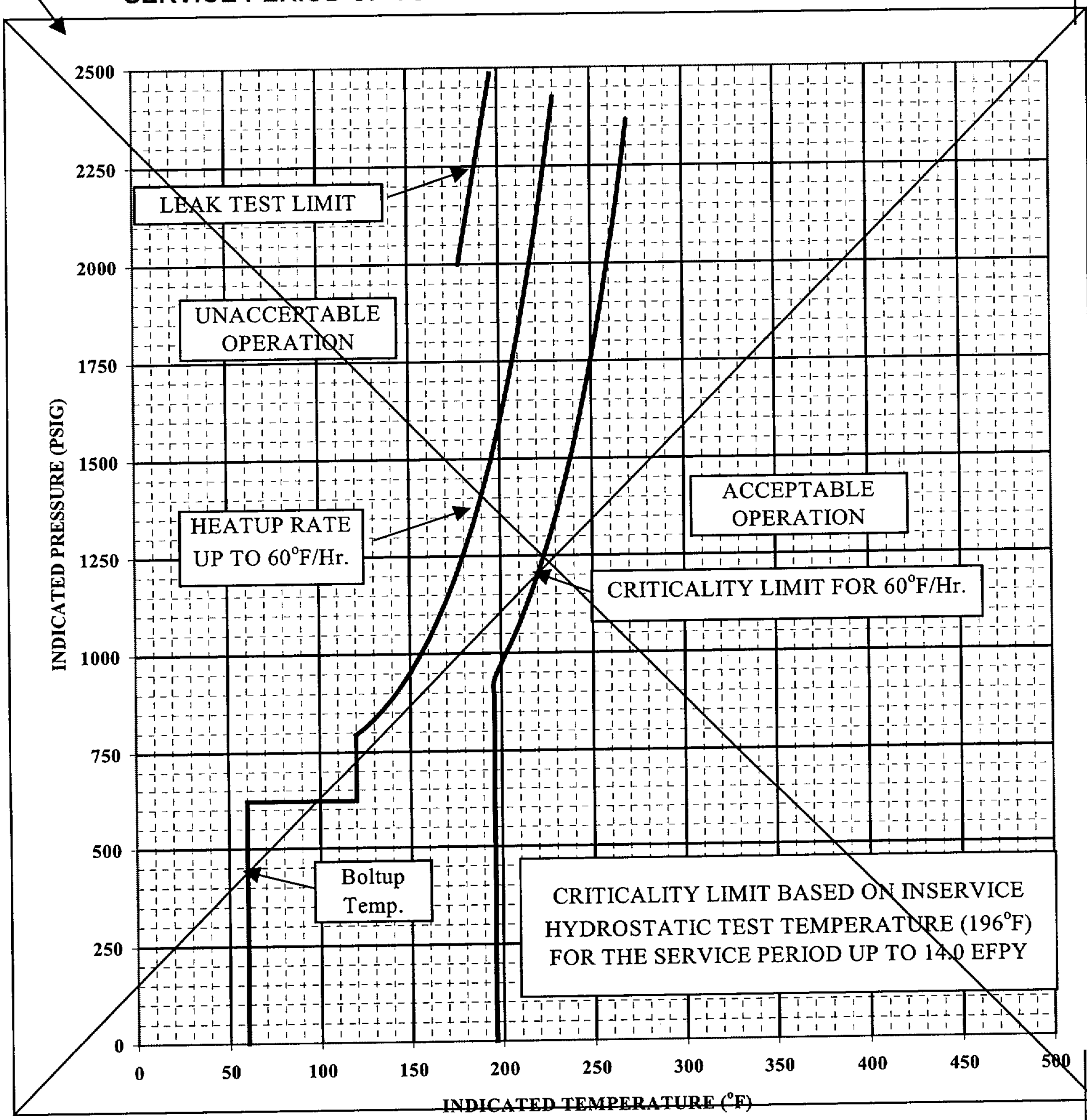
4.4.9.1 (Continued)

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined in accordance with 10 CFR 50, Appendix H, to determine changes in material properties. The results of these examinations shall be used to update the PTLR~~Figures 3.4-2 and 3.4-3~~.

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~

Moved to PTLR.

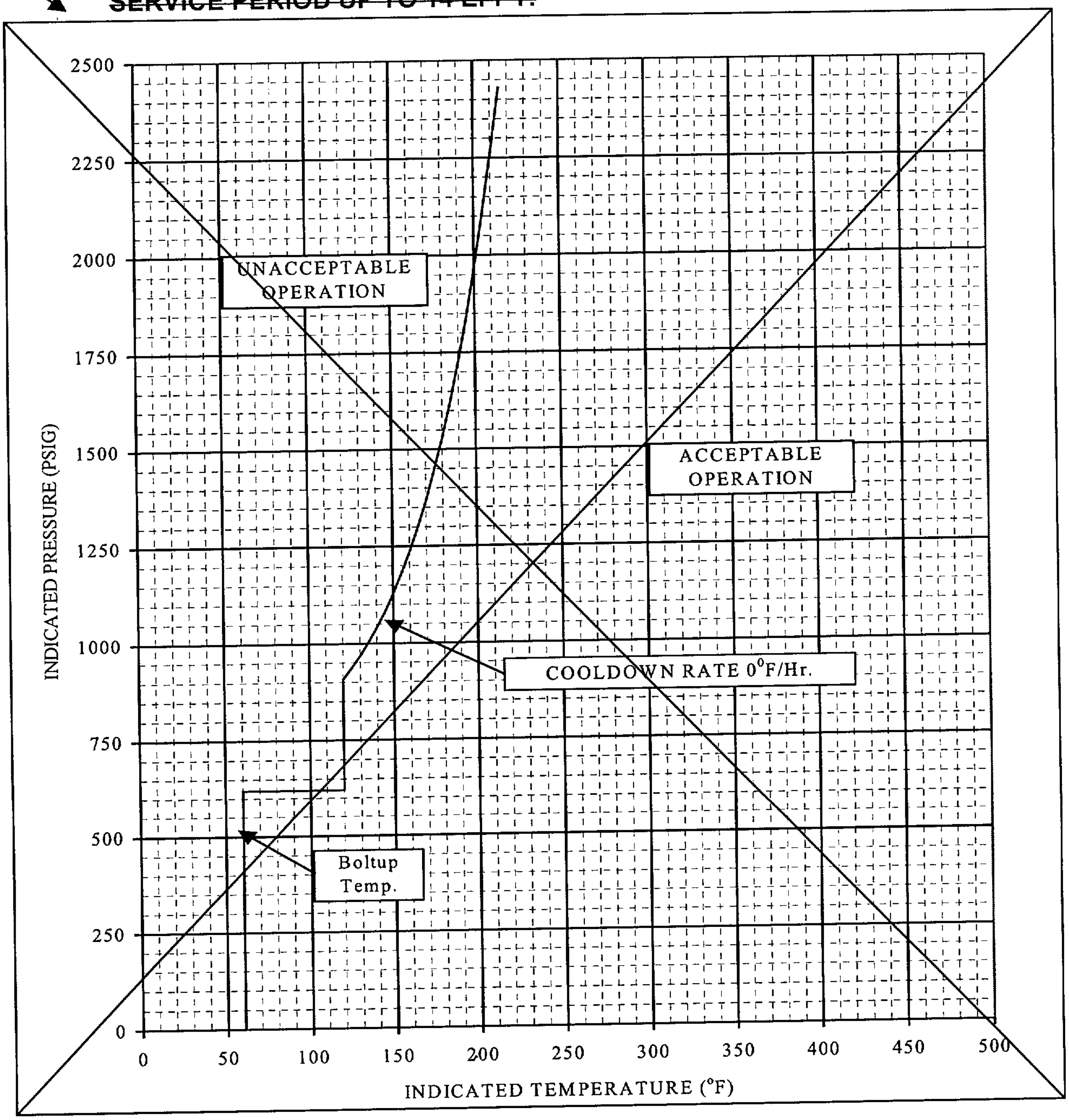


~~FIGURE 3.4-2~~
~~Beaver Valley Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~

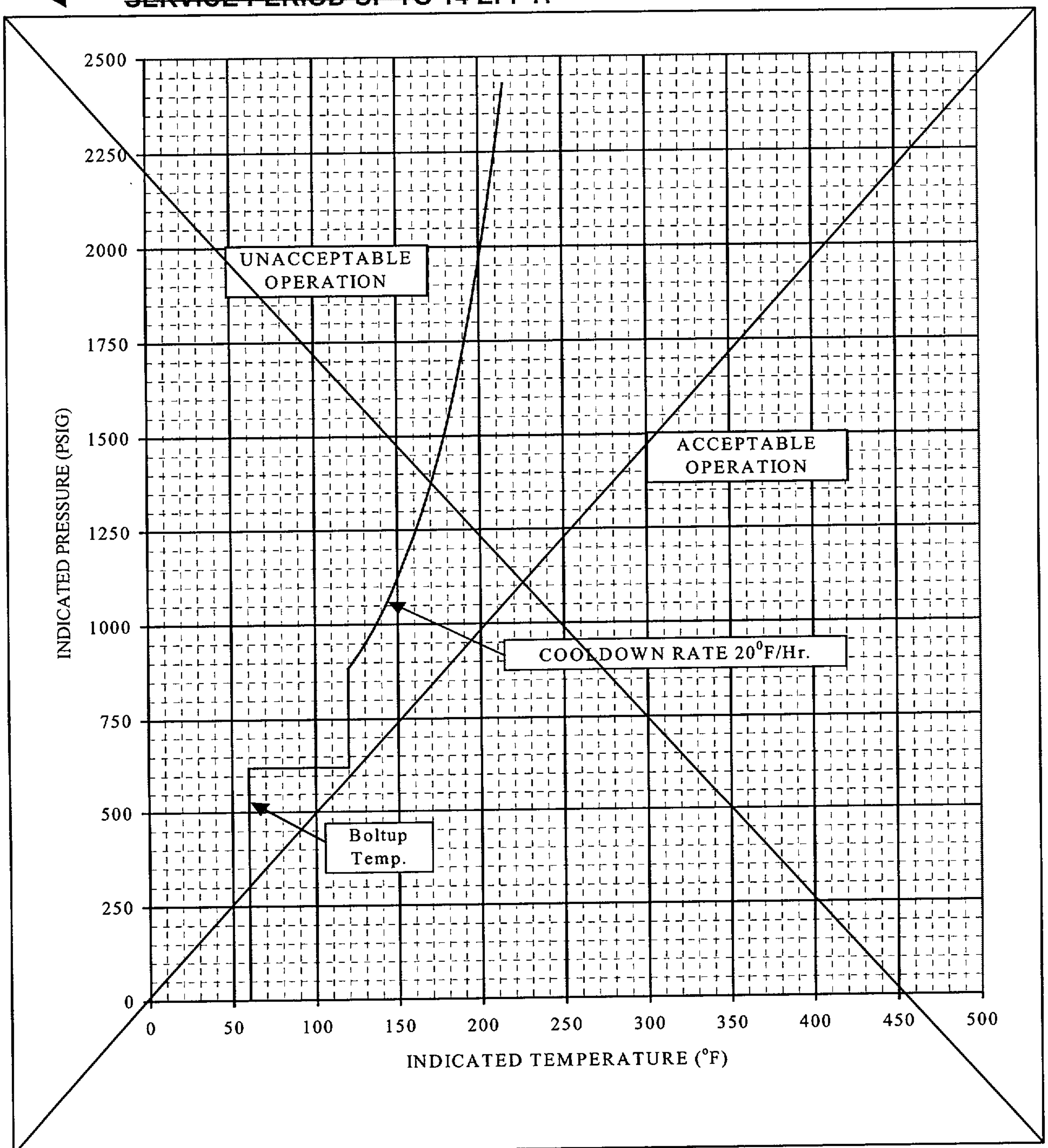


~~FIGURE 3.4-3 (Sheet 1 of 5)~~
~~Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~

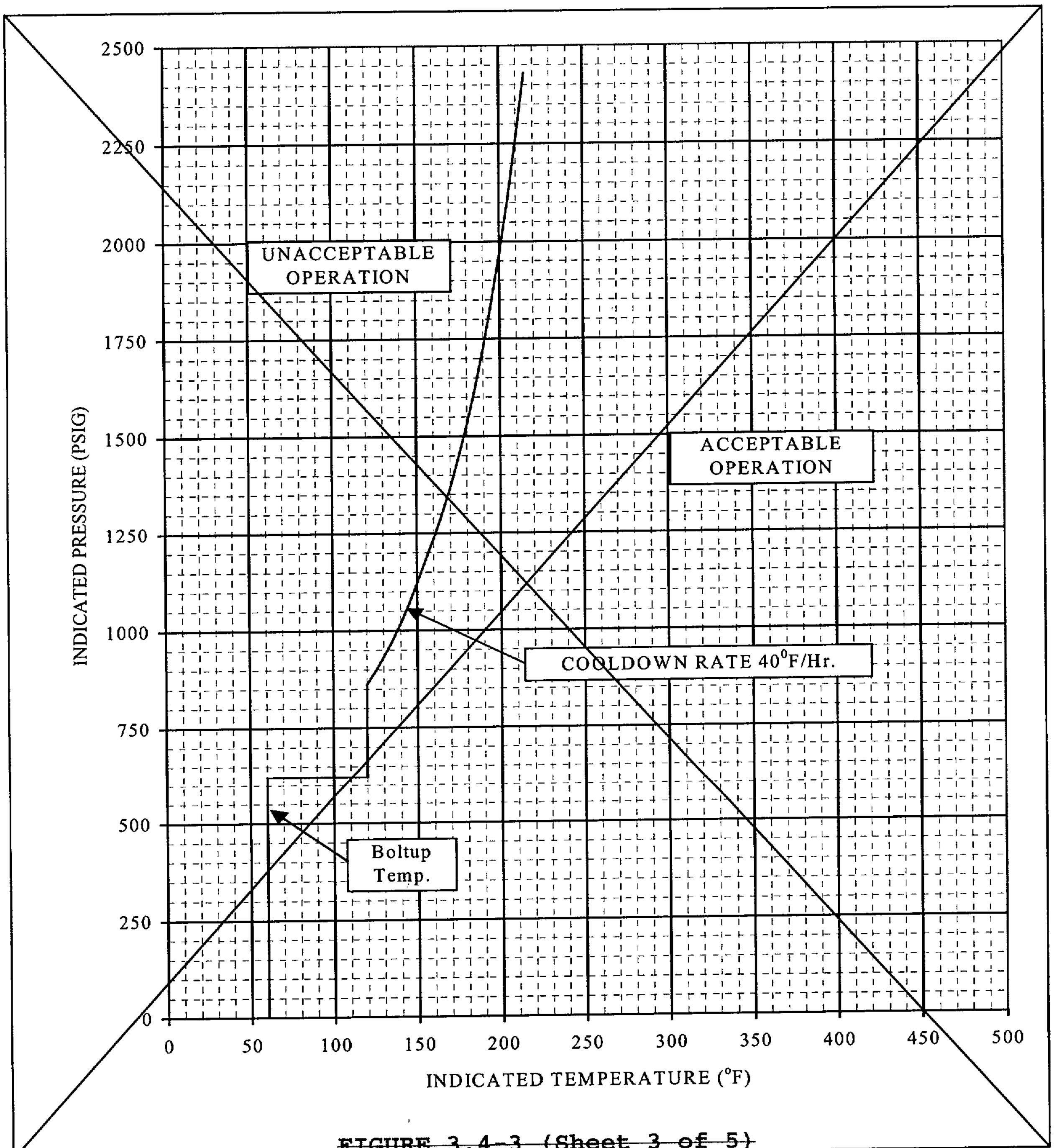


~~FIGURE 3.4-3 (Sheet 2 of 5)~~
~~Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~

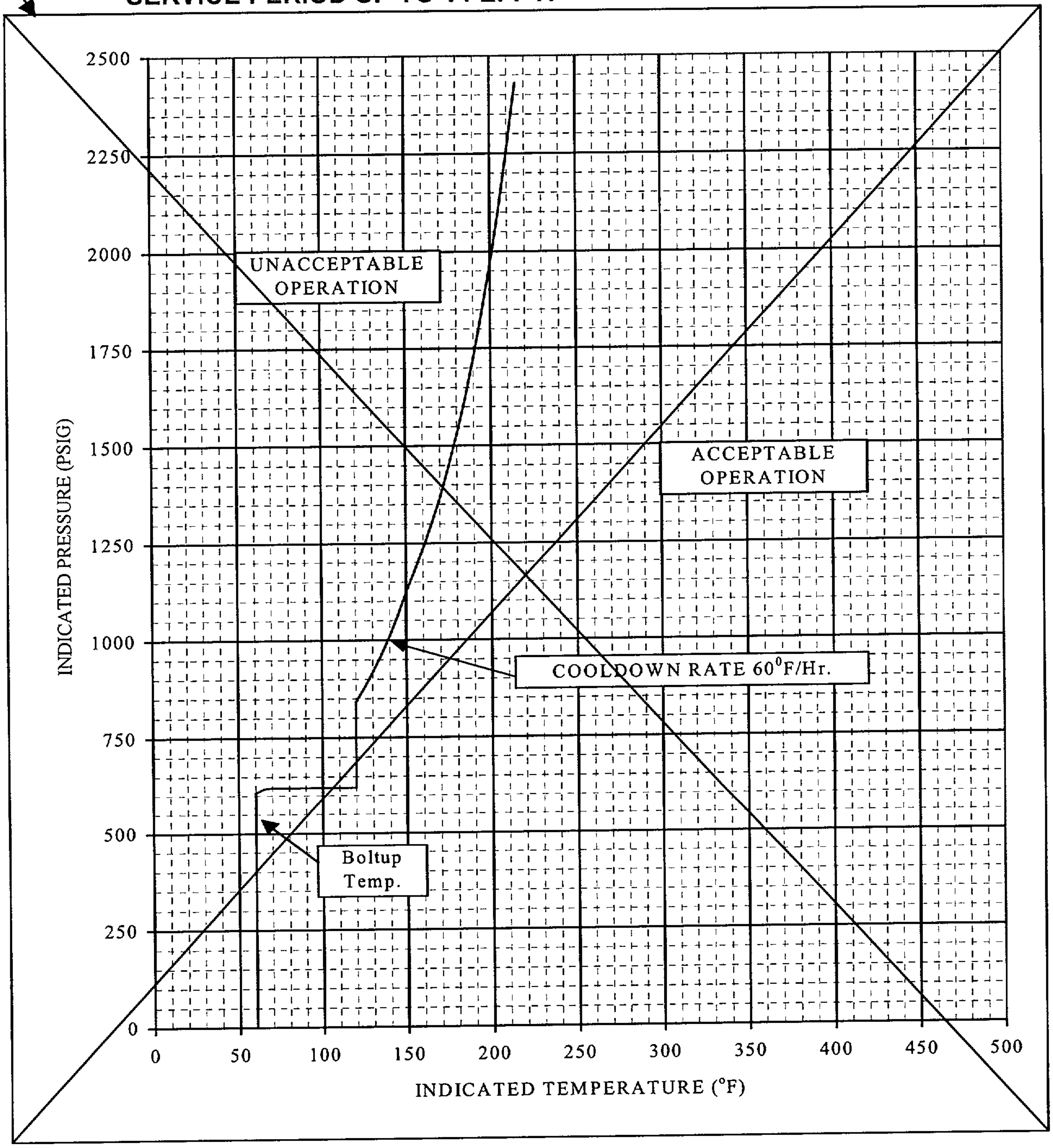


~~FIGURE 3.4-3 (Sheet 3 of 5)~~
~~Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~

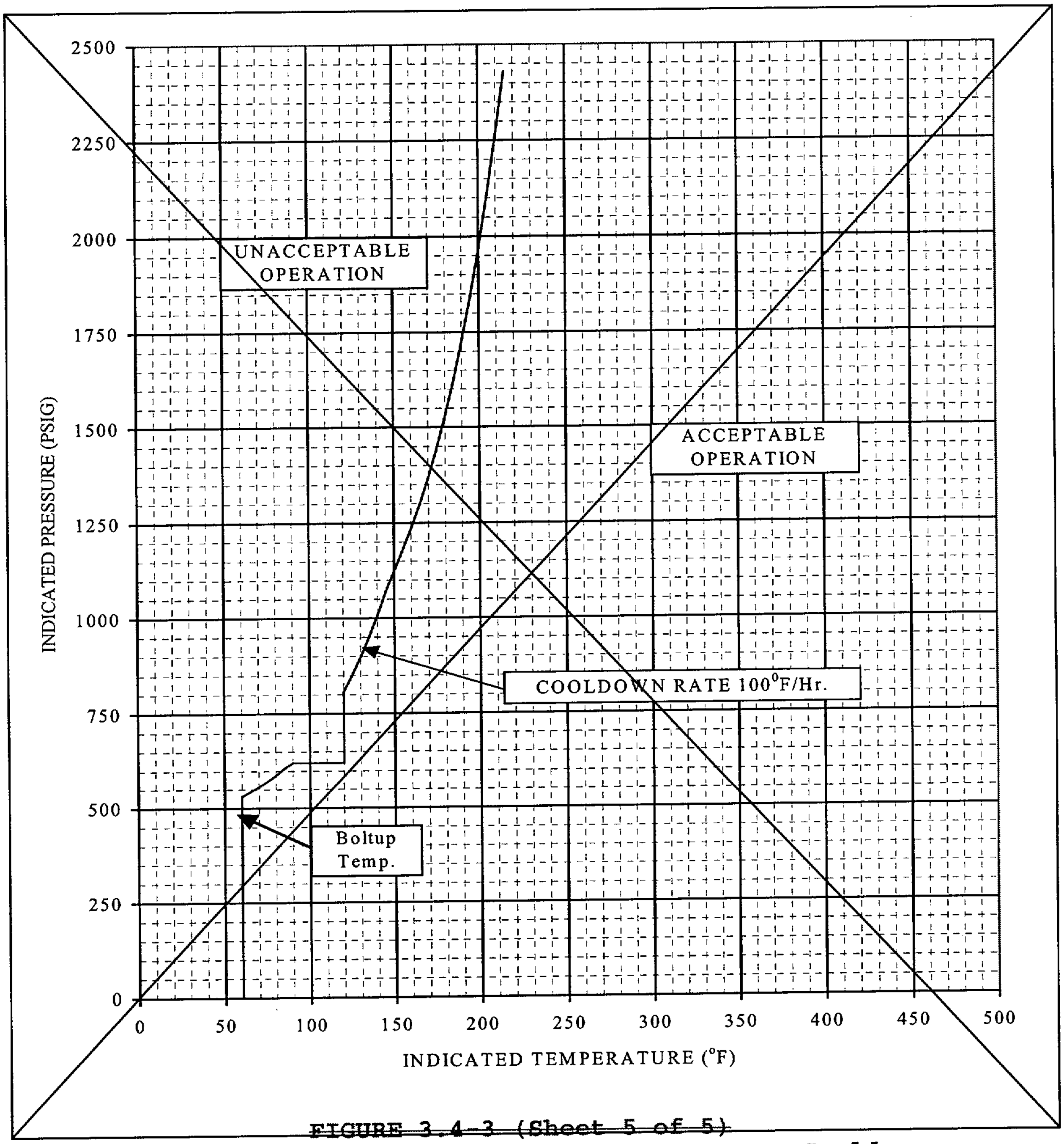


~~FIGURE 3.4-3 (Sheet 4 of 5)~~
~~Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

Moved to PTLR.

~~MATERIAL PROPERTY BASIS~~
~~CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1~~
~~INITIAL RT_{NDT}: 60°F~~
~~RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F~~
~~3/4T, 128°F~~
~~CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.~~



~~FIGURE 3.4-3 (Sheet 5 of 5)~~
~~Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 14 EFPY~~

(Proposed Wording)

3.4.9.3 An overpressure protection system shall be OPERABLE with a maximum of one charging pump(1) capable of injecting into the RCS and the accumulators isolated(2) and either a or b below:

a. Two power-operated relief valves (PORVs) with nominal maximum lift settings which vary with the RCS temperature and which do not ~~exceed the~~ limits specified in the PTLR~~established in Figure 3.4-4~~, or

b. The RCS depressurized and an RCS vent of greater than or equal to 3.14 square inches.

Moved to PTLR.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is less than or equal to an enable temperature specified in the PTLR~~of 350°F~~,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTION:

a. With two or more charging pumps capable of injecting into the RCS, immediately initiate action to verify a maximum of one charging pump is capable of injecting into the RCS or depressurize and vent the RCS through a 3.14 square inch or larger vent within 12 hours.

b. With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves in the PTLR, isolate the affected accumulator within 1 hour or increase the RCS cold leg temperature above the enable temperature specified in the PTLR within the next 12 hours or depressurize the affected accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed by the heatup and cooldown curves in the PTLR within the next 12 hours.

---

(1) Two charging pumps may be capable of injecting into the RCS for pump swap operation for less than or equal to 15 minutes. All charging pumps may be capable of injecting into the RCS for less than or equal to 4 hours immediately following a change from MODE 3 to MODE 4 or prior to the temperature of one or more of the RCS cold legs decreasing below the enable temperature specified in the PTLR minus 25°F~~325°F~~, whichever comes first.

(2) Accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves provided in the PTLR.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

c. With one PORV inoperable in MODE 4 (when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 3.14 square inch or larger vent within the next 12 hours. The provisions of Specification 3.0.4 are not applicable when in this action.

d. With one PORV inoperable in MODES 5 or 6, restore the inoperable PORV to OPERABLE status within 24 hours or depressurize and vent the RCS through a 3.14 square inch or larger vent within the next 12 hours.

e. With two PORVs inoperable, depressurize and vent the RCS through a 3.14 square inch or larger vent within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Verify at least once per 12 hours that:

a. A maximum of one charging pump is capable of injecting into the RCS, and

b. Each accumulator is isolated; however, with the accumulator pressure less than the low temperature overpressure protection setpoint, the accumulator discharge isolation valves may be opened to perform accumulator discharge check valve testing.

4.4.9.3.2 When PORVs are being used for overpressure protection, demonstrate each PORV is OPERABLE by:

a. Verifying each PORV block valve is open for each required PORV at least once per 72 hours, and

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and placed in operation after decreasing the RCS cold leg temperature to less than or equal to the enable temperature specified in the PTLR and at least once per 31 days, and

b. Performance of a CHANNEL CALIBRATION on each required PORV actuation channel at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.3.3 When a vent is being used for overpressure protection, verify the required vent is open:

a. At least once per 12 hours for an open vent or unlocked open vent valve(s), except

b. At least once per 31 days for a valve which is locked, or provided with remote position indication, or sealed, or otherwise secured in the open position.

Moved to PTLR.

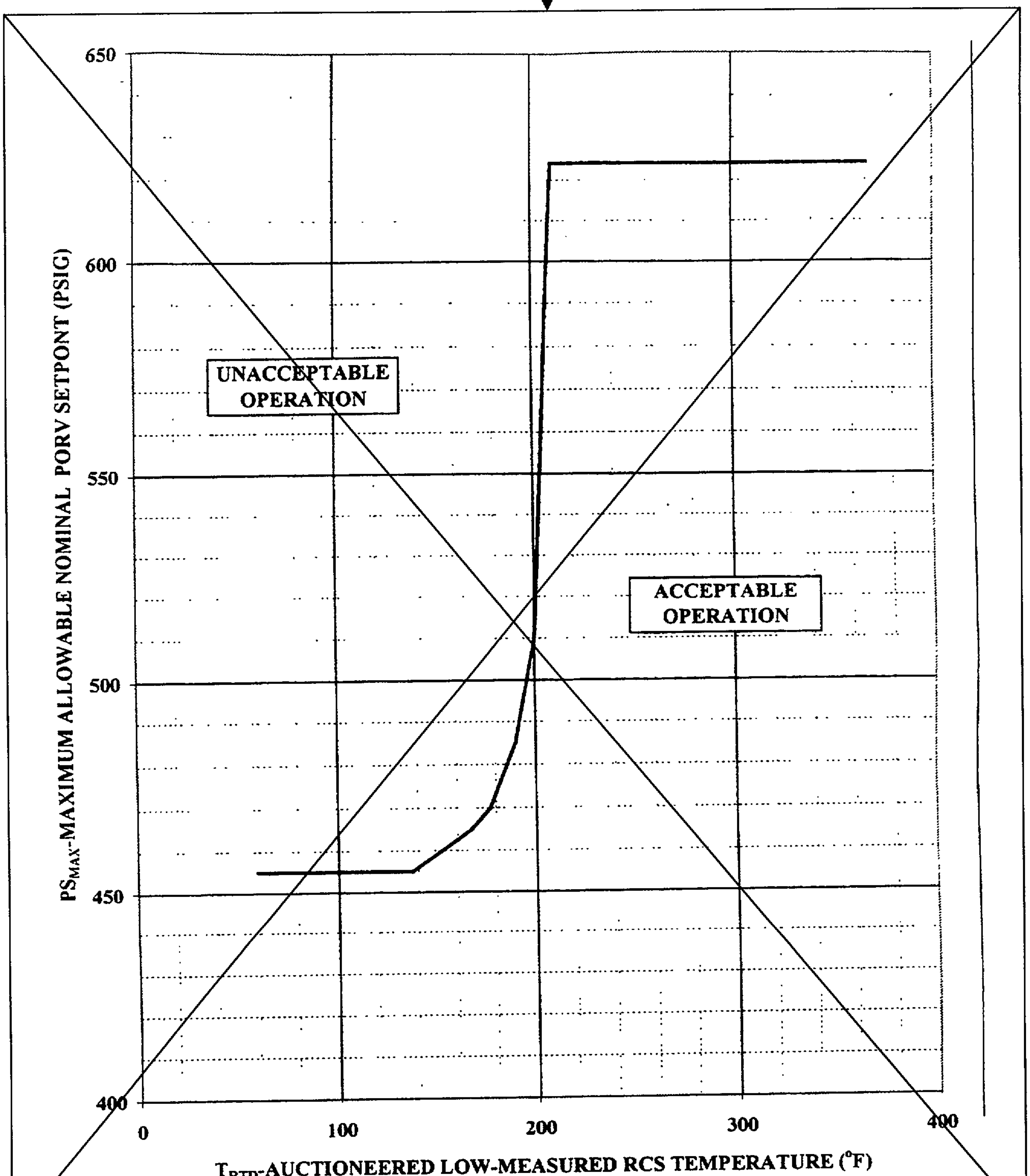


FIGURE 3.4-4

~~MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT~~
~~FOR THE OVERPRESSURE PROTECTION SYSTEM~~

3/4.5.2 ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two separate and independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump,

c. One OPERABLE recirculation spray pump(1) capable of supplying the safety injection flow path during recirculation phase, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.(2)

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted in accordance with 10 CFR 50.4 within 30 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a.1. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operator control circuits disconnected by removal of the plug in the lock out circuit from each circuit:

---

(1) Recirculation spray pump 2RSS-P21C or 2RSS-P21D.

(2) The provisions of Specifications 3.0.4 and 4.0.4 are not applicable for entry into MODE 3 for the centrifugal charging pumps declared inoperable pursuant to Specification 4.5.3.2 provided the centrifugal charging pumps are restored to OPERABLE status within 4 hours or prior to the temperature of one or more of the RCS cold legs exceeding the enable temperature specified in the PTLR plus 25°F~~375°F~~, whichever comes first.

ECCS SUBSYSTEMS - T_{avg} < 350°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE Low Head Safety Injection Pump, and

c. One OPERABLE recirculation spray pump* capable of supplying the safety injection flow path during recirculation phase, and

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

APPLICABILITY: MODE 4.

ACTION:

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

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted in accordance with 10 CFR 50.4 within 30 days describing the circumstances of the actuation and the total accumulated actuation cycle to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 All charging pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable** by verifying that the control switches are placed in the PULL-TO-LOCK position and tagged within 4 hours after entering MODE 4 from MODE 3 prior to the temperature of one or more of the RCS cold legs decreasing below the enable temperature specified in the PTLR minus 25°F~~325°F~~, whichever comes first, and at least once per 12 hours thereafter.

* Recirculation spray pump 2RSS-P21C or 2RSS-P21D.

** An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

REPORTING REQUIREMENTS (Continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

6.9.6 PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. Reactor Coolant System pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, hydrostatic testing, Overpressure Protection System (OPPS) enable temperature, and Power Operated Relief Valve (PORV) lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Specification 3.4.9.1, "Reactor Coolant System Pressure/Temperature Limits", and

   2. Specification 3.4.9.3, "Reactor Coolant System Overpressure Protection Systems".

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

   1. The analytical methods used to determine the RCS pressure and temperature limits were developed in accordance with WCAP-14040-NP-A, Rev. 2, and

   2. the OPPS limits, i.e., PORV pressure relief setpoint and OPPS enable temperature, were developed in accordance with WCAP-14040-NP-A, Rev. 2.

   The methodology listed in WCAP-14040-NP-A was used with two exceptions:

   a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

   b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring

ATTACHMENT B

Beaver Valley Power Station, Unit Nos. 1 and 2
License Amendment Request No. 295 and 167
Creation of a Pressure and Temperature Limits Report

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed license amendment is applicable to Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. The proposed changes consist of creating a Pressure and Temperature Limits Report (PTLR) for both BVPS units. A PTLR for each unit will be incorporated into each unit's Licensing Requirements Manual (LRM).

Creation of a PTLR is consistent with the guidance provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits". The Pressure/Temperature (P/T) limits contained in the proposed PTLR have been prepared using the NRC-approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Revision 2. The proposed changes to the Technical Specifications are consistent with the content of NUREG-1431, "Standard Technical Specifications Westinghouse Plants", Revision 2.

The proposed PTLR is provided in Attachments C-1 and C-2 for Units 1 and 2, respectively. The changes proposed to the Technical Specification Bases are provided in Attachments D-1 and D-2 for Units 1 and 2, respectively. The proposed Technical Specification Bases changes do not require NRC approval. The BVPS Technical Specification Bases Control Program controls the review, approval and implementation of Technical Specification Bases changes. They are provided for information only.

The proposed changes to the Technical Specifications and Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

The proposed changes are described below.

1. Revise the Index to include "Pressure and Temperature Limits Report" in sections 1.0 and 6.9 for both units.

2. Revise the Definition Section to include the "Pressure and Temperature Limits Report" for both units.

3. Relocate limits from Technical Specification 3.4.9.1, "Reactor Coolant System - Pressure/Temperature Limits" to the PTLR for both units.

4. Relocate the heatup/cooldown curves, Figures 3.4-2 and 3.4-3, to the PTLR for both units.

5. Relocate the Overpressure Protection System (OPPS) Power Operated Relief Valve (PORV) setting and OPPS enable temperature from Technical Specification 3.4.9.3, "Overpressure Protection Systems" to the PTLR for both units.

6. Relocate Table B 3/4.4-1 to the PTLR for both units.

7. Relocate Figure B 3/4.4-2 to the PTLR for both units.

8. Revise Figure Index to show the deletion of Figures 3.4-2, 3.4-3, B 3/4.4-1 and B 3/4.4-2 for Unit 1 only.

9. Relocate Figure 3.4-4 to the PTLR for Unit 2 only.

10. For Unit 2 only, revise 375°F referenced in the following Technical Specifications to "the enable temperature specified in the PTLR plus 25°F".

    3/4.1.2.2 Reactivity Control Systems – Flow Paths – Operating,
    3/4.1.2.4 Reactivity Control Systems – Charging Pumps – Operating, and
    3/4.5.2 ECCS Subsystems – $T_{avg} \geq 350°F$.

11. For Unit 2 only, revise 325°F referenced in the following Technical Specifications to "the enable temperature specified in the PTLR minus 25°F".

    3/4.4.9.3 Overpressure Protection Systems" and
    3/4.5.3 ECCS Subsystems – Tavg < 350°F",

12. Revise the following Technical Specifications and associated Bases to refer to the PTLR.

    For Unit 1 the applicable Technical Specifications are:

    3.1.2.4, Reactivity Control Systems – Charging Pumps – Operating,
    3.4.1.3, Reactor Coolant System – Shutdown,
    3.4.9.1, Reactor Coolant System - Pressure/Temperature Limits,
    3.4.9.3, Overpressure Protection Systems,
    3.5.3, ECCS Subsystems – $T_{avg} < 350°F$, and
    3.5.4.1.2, Boron Injection Tank < 350°F,
    3.10.3, Special Test Exceptions - Pressure/Temperature Limitations - Reactor Criticality.

    For Unit 2 the applicable Technical Specifications are:

    3.1.2.2, Reactivity Control Systems – Flow Paths – Operating,
    3.1.2.4, Reactivity Control Systems – Charging Pumps – Operating,
    3.4.1.3, Reactor Coolant System – Shutdown,
    3.4.9.1, Reactor Coolant System - Pressure/Temperature Limits,
    3.4.9.3, Overpressure Protection Systems,
    3.5.2, ECCS Subsystems – $T_{avg} \geq 350°F$, and
    3.5.3, ECCS Subsystems – Tavg < 350°F.

13. Add Technical Specification 6.9.6, which provides the reporting requirements associated with the PTLR, to the Administrative Controls Section of the Technical Specifications for both units.

Several of the pages affected by this license amendment contain changes that have been previously submitted for approval as other license amendment requests (LARs). Since approval of the previously submitted LARs is expected prior to the approval of this request, the pages affected by this request include those previously submitted changes that are germane to this request. The cover page of Attachments A-1 and A-2 list the pages affected by this license amendment for each unit. The applicable LAR number identifies the pages being changed by other LARs. The previously submitted LAR for Unit 1 is 292. The previously submitted LAR for Unit 2 is 157.

LAR 292 for Unit 1 was submitted by FENOC letter L-01-087, dated June 29, 2001. The changes germane to this request are changes to the Unit 1 heatup/cooldown curves, the PORV setpoint and the OPPS enable temperature to reflect 22 Effective Full Power Years (EFPY), the methodology of WCAP-14040,

Rev.2, and the applicability of Code Case N-640. The applicable pages for this request therefore reflect these proposed changes.

LAR 157 for Unit 2 was submitted by FENOC letter L-00-131, dated November 8, 2000. The changes germane to this is request are the deletion of Technical Specification 3.4.1.6, "Reactor Coolant Startup" and the addition a note to Technical Specification 3.4.1.3, Reactor Coolant System – Shutdown", addressing restrictions imposed on starting an idle Reactor Coolant Pump. The applicable pages for this request therefore reflect these proposed changes.

The capsule withdrawal schedule is provided in each unit's Updated Final Safety Analysis Report (UFSAR), i.e., Table 4.5-3 for Unit 1 and Table 5.3-6 for Unit 2. These tables will remain in the UFSAR. They will be updated following NRC approval of the schedule changes proposed in the applicable analysis WCAP.

To meet format requirements the Index and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

B. DESIGN BASES

Generic Letter 96-03 requires that the P/T limits are generated in accordance with the requirements of 10 CFR 50, Appendix G, documented in an NRC-approved topical report incorporated by reference in the Technical Specifications. Accordingly, the Beaver Valley Power Station (BVPS) heatup/cooldown curves have been generated using the NRC-approved methods described in WCAP-14040-NP-A, Revision 2, and meet the requirements of 10 CFR 50, Appendix G with the exceptions noted in Technical Specification Section 6.9.6. The use of Code Case N-640 results in an increase in the safety of operating plants, as the likelihood of pump seal failure will decrease.

Technical Specifications 3.4.9.1 and 3.4.9.3 will continue to require that the RCS P/T, OPPS limit, and the enable temperature be limited in accordance with the limits specified in the PTLR. The NRC-approved methodology for generating the P/T limits, WCAP-14040-NP-A, Revision 2, will be specified in Technical Specification 6.9.6 and NRC approval will be required in the form of a Technical Specification Amendment prior to changing the methodology. Use of P/T limit curves generated using the NRC-approved methods described in WCAP-14040-NP-A, Revision 2, as specified by Technical Specification 6.9.6, will provide sufficient protection for the integrity of the reactor vessel, thereby assuring that the reactor vessel is capable of providing its function as part of the radiological barrier.

The P/T limits, OPPS setpoint, and the enable temperature proposed for inclusion in the PTLR are based on fluence that bound the 2689 MW thermal power, 1.4% uprated power level, and operation through 22 EFPY for Unit 1 and 14 EFPY for Unit 2.

C. JUSTIFICATION

During the development of the improved standard technical specifications (ISTS), a change was proposed to relocate the P/T curves and the OPPS setpoint currently contained in the Technical Specifications to a Licensee-controlled document. The NRC agreed with the industry that the P/T curves and the OPPS setpoint may be relocated outside of the Technical Specifications to a licensee-controlled document so that the licensee could maintain these limits efficiently and at a lower cost, provided that the parameters for constructing the curves and setpoint are derived using an NRC-approved methodology. Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits", provided guidance to licensees for implementing this line item Technical Specification improvement. Those guidance are used to relocate the subject parameters to ensure that NRC approved methodology will be continued to be used to generate the relocated limits.

Attachment E is provided as an aid in the review of the changes proposed in this license amendment request. The attachment contains a duplication of the table appearing in Attachment 1 of GL 96-03, with additional columns that identify the location of each requirement in the BVPS Unit 1 and Unit 2 PTLRs.

The specific enable temperature for the limiting RCS cold leg temperature, below which the reactor vessel may suffer damage from a cold overpressure event, is reactor vessel plant specific and varies with vessel fluence. Use of a specific value, which will require periodic amendments, is not consistent with the PTLR philosophy and is therefore proposed to be relocated to the PTLR. Reference to the PTLR for other plant specific parameters (e.g., Technical Specification 3.4.9.1) are acceptable, and result in simplifying the revision process when those values change with reactor fluence. Periodic updates to the vessel limiting temperature can be accommodated without going through the license amendment process. The methodology used to determine the limiting temperature is controlled by Technical Specifications and requires NRC approval for changes. Relocating this value to the PTLR is consistent with the Westinghouse PTLR methodology Topical Report, WCAP 14040-NP-A, Rev. 2.

Several Technical Specifications for Unit 2 specify a temperature limit (325°F or 375°F) near the OPPS enable temperature that addresses charging pump operability during the transition from Mode 3 to Mode 4. These Technical Specifications are:

3.1.2.2	Reactivity Control Systems – Flow Paths – Operating,
3.1.2.4	Reactivity Control Systems – Charging Pumps – Operating,
3.4.9.3	Overpressure Protection Systems,
3.5.2	ECCS Subsystems – $T_{avg} \geq 350°F$, and
3.5.3	ECCS Subsystems – Tavg < 350°F.

For those Technical Specifications that specify 325°F, the value is replaced with "the enable temperature specified in the PTLR minus 25°F". For those Technical Specifications that specify 375°F, the value is replaced with "the enable temperature specified in the PTLR plus 25°F". This replacement of a specific value with a temperature range near the enable temperature is consistent with specification 3.5.2, "ECCS - Operating", of NUREG-1431. As stated in Bases B 3.5.2 of NUREG-1431, when the OPPS enable temperature is near the Mode 3 boundary temperature of 350°F, time is needed to make the charging pumps incapable of injection prior to entering the Applicability of Technical Specification 3.4.9.3. Time is also needed to restore inoperable charging pumps to operable status on exiting the Applicability of Technical Specification 3.4.9.3. The proposed change provides a range equivalent to the stated temperatures when applied to the enable temperature appearing in the PTLR.

D. SAFETY ANALYSIS

Technical specifications include limiting conditions for operation (LCOs) that establish P/T and OPPS limits for the reactor coolant system. The limits are defined by figures and values that provide an acceptable range of operating temperatures and pressures for heatup, cooldown, low temperature overpressure, criticality, and inservice leak and hydrostatic testing conditions. These parameters are generally valid for a specified number of effective full-power years or for a specified period. License amendments are generally required at the end of the effective period for P/T limit curves or when surveillance specimens are withdrawn and tested. Also, each time the P/T curves are revised, the OPPS must be reevaluated to ensure that its functional requirements can still be met. Processing amendments request for changes to Technical Specification that are developed using an accepted methodology places an unnecessary burden on licensee and NRC resources. An alternative approach for controlling these limits was proposed during the development of the ISTS. This approach, like the one used for the core

operating limits report, would relocate the P/T curves and OPPS setpoint value to a PTLR and would reference that document in the affected LCOs and Technical Specification Bases. The guidance contained in GL 96-03 specifically requires licensees wishing to implement this line item Technical Specification improvement to:

(1) reference a methodology for developing the curves and setpoint that has been approved by the NRC;

(2) develop a PTLR or a similar document that contains the figures, values, parameters, and any explanations derived from the methodology; and

(3) make appropriate changes to the applicable sections of the Technical Specifications.

The following provides a description of the BVPS compliance with the listed requirements of GL 96-03:

(1) The P/T limits contained in the proposed PTLR (applicable through 22 EFPY for Unit 1 and 14 EFPY for Unit 2) were generated in accordance with the methods described in WCAP-14040-NP-A, Revision 2, consistent with the requirements of 10 CFR 50 Appendix G, and Regulatory Guide 1.99, Revision 2, with the exceptions noted in Technical Specification Section 6.9.6. The use of Code Case N-640 results in an increase in the safety of operating plants, as the likelihood of pump seal failure will decrease. Additionally, the proposed OPPS limit has been adjusted to account for the static and dynamic pressure differential between the reactor vessel beltline and the RCS wide range pressure transmitter which is used to provide overpressure protection for the RCS.

The NRC has reviewed the methods described in WCAP-14040-NP-A, and approved the topical report by issuance of a Safety Evaluation Report (SER) dated October 16, 1995. The NRC concluded in its SER that WCAP-14040, Revision 1, satisfies the provisions described in a draft generic letter published in the Federal Register (60 FR 28805) for public comment on June 2, 1995, which was subsequently issued as GL 96-03, January 31, 1996. Revision 2 to WCAP- 14040-NP-A simply incorporates the Westinghouse Owners Group response to NRC comments on Revision l; incorporates the NRC SER approving WCAP-14040-NP-A, Revision 1; and adds the suffix NP-A to the report number to designate NRC approval of the report.

(2) The proposed PTLRs for BVPS Units 1 and 2 meet the requirements contained in GL 96-03 and are included as Attachments C-1 and C-2.

FENOC has evaluated the ability of the PORVs to provide low temperature protection based on the P/T limit curves and determined that the PORVs provide adequate relief capability to prevent the RCS pressure from exceeding the 10 CFR 50, Appendix G, steady-state limit during the worst-case heat or mass input transient at RCS temperatures. For Unit 1 the evaluation is documented in LAR 292. For Unit 2 the evaluation was approved by issuance of Amendment 69.

WCAP-14040-NP-A, Rev. 2, includes the method for determination of the OPPS enable temperature. For Unit 1 the evaluation is documented in LAR 292. For Unit 2 the evaluation was approved by issuance of Amendment 69.

Generic Letter 96-03 also requires that licensees address the minimum boltup temperature for the reactor vessel head and closure flange. Consistent with the methods described in WCAP-14040-NP-A, Revision 2, the minimum boltup temperature is 60°F for Unit 1 and Unit 2. For Unit 1 the evaluation is documented in LAR 292. For Unit 2 the evaluation was approved by issuance of Amendment 113.

(3) Consistent with the guidance provided in GL 96-03, FENOC provides the proposed Technical Specification changes associated with the PTLR as Attachments A-1 and A-2.

Based on item (1), (2), and (3) above, the proposed PTLR for BVPS Unit 1 and Unit 2, and the proposed changes to the Technical Specifications, meet the requirements of GL 96-03 and are consistent with the content of NUREG-1431. The BVPS P/T limits will be generated in accordance with the NRC-approved methodology described in WCAP-14040-NP-A, Revision 2 and those exceptions noted in Technical Specification Section 6.9.6. The plant will continue to be operated in accordance with the RCS P/T limits, OPPS limit, and the enable temperature as required by Technical Specifications 3 4.9.1 and 3.4.9.3. Therefore, BVPS will continue to meet the requirements of 10 CFR 50, Appendix G with the exemption identified in item (1) above, thus assuring that the integrity of the reactor vessel will be maintained.

E. NO SIGNIFICANT HAZARDS EVALUATION

The changes being proposed for the Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2 Technical Specifications consist of creating a Pressure and Temperature Limits Report (PTLR) for both BVPS units. Creation of a PTLR is consistent with the guidance provided in Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits". The Pressure/Temperature (P/T) limits contained in the proposed PTLR have been prepared using the NRC-approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Revision 2. The proposed changes to the Technical Specifications are consistent with the content of NUREG-1431, "Standard Technical Specifications Westinghouse Plants", Revision 2.

To meet format requirements the Index and Bases pages are also revised and repaginated as necessary to reflect the changes being proposed.

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focused 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?

   No. The proposed changes are a relocation of the Reactor Coolant System (RCS) pressure/temperature (P/T) limits, overpressure protection system (OPPS) setpoint, and the enable temperature from the Technical Specifications to the proposed Pressure and Temperature Limits Report (PTLR). The PTLR is created in accordance with the guidance provided by Generic Letter (GL) 96-03 and is consistent with the content of NUREG-1431. The RCS P/T limits, OPPS setpoint, and enable temperature will continue to meet the requirements of 10 CFR 50, Appendix G, and will be generated in accordance with the NRC approved methodology described in WCAP-14040-NP-A, Rev. 2 with the exceptions noted in Technical Specification Section 6.9.6.

   Since the proposed changes are administrative in nature and do not involve any change to any values being relocated, the proposed changes do not involve a significant increase in 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?

   No. As stated above, the proposed changes to relocate the RCS P/T limits, OPPS setpoint, and the enable temperature from the Technical Specifications to the PTLR are administrative changes. The proposed changes do not result in a physical change to the plant or add any new or different operating requirements on plant systems, structures, or components.

   Therefore, the proposed changes do not result in a significant increase in the possibility of a new or different accident from any previously evaluated.

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

   No. The margin of safety is not affected by the creation of the proposed PTLR. Operation of the plant in accordance with the limits specified in the PTLR will continue to meet the requirements of 10 CFR 50, Appendix G, with the identified exceptions, and will assure that a margin of safety is not significantly decreased as the result of the proposed changes.

   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 satisfy the requirements of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request does not change any requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. It has been determined that this license amendment request involves no significant increase in the amounts, and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. This license amendment request does not change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement. In addition, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this license amendment request.

ATTACHMENT C-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 295

Proposed PTLR for Unit 1

BVPS-1 Technical Specification to PTLR Cross-Reference			
Technical Specification	PTLR		
	Section	Figure	Table
3.1.2.4	N/A	N/A	4.2-3
3.4.1.3	N/A	N/A	4.2-3
3.4.9.1	4.2.1.1	4.2-1 4.2-2	N/A
3.4.9.3	4.2.1.2 4.2.1.3	N/A	4.2-3
3.5.3	N/A	N/A	4.2-3
3.5.4.1.2	N/A	N/A	4.2-3
3.10.3	N/A	4.2-1 4.2-2	N/A

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

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LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

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4.2 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Unit 1 has been prepared in accordance with the requirements of Technical Specification 6.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in, or make reference to, this report are listed below:

TS 3.1.2.4 Reactivity Control Systems – Charging Pumps – Operating,

TS 3.4.1.3 Reactor Coolant System – Shutdown,

TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,

TS 3.4.9.3 Overpressure Protection Systems,

TS 3.5.3 ECCS Subsystems – $T_{avg} < 350°F$,

TS 3.5.4.1.2 Boron Injection Tank < 350°F, and

TS 3.10.3 Special Test Exceptions - Pressure/Temperature Limitations -Reactor Criticality.

4.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 1 were developed using a methodology specified in the Technical Specifications. The methodology listed in Reference 1 was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

4.2.1.1 RCS Pressure and Temperature (P/T) Limits (TS 3.4.9.1)

The RCS temperature rate-of-change limits defined in Reference 2 are:

a. A maximum heatup of 100°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period, and

c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic testing operations above system design pressure.

The RCS P/T limits for heatup, leak testing, and criticality are specified by Figure 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 and 4.2-2 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The heatup and cooldown curves also include the effect of the reactor vessel flange.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

Pressure-temperature limit curves shown in Figure 4.2-3 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640.

4.2.1.2 Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

The power operated relief valves (PORVs) shall each have maximum lift setting and enable temperature in accordance with Table 4.2-3. The lift setting provided does not impose any reactor coolant pump restrictions.

The PORV setpoint is based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1, including the exceptions noted in Section 4.2.1. The PORV lift setting shown in Table 4.2-3 accounts for appropriate instrument error.

4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

Two different temperatures are used to determine the OPPS enable temperature, they are the arming temperature and the calculated enable temperature. The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature is 343°F.

The calculated enable temperature is based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The calculated enable temperature does not address the piping limit attributed to a water hammer discharge. The calculated enable temperature is 308°F.

As the arming temperature is higher and, therefore, more conservative than the calculated enable temperature, the OPPS enable temperature, as shown in Table 4.2-3, is set to equal the arming temperature.

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature specified in Table 4.2-3, and disarming of the OPPS above this temperature. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

From a plant operations viewpoint the terms "armed" and "enabled" are synonymous when it comes to activating the OPPS. As stated in the applicable operating procedure, the OPPS is activated (armed/enabled) manually before entering the applicability of TS 3.4.9.3. This is accomplished by placing two keylock switches (one in each train) into their "automatic" position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the OPPS setpoint. This will initiate an alarm in the control room and open the OPPS PORVs.

4.2.1.4 Reactor Vessel Boltup Temperature (TS 3.4.9.1)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the reactor vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

4.2.2 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and analyzed to determine changes in material properties. The capsule withdrawal schedule is provided in Table 4.5-3 of the UFSAR. Also, the results of these analyses shall be used to update Figures 4.2-1 and 4.2-2, and Tables 4.2-1 and 4.2-2. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

The pressure vessel material surveillance program (References 3 and 4) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME, Section III, NB-2331. 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 XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E 185-82.

4.2.3 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 4.2-4, taken from Reference 5, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2-5, taken from Reference 2, provides the reactor vessel beltline material property table.

Table 4.2-6, taken from Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 22 EFPY.

Table 4.2-7, taken from Reference 2, shows the calculation of ARTs for 22 EFPY.

Table 4.2-8 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 4.2-9, taken from Reference 5, provides RT_{PTS} values for 28 EFPY.

Table 4.2-10, taken from Reference 5, provides RT_{PTS} values for 45 EFPY.

4.2.4 References

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

2. WCAP-15570, Revision 2, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," T. J. Laubham, April 2001.

3. WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," C. Brown, et. al., November 2000.

4. WCAP-8475, "Duquesne Light Company, Beaver Valley Unit No. 1 Reactor Vessel Radiation Surveillance Program," J. A. Davidson, October 1974.

5. WCAP-15569, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 1," C. Brown, et al., November 2000.

6. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

7. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)

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

9. Westinghouse Report, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule", Revision 1, April 2001.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F
3/4T, 196°F

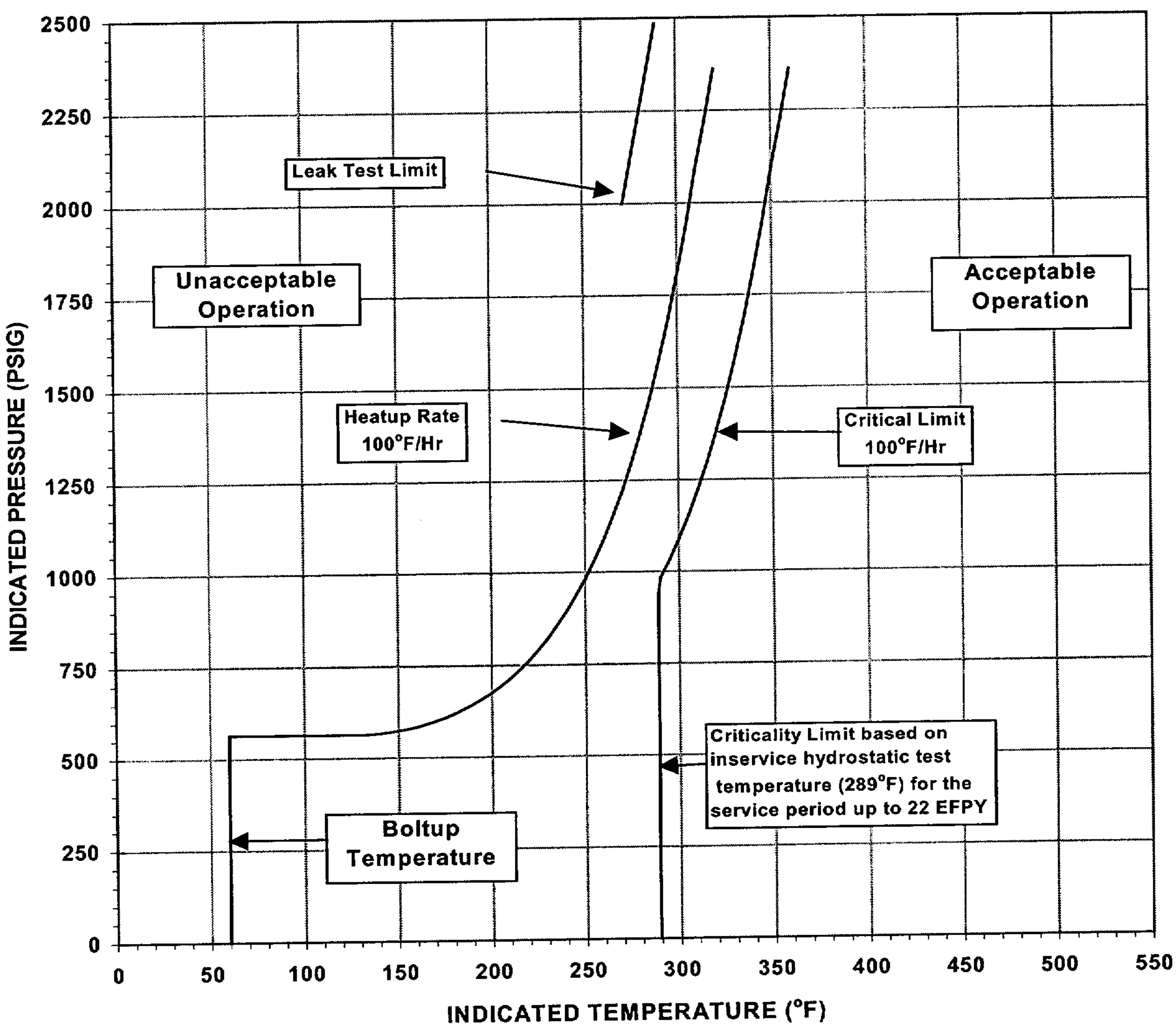

Figure 4.2-1
Reactor Coolant System Heatup
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL:	INTERMEDIATE & LOWER SHELL PLATE
LIMITING ART VALUES AT 22 EFPY:	1/4T, 233°F
	3/4T, 196°F

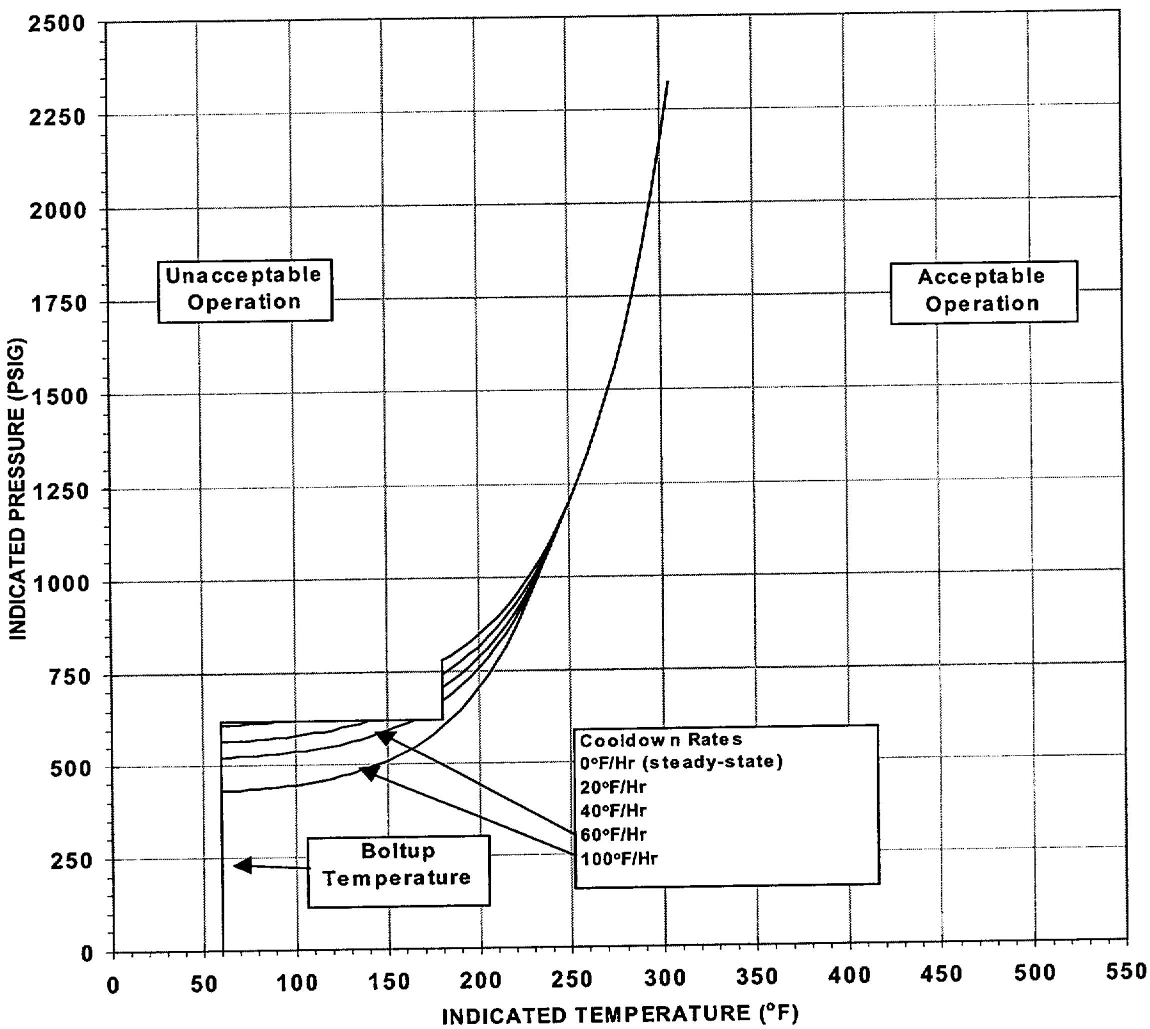

Figure 4.2-2
Reactor Coolant System Cooldown
Limitations Applicable for the First 22 EFPY (TS 3.4.9.1)

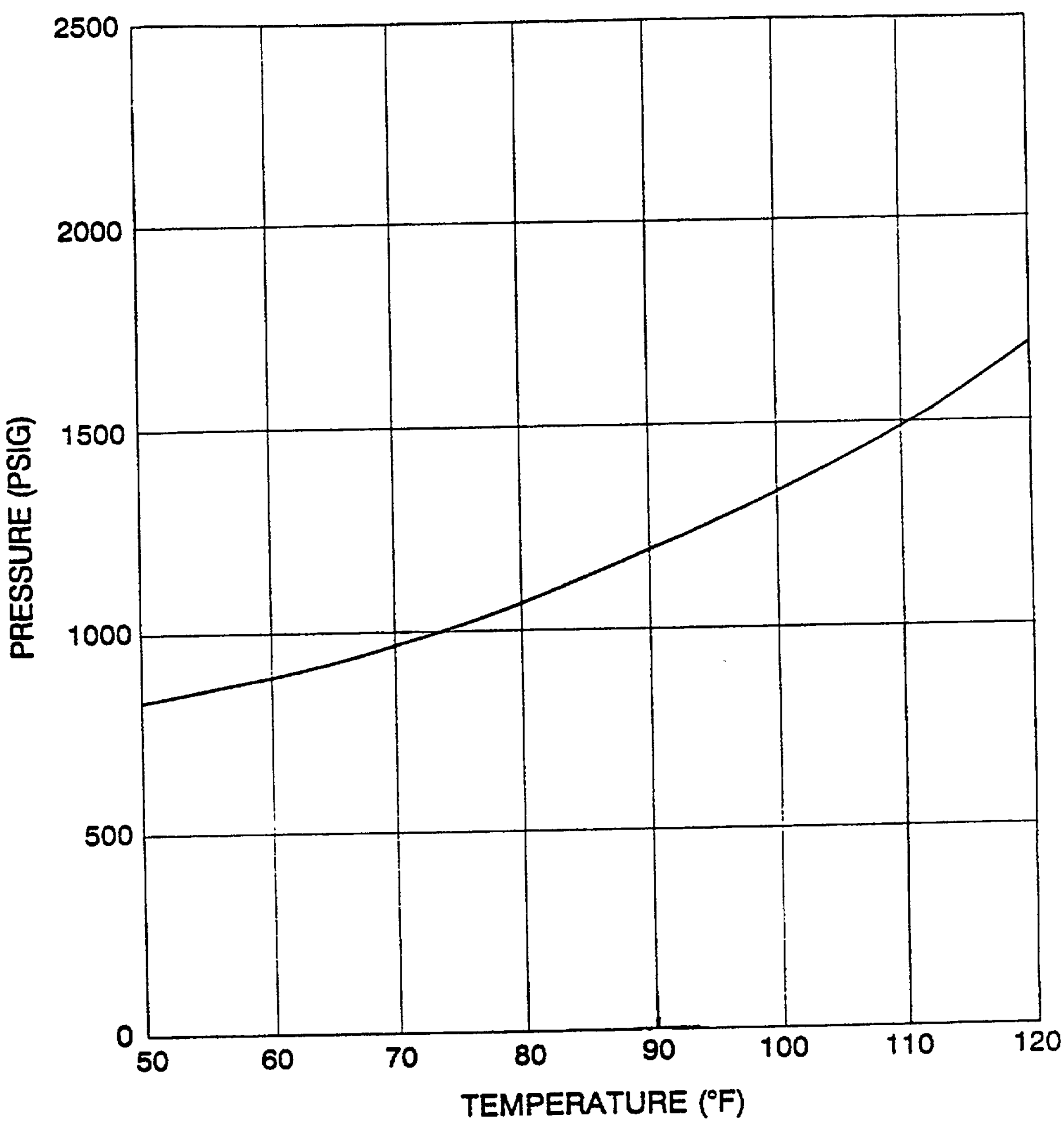

Figure 4.2-3
Isolated Loop Pressure – Temperature Limit Curve (TS 3.4.9.1)

Table 4.2-1
Heatup Curve Data Points for 22 EFPY (TS 3.4.9.1)

100°F/HR HEATUP				100°F/HR CRITICALITY				LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	200	677	289	0	289	716	271	2000
60	564	205	696	289	564	289	739	289	2485
65	564	210	716	289	565	289	764		
70	564	215	739	289	565	289	792		
75	564	220	764	289	566	289	822		
80	564	225	792	289	566	289	856		
85	564	230	822	289	568	289	894		
90	564	235	856	289	569	289	936		
95	564	240	894	289	571	290	982		
100	564	245	936	289	572	295	1033		
105	564	250	982	289	575	300	1089		
110	564	255	1033	289	577	305	1151		
115	564	260	1089	289	580	310	1219		
120	564	265	1151	289	583	315	1294		
125	564	270	1219	289	586	320	1378		
130	565	275	1294	289	591	325	1470		
135	566	280	1378	289	593	330	1571		
140	568	285	1470	289	600	335	1682		
145	571	290	1571	289	601	340	1806		
150	575	295	1682	289	611	345	1941		
155	580	300	1806	289	612	350	2091		
160	586	305	1941	289	621	355	2222		
165	593	310	2091	289	621	360	2361		
170	601	315	2222	289	621				
175	611	320	2361	289	621				
180	621			289	621				
180	621			289	633				
180	621			289	646				
185	633			289	661				
190	646			289	677				
195	661			289	696				

Table 4.2-2 (Page 1 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR.		40°F/HR.		60°F/HR.		100°F/HR.	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60	0	60	0	60	0	60	0	60	0
60	621	60	609	60	566	60	521	60	430
65	621	65	611	65	567	65	523	65	431
70	621	70	612	70	568	70	524	70	432
75	621	75	614	75	570	75	525	75	433
80	621	80	615	80	572	80	527	80	435
85	621	85	617	85	574	85	529	85	437
90	621	90	619	90	576	90	531	90	439
95	621	95	621	95	578	95	534	95	442
100	621	100	621	100	581	100	536	100	445
105	621	105	621	105	584	105	540	105	448
110	621	110	621	110	587	110	543	110	452
115	621	115	621	115	591	115	547	115	457
120	621	120	621	120	596	120	552	120	462
125	621	125	621	125	600	125	557	125	468
130	621	130	621	130	606	130	562	130	474
135	621	135	621	135	612	135	569	135	481
140	621	140	621	140	618	140	576	140	490
145	621	145	621	145	621	145	584	145	499
150	621	150	621	150	621	150	592	150	509
155	621	155	621	155	621	155	602	155	520
160	621	160	621	160	621	160	613	160	533
165	621	165	621	165	621	165	621	165	547
170	621	170	621	170	621	170	621	170	563
175	621	175	621	175	621	175	621	175	581
180	621	180	621	180	621	180	621	180	600
180	621	180	621	180	621	180	621	185	622
180	778	180	742	180	706	180	670	190	647
185	792	185	757	185	723	185	689	195	674
190	808	190	775	190	742	190	709	200	704
195	826	195	794	195	762	195	732	205	737

Table 4.2-2 (Page 2 of 2)
Cooldown Curve Data Points for 22 EFPY (TS 3.4.9.1)

STEADY STATE		20°F/HR.		40°F/HR.		60°F/HR.		100°F/HR.	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
200	846	200	815	200	785	200	757	210	774
205	868	205	839	205	811	205	785	215	815
210	892	210	865	210	839	210	815	220	861
215	918	215	894	215	871	215	850	225	911
220	947	220	925	220	905	220	888	230	967
225	980	225	961	225	944	225	930	235	1030
230	1016	230	1000	230	986	230	976	240	1099
235	1055	235	1043	235	1033	235	1028	245	1147
240	1099	240	1090	240	1086	240	1085	250	1201
245	1147	245	1143	245	1143	245	1147	255	1260
250	1201	250	1201	250	1201	250	1201	260	1325
255	1260	255	1260	255	1260	255	1260	265	1397
260	1325	260	1325	260	1325	260	1325	270	1477
265	1397	265	1397	265	1397	265	1397	275	1565
270	1477	270	1477	270	1477	270	1477	280	1662
275	1565	275	1565	275	1565	275	1565	285	1770
280	1662	280	1662	280	1662	280	1662	290	1888
285	1770	285	1770	285	1770	285	1770	295	2020
290	1888	290	1888	290	1888	290	1888	300	2165
295	2020	295	2020	295	2020	295	2020	305	2325
300	2165	300	2165	300	2165	300	2165		
305	2325	305	2325	305	2325	305	2325		

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

FUNCTION	SETPOINT
OPPS Enable Temperature	343°F
PORV Setpoint	403 psig

Table 4.2-4

Calculation of Chemistry Factors Using Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF *ΔRT_{NDT}	FF^2
Lower Shell Plate B6903-1[d] (Longitudinal)	V	.323	.689	128.49	88.53	.475
	U	.646	.878	118.93	104.42	.771
	W	.986	.996	148.52	147.93	.992
	Y	2.15	1.21	142.18	172.04	1.464
Lower Shell Plate B6903-1[d] (Transverse)	V	.323	.689	137.81	94.95	.475
	U	.646	.878	131.84	115.76	.771
	W	.986	.996	179.99	179.27	.992
	Y	2.15	1.21	166.93	201.99	1.464
				SUM:	1104.89	7.404
	CF = $\Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ = (1104.89) ÷ (7.404) = 149.2°F					
Beaver Valley	V	.323	.689	169.30	116.65	.475
Surv. Weld Material 305424[d]	U	.646	.878	176.30	154.79	.771
	W	.986	.996	198.99	198.19	.992
	Y	2.15	1.21	189.41	229.19	1.464
				SUM:	698.82	3.702
	CF = $\Sigma(FF*RT_{NDT}) \div \Sigma(FF^2)$ = (698.82) ÷ (3.702) = 188.8°F					

Notes:

(a) F= Calculated fluence from Beaver Valley Unit 1 capsule Y dosimetry analysis results, (x 10^{19} n/cm^2, E > 1.0 Mev).

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

(c) The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ration factor of 1.06.

(d) Data not credible.

Table 4.2-5

Reactor Vessel Beltline Material Properties

Material Description	Cu(%)	Ni(%)	Chemistry Factor	Initial RT_{NDT}(°F) [a]
Intermediate Shell Plate B6607-1	0.14	0.62	100.5	43
Intermediate Shell Plate B6607-2	0.14	0.62	100.5	73
Lower Shell Plate B6903-1	0.21	0.54	147.2	27
Lower Shell Plate B7203-2	0.14	0.57	98.7	20
Intermediate to Lower Shell Weld Seam (Heat 90136) 11-714	0.27	0.07	124.3	-56
Intermediate Longitudinal Shell Weld Seams (Heat 305424) 19-714 A&B	0.28	0.63	191.7	-56
Lower Longitudinal Weld Seams (Heat 305414) 20-714 A&B	0.34	0.61	210.5	-56
Surveillance Weld (Heat 305424)	0.26	0.61	181.6	---

Note:

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

Table 4.2-6

Summary of Adjusted Reference Temperature (ARTs) for 22 EFPY

MATERIAL DESCRIPTION	22 EFPY	
	1/4T ART(°F)[a]	3/4T ART(°F)[a]
Intermediate Shell Plate B6607-1	193	166
Intermediate Shell Plate B6607-2	223	196
Lower Shell Plate B7203-2	168	141
Lower Shell Plate B6903-1	230	191
- Using S/C Data[b]	233	193
Intermediate Shell Longitudinal Weld 19-714A/B	145	102
- Using S/C Data[b]	143	100
Intermediate to Lower Shell Circ. Weld 11-714	152	119
- Using S/C Data [c]	86	63
Lower Shell Longitudinal Weld 20-714A/B	159	111
- Using S/C Data[d]	168	117

Notes:

(a) $ART = I + \Delta RT_{NDT} + M$.

(b) Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_Δ.)

(c) Based on credible St. Lucie Unit 1 surveillance data.

(d) Based on Fort Calhoun Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_Δ.)

Table 4.2-7

Calculation of Adjusted Reference Temperatures (ARTs) for 22 EFPY

PARAMETER	VALUES	
Operating Time	22 EFPY	
Material	Plate B6903-1	Plate B6607-2
Location	Lower Shell Plate 1/4T ART(°F)	Intermediate Shell Plate 3/4T ART(°F)
Chemistry Factor, CF (°F)	149.2	100.5
Fluence (f), n/cm^2 (E>1.0 Mev)[a]	1.70 x 10^{19}	6.62 x 10^{18}
Fluence Factor, FF	1.15	.884
ΔRT_{NDT} = CF x FF(°F)[c]	171.6 [c]	88.84
Initial RT_{NDT}, I(°F)[a]	27	73
Margin, M(°F)	34 [c]	34
ART = I+(CF*FF)+M, °F[b] per RG 1.99, Revision 2	233	196

Notes:

(a) Initial RT_{NDT} values are measured values for plate material.

(b) This value was rounded per ASTM E29, using the "Rounding Method."

(c) Based on Beaver Valley Unit 1 surveillance data. (Data not credible. ART calculated with a full σ_Δ.)

Table 4.2-8
Reactor Vessel Toughness Data (Unirradiated)

COMPONENT	HEAT NO.	CODE NO.	MATERIAL TYPE	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	UPPER SHELF ENERGY (FT-LB) MWD	NMWD
Closure Head Dome	C6213-1B	B6610	A533B CL. 1	.15	---	.010	-40	0*	121	---
Closure Head Seg.	A5518-2	B6611	A533B CL. 1	.14	---	.015	-20	-20*	131	---
Closure Head Flange	ZV3758	---	A508 CL. 2	.08	---	.007	60*	60*	>100	---
Vessel Flange	ZV3661	---	A508 CL. 2	.12	---	.010	60*	60*	166	---
Inlet Nozzle	9-5443	---	A508 CL. 2	.10	---	.008	60*	60*	82.5	---
Inlet Nozzle	9-5460	---	A508 CL. 2	.10	---	.010	60*	60*	94	---
Inlet Nozzle	9-5712	---	A508 CL. 2	.08	---	.007	60*	60*	97	---
Outlet Nozzle	9-5415	---	A508 CL. 2	---	---	.008	60*	60*	97	---
Outlet Nozzle	9-5415	---	A508 CL. 2	---	---	.007	60*	60*	112.5	---
Outlet Nozzle	9-5444	---	A508 CL. 2	.09	---	.007	60*	60*	103	---
Upper Shell	123V339	---	A508 CL. 2	---	---	.010	40	40*	155	---
Inter Shell	C4381-2	B6607-2	A533B CL. 1	.14	.62	.015	-10	73	123	82.5
Inter Shell	C4381-1	B6607-1	A533B CL. 1	.14	.62	.015	-10	43	128.5	90
Lower Shell	C6317-1	B6903-1	A533B CL. 1	.20	.54	.010	-50	27	134	80
Lower Shell	C6293-2	B7203-2	A533B CL. 1	.14	.57	.015	-20	20	129.5	83.5
Trans Ring	123V223	---	A508 CL. 2	---	---	---	30	30*	143	---
Bottom Hd Seg	C4423-3	B6618	A533B CL. 1	.13	---	.008	-30	-29*	124	---
Bottom Hd Dome	C4482-1	B6619	A533B CL. 1	.13	---	.015	-50	-33*	125.5	---
Inter to Lower Shell Weld	90136	---	---	.27	.07	---	---	-56	---	> 100
Inter Shell Long. Weld	305424	---	---	.28	.63	---	---	-56	---	> 100
Lower Shell Long. Weld	305414	---	---	.34	.61	---	---	-56	---	> 100
Weld HAZ				---	---	---	-40	-40	---	136.5

*Estimated Per NRC Standard Review Plan Branch Technical Position MTEB 5-2
MWD – Major Working Direction
NMWD – Normal to Major Working Direction
Note: For evaluation of Inservice Reactor Vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter 92-01 or equivalent document are applicable.

Table 4.2-9

RT_{PTS} Calculation for Beltline Region Materials at EOL (28 EFPY)

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} (a) (°F)	Margin (°F)	$RT_{NDT(U)}$ (b) (°F)	RT_{PTS} (c) (°F)
Intermediate Shell Plate B6607-1	3.54	1.329	100.5	133.6	34	43	211
Intermediate Shell Plate B6607-2	3.54	1.329	100.5	133.6	34	73	241
Lower Shell Plate B7203-2	3.54	1.329	98.7	131.2	34	20	185
Lower Shell Plate B6903-1	3.54	1.329	147.2	195.6	34	27	257
→ Using S/C Data(e)	3.54	1.329	149.2	198.3	34	27	259
Inter. Shell Long. Weld 19-714A/B	0.708	0.903	191.7	173.1	65.5	-56	183
→ Using S/C Data(e)	0.708	0.903	188.8	170.5	65.5	-56	180
Lower Shell Long. Weld 20-714A/B	0.708	0.903	210.5	190.1	65.5	-56	200
→ Using S/C Data(f)	0.708	0.903	223.9	202.2	65.5	-56	212
Circumferential Weld 11-714	3.53	1.329	124.3	165.2	65.5	-56	175
→ Using S/C Data(d)	3.53	1.329	84.8	112.3	44	-56	101

Notes:

(a) $\Delta RT_{PTS} = CF * FF$.

(b) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.

(c) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} +$ Margin (°F).

(d) Based on credible St. Lucie Unit 1 surveillance data.

(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_Δ.

(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_Δ.

Table 4.2-10

RT_{PTS} Calculation for Beltline Region Materials at Life extension (45 EFPY)

Material	Fluence (10^{19} n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Plate B6607-1	5.85	1.43	100.5	143.7	34	43	221
Intermediate Shell Plate B6607-2	5.85	1.43	100.5	143.7	34	73	251
Lower Shell Plate B7203-2	5.85	1.43	98.7	141.1	34	20	195
Lower Shell Plate B6903-1	5.85	1.43	147.2	210.5	34	27	272
→ Using S/C Data[e]	5.85	1.43	149.2	213.4	34	27	274
Inter. Shell Long. Weld 19-714A/B	1.13	1.03	191.7	197.5	65.5	-56	207
→ Using S/C Data[e]	1.13	1.03	188.8	194.5	65.5	-56	204
Lower Shell Long. Weld 20-714A/B	1.13	1.03	210.5	216.8	65.5	-56	226
→ Using S/C Data[f]	1.13	1.03	223.9	230.6	65.5	-56	240
Circumferential Weld 11-714	5.82	1.43	124.3	177.7	65.5	-56	187
→ Using S/C Data[d]	5.82	1.43	84.8	121.3	44	-56	109

Notes:

(a) Initial RT_{NDT} values of the plate material are measured values while the weld material values are generic.

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F).

(c) $\Delta RT_{PTS} = CF * FF$.

(d) Based on credible St. Lucie Unit 1 surveillance data.

(e) Based on non-credible Beaver Valley Unit 1 surveillance data with a full σ_Δ.

(f) Based on non-credible Fort Calhoun Unit 1 surveillance data with a full σ_Δ.

ATTACHMENT C-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 167

Proposed PTLR for Unit 2

BVPS-2 Technical Specification to PTLR Cross-Reference			
Technical Specification	PTLR		
	Section	Figure	Table
3.1.2.2	N/A	N/A	4.2-3
3.1.2.4	N/A	N/A	4.2-3
3.4.1.3	N/A	N/A	4.2-3
3.4.9.1	4.2.1.1	4.2-1 4.2-2 4.2-3 4.2-4 4.2-5 4.2-6	N/A
3.4.9.3	4.2.1.2 4.2.1.3	4.2-8	4.2-3
3.5.2	N/A	N/A	4.2-3
3.5.3	N/A	N/A	4.2-3

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4.2 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

This Pressure and Temperature Limits Report (PTLR) for Unit 2 has been prepared in accordance with the requirements of Technical Specification 6.9.6. Revisions to the PTLR shall be provided to the NRC after issuance.

The Technical Specifications (TS) addressed in, or make reference to, this report are listed below:

TS 3.1.2.2 Reactivity Control Systems – Flow Paths – Operating,

TS 3.1.2.4 Reactivity Control Systems – Charging Pumps – Operating,

TS 3.4.1.3 Reactor Coolant System – Shutdown,

TS 3.4.9.1 Reactor Coolant System - Pressure/Temperature Limits,

TS 3.4.9.3 Overpressure Protection Systems,

TS 3.5.2 ECCS Subsystems – $T_{avg} \geq 350°F$, and

TS 3.5.3 ECCS Subsystems – $T_{avg} < 350°F$.

4.2.1 Operating Limits

The PTLR limits for Beaver Valley Power Station (BVPS) Unit 2 were developed using a methodology specified in the Technical Specifications. The methodology listed in Reference 1 was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

4.2.1.1 RCS Pressure and Temperature (P/T) Limits (TS 3.4.9.1)

The RCS temperature rate-of-change limits defined in Reference 2 are:

a. A maximum heatup of 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period, and

c. A maximum temperature change of less than or equal to 5°F in any one hour period during inservice hydrostatic testing operations above system design pressure.

The RCS P/T limits for heatup, leak testing, and criticality are specified by Figure 4.2-1 and Table 4.2-1. The RCS P/T limits for cooldown are shown in Figure 4.2-2 through 4.2-6 and Table 4.2-2. These limits are defined in Reference 2. Consistent with the methodology described in Reference 1, including the exceptions as noted in Section 4.2.1, the RCS P/T limits for heatup and cooldown shown in Figures 4.2-1 through and 4.2-6 are provided without margins for instrument error. The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in 10 CFR 50, Appendix G. The heatup and cooldown curves also include the effect of the reactor vessel flange.

The heatup and cooldown curves are shown for 14 effective full power years (EFPY), although the capsule data provided in Tables 4.2-5 through 4.2-10 state 15 EFPY. The EFPY for the heatup and cooldown curves were generated for 15 EFPY. However, the heatup and cooldown curves were revised by license amendment 243 (1.4 % power uprate). The change from 15 to 14 EFPY was done to impose a conservative administrative limit due to the increased neutron fluence associated with the 1.4% increase in reactor power. The capsule data tables however, continue to reflect 15 EFPYs to retain actual capsule analysis results.

The P/T limits for core operation (except for low power physics testing) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown.

Pressure-temperature limit curves shown in Figure 4.2-7 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640.

4.2.1.2 Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3)

The power operated relief valves (PORVs) shall each have nominal maximum lift setting is in accordance with Figure 4.2-8. The OPPS enable temperature is in accordance with Table 4.2-3. The PORV lift setting provided is for the case with reactor coolant pump (RCP) restrictions. These restrictions are shown in Table 4.2-4, which is taken from Reference 10. Due to the setpoint limitations as a result of the reactor vessel flange requirements, there is no operational benefit achieved by restricting the number of RCPs running to less than two below an indicated RCS temperature of 190°F. Therefore, the PORV setpoints shown in Table 4.2-4 will protect the Appendix G limits for the combinations shown.

The PORV setpoint is based on P/T limits which were established in accordance with 10 CFR 50, Appendix G without allowance for instrumentation error and in accordance with the methodology described in Reference 1, including the exceptions noted in Section 4.2.1. The PORV lift setting shown in Figure 4.2-8 accounts for appropriate instrument error.

4.2.1.3 OPPS Enable Temperature (TS 3.4.9.3)

Two different temperatures are used to determine the OPPS enable temperature, they are the arming temperature and the calculated enable temperature. The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature with uncertainty is 236.5°F.

The calculated enable temperature is based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The calculated enable temperature does not address the piping limit attributed to a water hammer discharge. The calculated enable temperature is 350°F.

As the calculated enable temperature is higher and, therefore, more conservative than the arming temperature, the OPPS enable temperature, as shown in Table 4.2-3, is set to equal the calculated enable temperature.

The calculation method governing the heatup and cooldown of the RCS requires the arming of the OPPS at and below the OPPS enable temperature specified in Table 4.2-3, and disarming of the OPPS above this temperature. The OPPS is required to be enabled, i.e., OPERABLE, when any RCS cold leg temperature is less than or equal to this temperature.

From a plant operations viewpoint the terms "armed" and "enabled" are synonymous when it comes to activating the OPPS. As stated in the applicable operating procedure, the OPPS is activated (armed/enabled) manually before entering the applicability of TS 3.4.9.3. This is accomplished by placing two keylock switches (one in each train) into their "ARM" position. Once OPPS is activated (armed/enabled) reactor coolant system pressure transmitters will signal a rise in system pressure above the variable OPPS setpoint. This will initiate an alarm in the control room and open the OPPS PORVs.

4.2.1.4 Reactor Vessel Boltup Temperature (TS 3.4.9.1)

The minimum boltup temperature for the Reactor Vessel Flange shall be ≥ 60°F. Boltup is a condition in which the reactor vessel head is installed with tension applied to any stud, and with the RCS vented to atmosphere.

4.2.2 Reactor Vessel Material Surveillance Program

The reactor vessel material irradiation surveillance specimens shall be removed and analyzed to determine changes in material properties. The capsule withdrawal schedule is provided in Table 5.3-6 of the UFSAR. Also, the results of these analyses shall be used to update Figures 4.2-1 through 4.2-6, and Tables 4.2-1 and 4.2-2. The time of specimen withdrawal may be modified to coincide with those refueling outages or reactor shutdowns most closely approaching the withdrawal schedule.

The pressure vessel material surveillance program (References 3 and 4) is in compliance with Appendix H to 10 CFR 50, "Reactor Vessel Radiation Surveillance Program." The material test requirements and the acceptance standards utilize the reference nil-ductility temperature, RT_{NDT}, which is determined in accordance with ASME, Section III, NB-2331. 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 XI of the ASME Boiler and Pressure Vessel Code. The surveillance capsule removal schedule meets the requirements of ASTM E 185-82.

4.2.3 Supplemental Data Tables

The following tables provide supplemental information on reactor vessel material properties and are provided to be consistent with Generic Letter 96-03. Some of the material property values shown were used as inputs to the P/T limits.

Table 4.2-5, taken from Reference 2, shows the calculation of the surveillance material chemistry factors using surveillance capsule data.

Table 4.2-6, taken from Reference 2, provides the reactor vessel beltline material property table.

Table 4.2-7, taken from Reference 2, provides a summary of the Adjusted Reference Temperature (ARTs) for 15 EFPY.

Table 4.2-8, taken from Reference 2, shows the calculation of ARTs for 15 EFPY.

Table 4.2-9 shows the Reactor Vessel Toughness Data (Unirradiated).

Table 4.2-10, taken from Reference 6, provides RT_{PTS} values for 32 EFPY.

4.2.4 References

1. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andrachek, et al., January 1996.
2. WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626," T. J. Laubham, January 1999.
3. WCAP-14484, Revision 0, "Analysis of Capsule V from the Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Grendys, S. L. Anderson, J. F. Williams, February 1996.
4. WCAP-9615, Revision 1, "Duquesne Light Company, Beaver Valley Unit No. 2 Reactor Vessel Radiation Surveillance Program," P. A. Peter, June 1995.
5. WCAP-14485, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves for Normal Operation," P. A. Grendys, March 1986.
6. WCAP-14784, Revision 2, "Evaluation of Pressurized Thermal Shock for Beaver Valley Unit 2," T. J. Laubham, February 1996.
7. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.
8. 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," May 15, 1991. (PTS Rule)
9. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
10. Westinghouse Report NPD-OPES(99)-055, "Low Temperature Overpressure Protection System Setpoint Review for Beaver Valley Unit 2 15 EFPY Heatup and Cooldown Curves," March 1999.
11. Westinghouse Letter FENOC-01-261, COMS Arming Temperature", E. A. Dzenis, September 10, 2001.

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MATERIAL PROPERTY BASIS	
CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

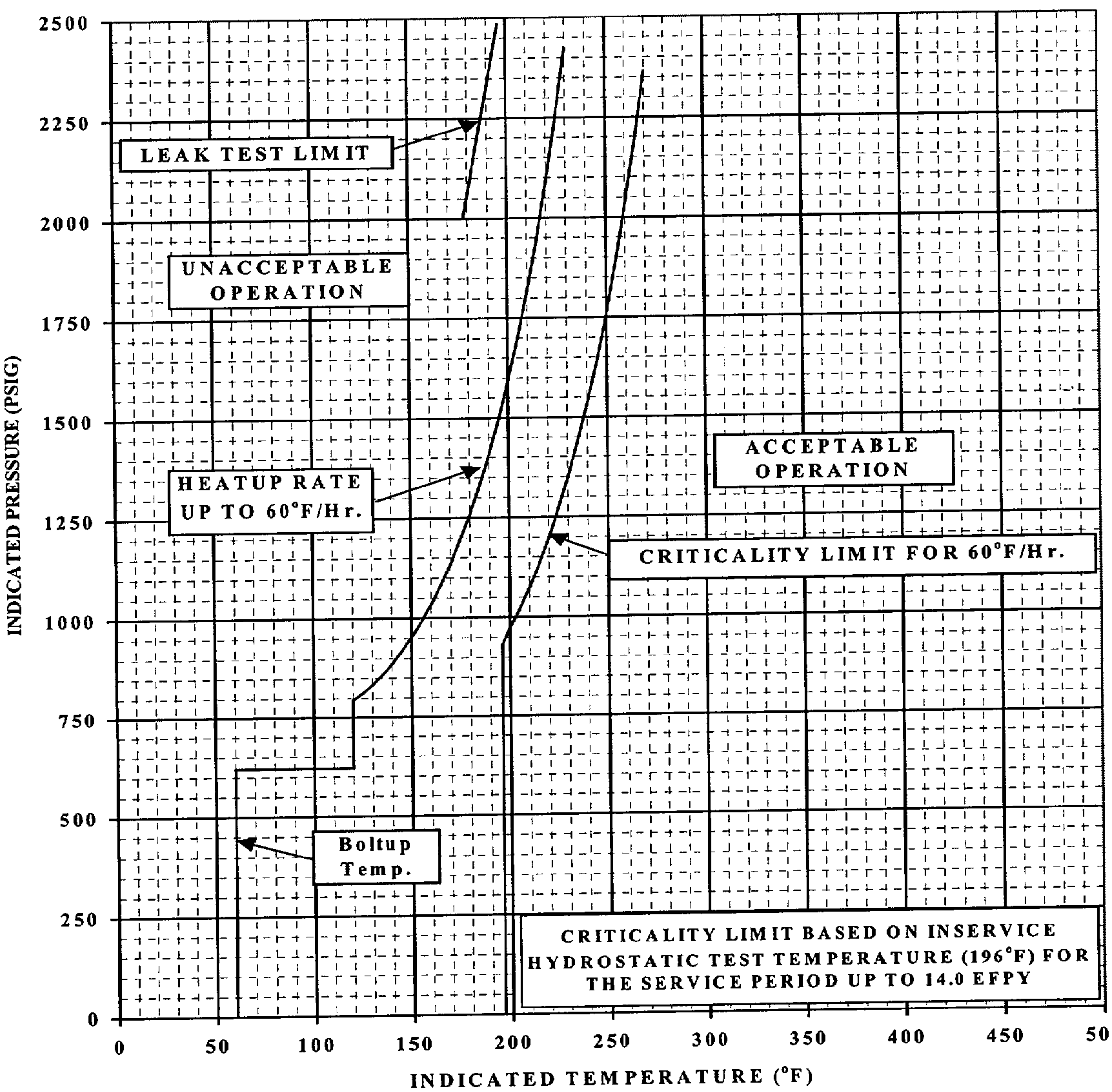


Figure 4.2-1
Reactor Coolant System Heatup
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 0°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

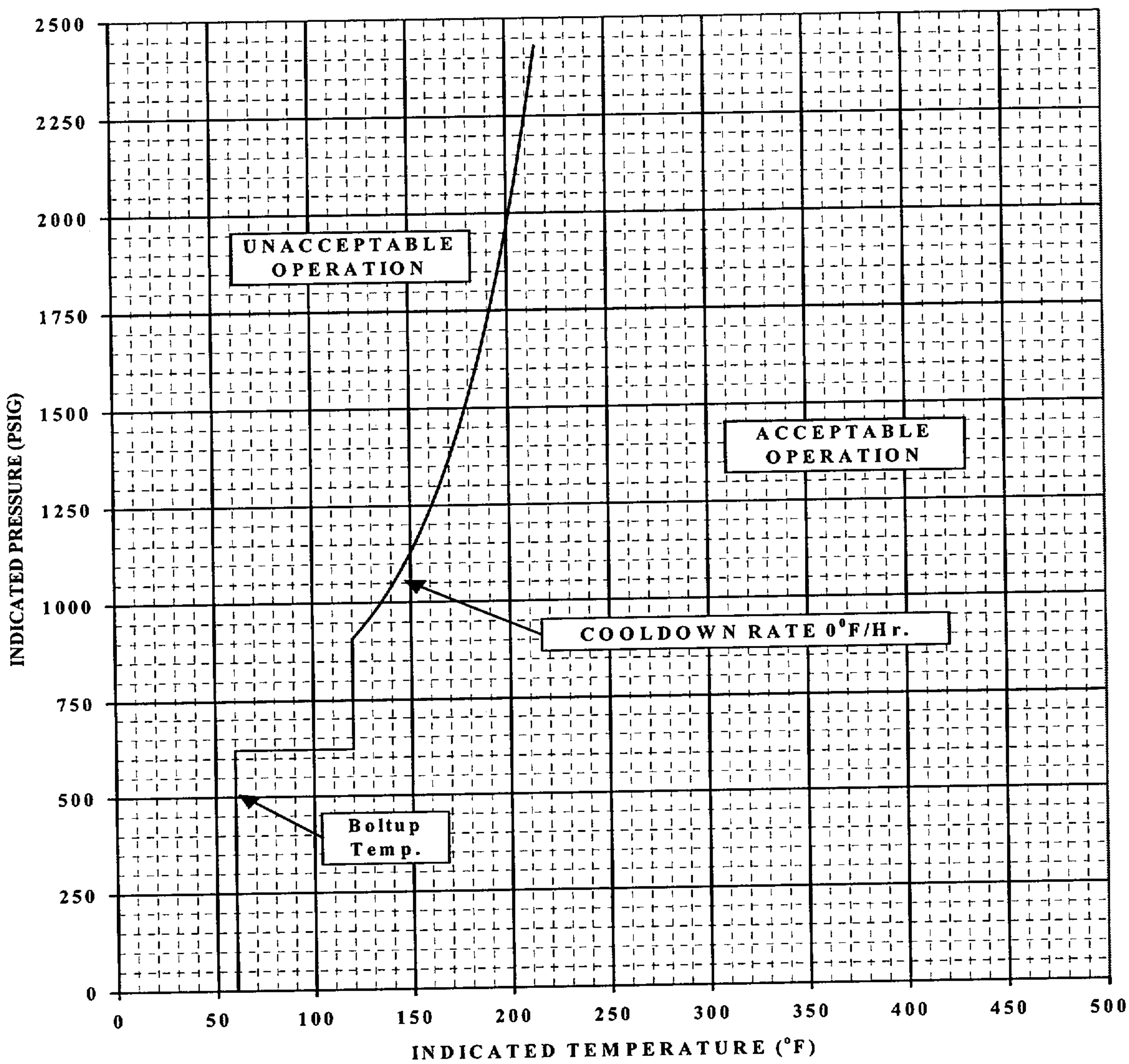


Figure 4.2-2
Reactor Coolant System Cooldown (up to 0°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 20°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

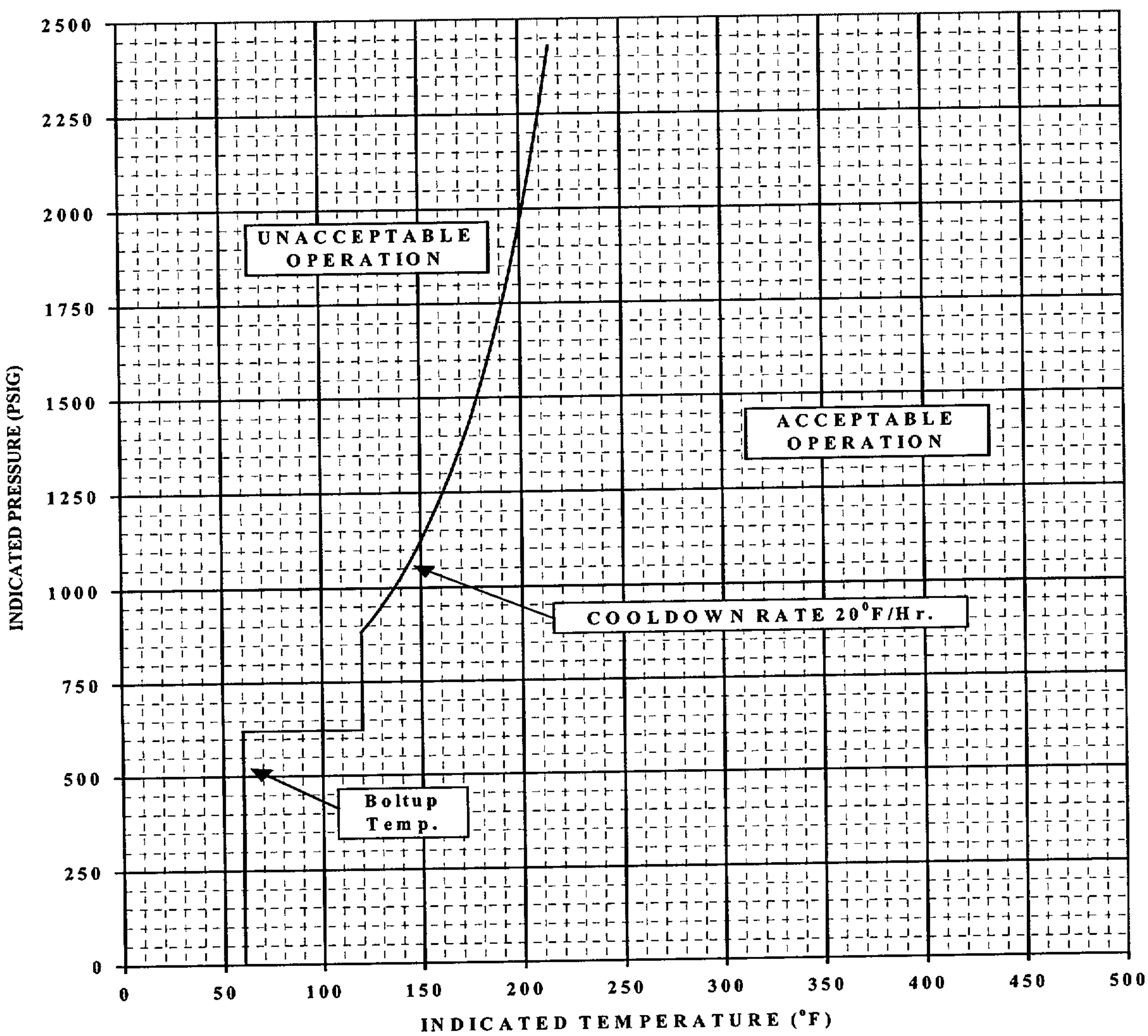


Figure 4.2-3
Reactor Coolant System Cooldown (up to 20°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 40°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

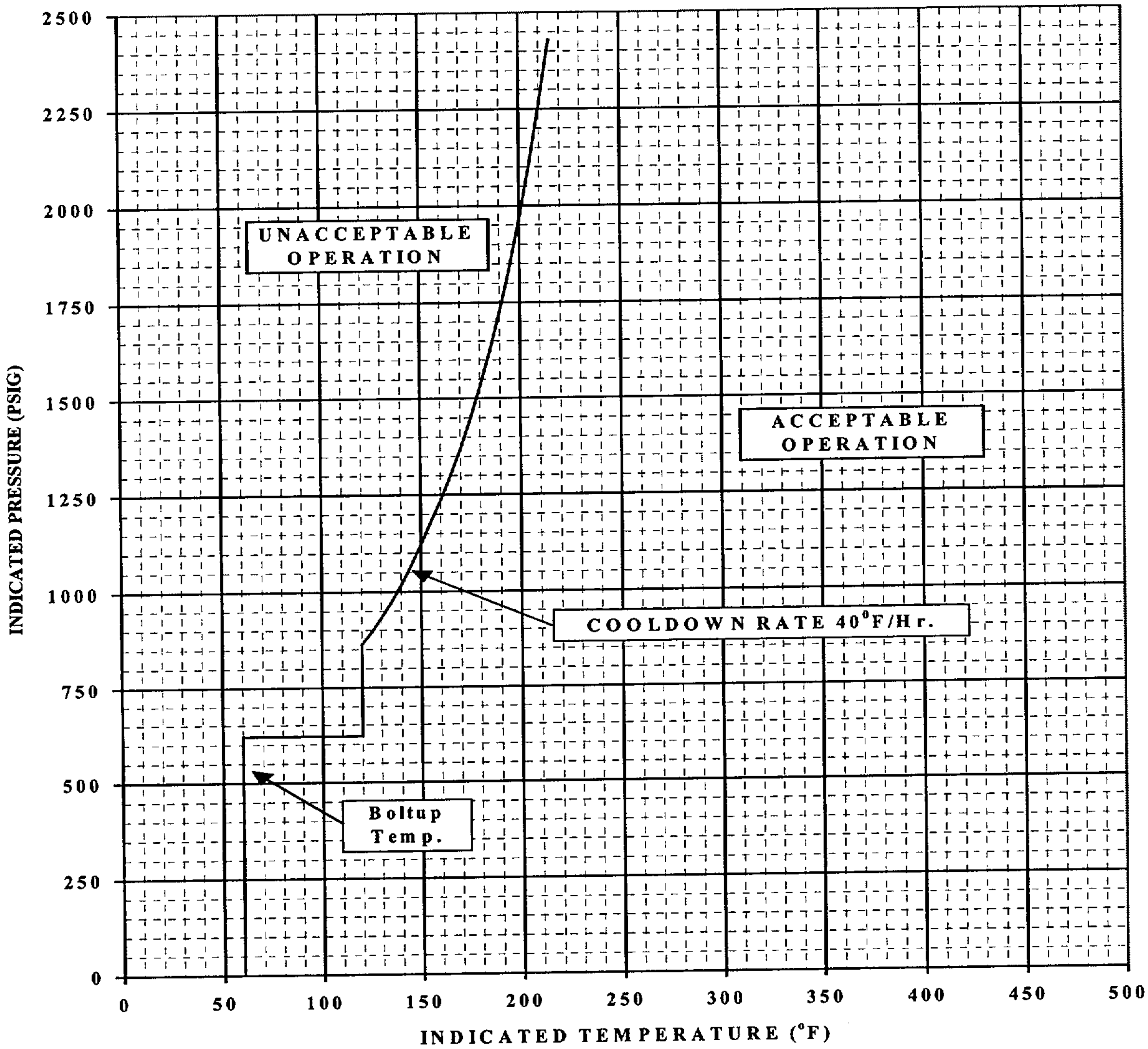


Figure 4.2-4
Reactor Coolant System Cooldown (up to 40°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL RT_{NDT}: 60°F

RT_{NDT} AFTER 14 EFPY: 1/4T, 140°F

3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

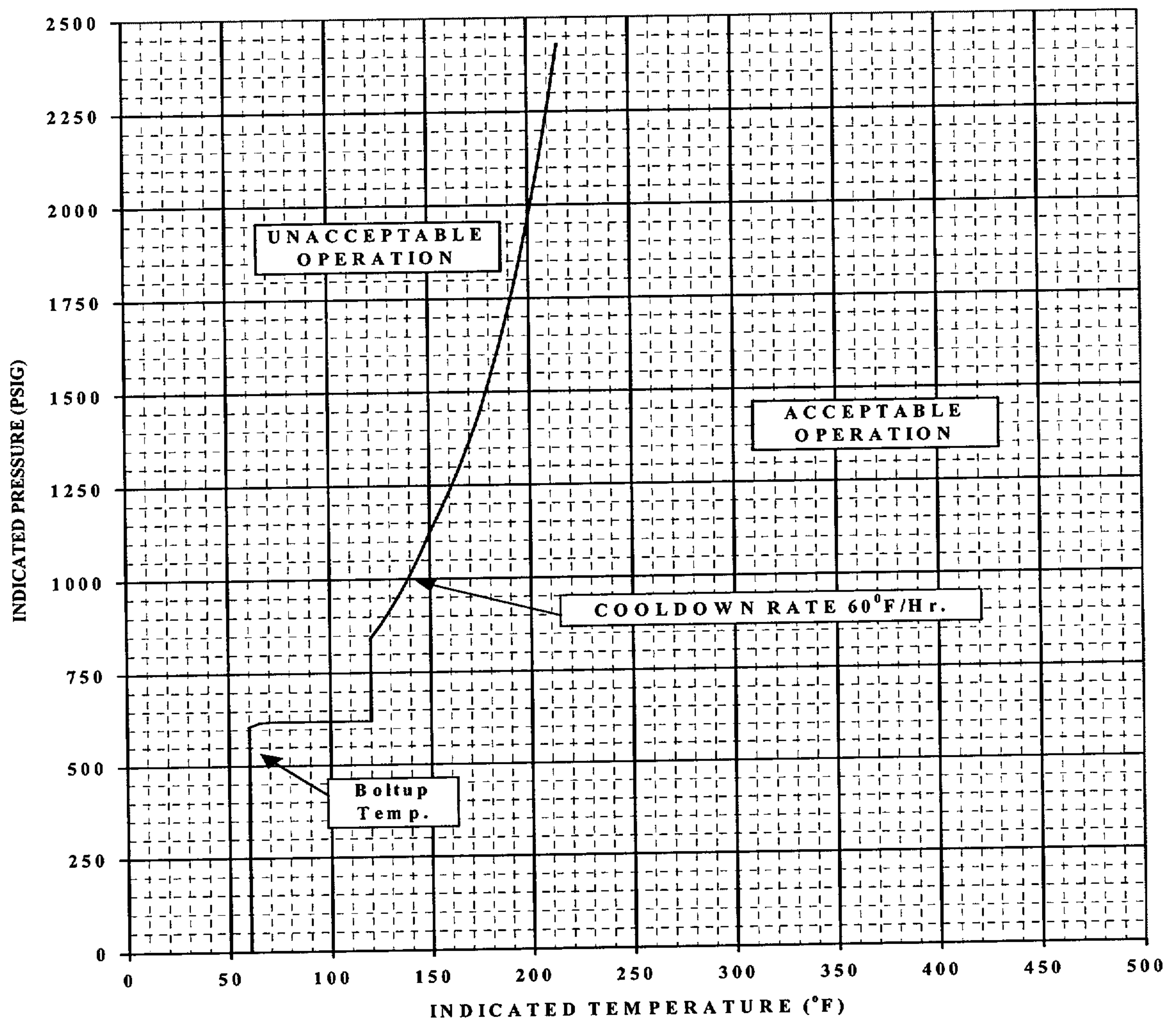


Figure 4.2-5
Reactor Coolant System Cooldown (up to 60°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL:	INTERMEDIATE SHELL PLATE B9004-1
INITIAL RT_{NDT}:	60°F
RT_{NDT} AFTER 14 EFPY:	1/4T, 140°F
	3/4T, 128°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 14 EFPY.

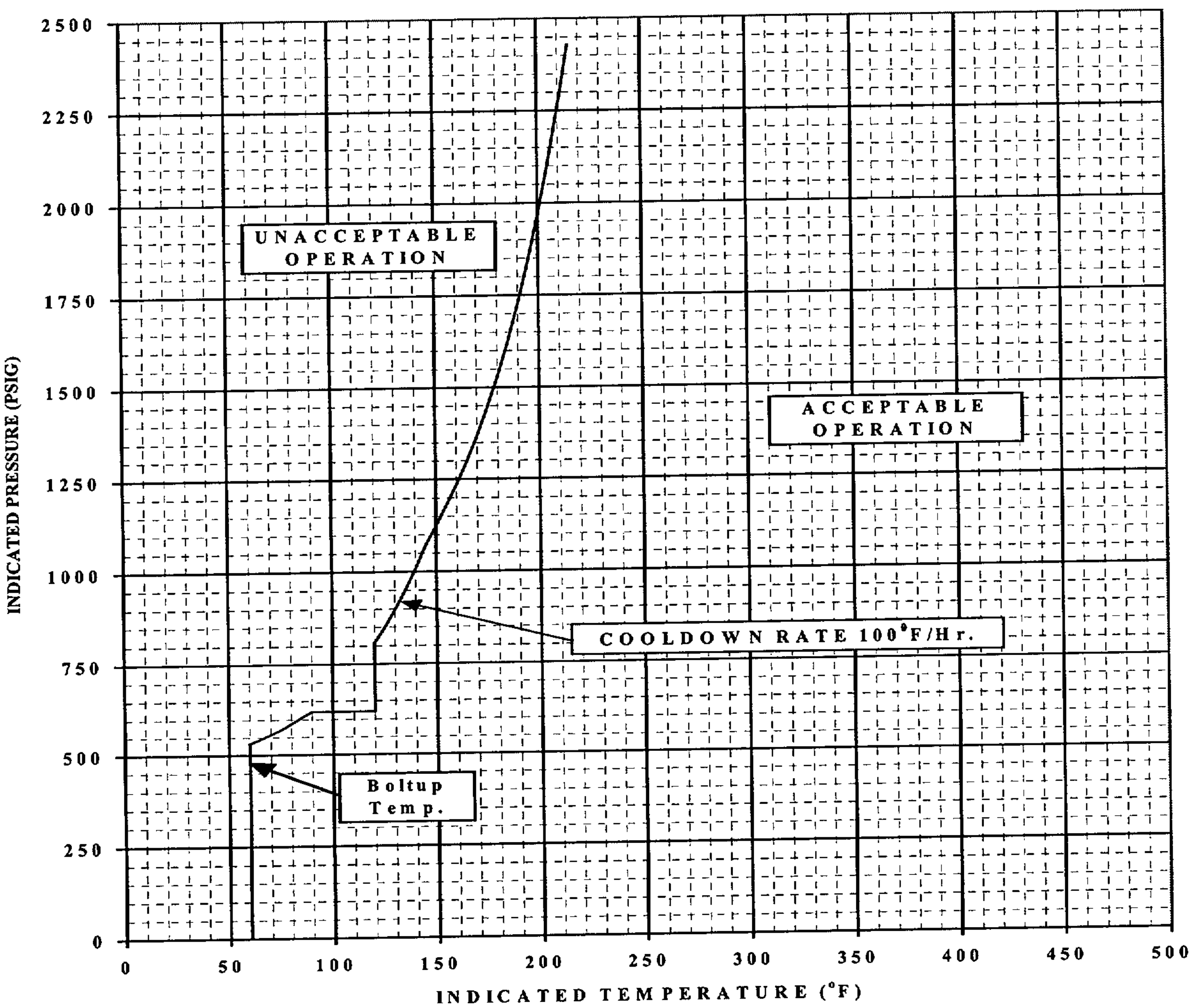


Figure 4.2-6
Reactor Coolant System Cooldown (up to 100°F/Hr.)
Limitations Applicable for the First 14 EFPY (TS 3.4.9.1)

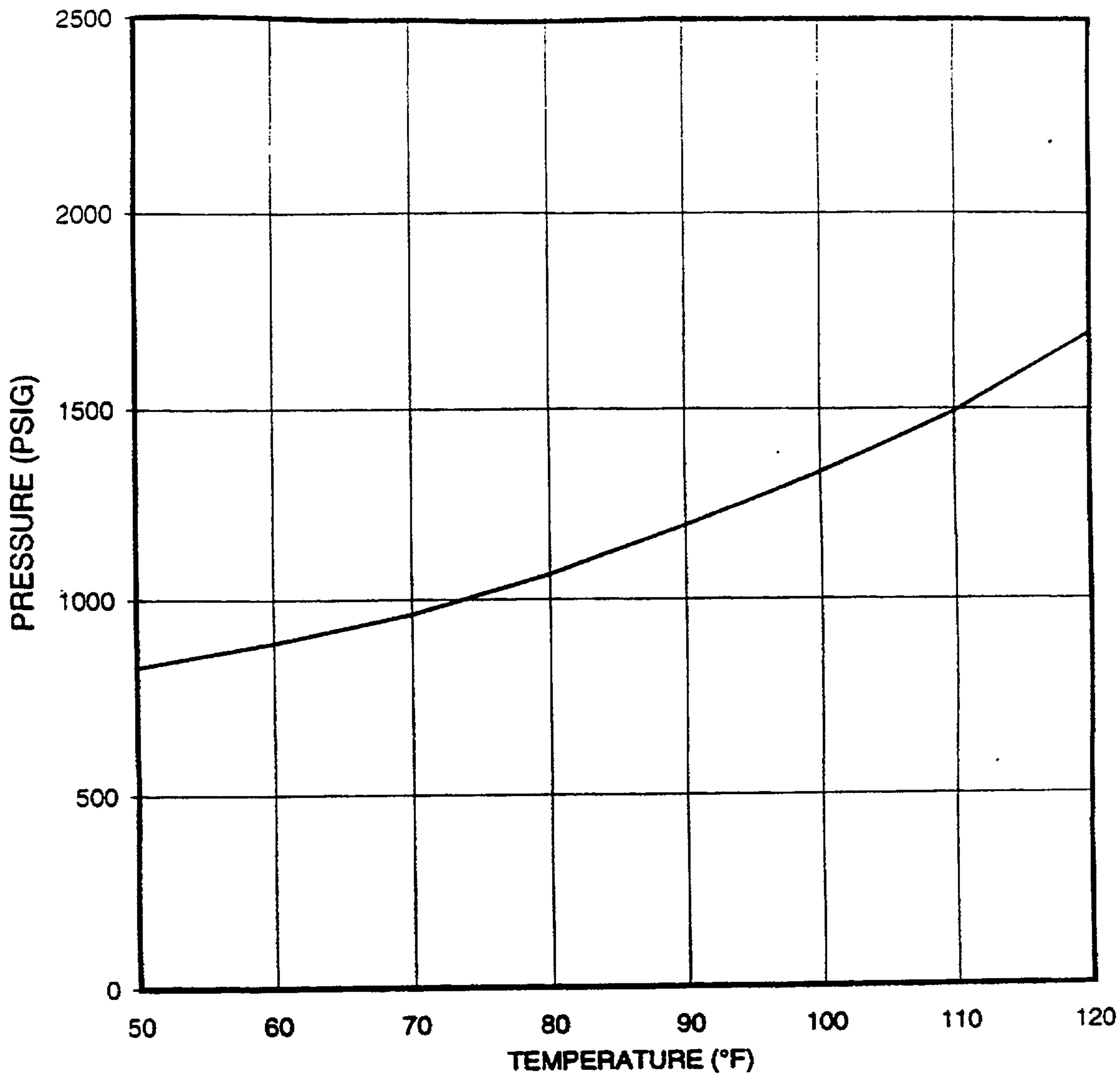


Figure 4.2-7
Isolated Loop Pressure – Temperature Limit Curve (TS 3.4.9.1)

See Table 4.2-4 for RCP restrictions.

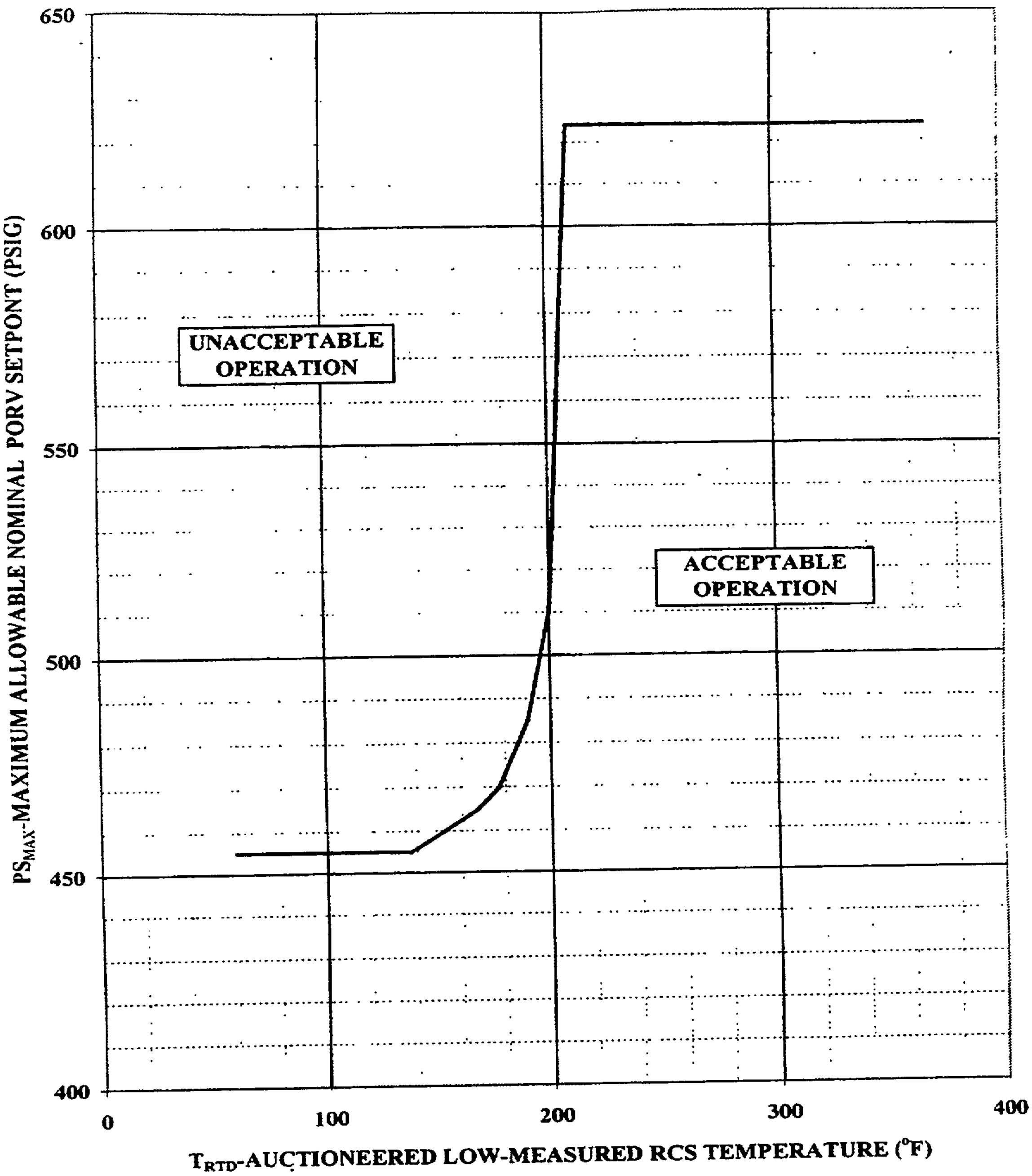


Figure 4.2-8
Maximum Allowable Nominal PORV Setpoint for the Overpressure Protection System (TS 3.4.9.3).

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-1
Heatup Curve Data Points for 14 EFPY (TS 3.4.9.1)

60°F/HR HEATUP		60°F/HR CRITICALITY		LEAK TEST LIMIT	
Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)	Temp. (°F)	Press. (psig)
60.00	621.00	196.00	0.00	178.00	2000.00
65.00	621.00	196.00	668.58	196.00	2485.00
85.00	621.00	196.00	693.10		
90.00	621.00	196.00	687.74		
95.00	621.00	196.00	686.96		
100.00	621.00	196.00	689.93		
105.00	621.00	196.00	696.41		
110.00	621.00	196.00	705.98		
115.00	621.00	196.00	718.57		
120.00	621.00	196.00	734.02		
120.00	794.02	196.00	752.40		
125.00	812.40	196.00	773.69		
130.00	833.69	196.00	798.06		
135.00	858.06	196.00	825.62		
140.00	885.62	196.00	856.64		
145.00	916.64	196.00	891.32		
150.00	951.32	196.00	930.00		
155.00	990.00	200.00	973.00		
160.00	1033.00	205.00	1020.74		
165.00	1080.74	210.00	1073.65		
170.00	1133.65	215.00	1132.24		
175.00	1192.24	220.00	1197.06		
180.00	1257.06	225.00	1268.73		
185.00	1328.73	230.00	1347.95		
190.00	1407.95	235.00	1435.48		
195.00	1495.48	240.00	1532.16		
200.00	1592.16	245.00	1638.92		
205.00	1698.92	250.00	1756.81		
210.00	1816.81	255.00	1886.95		
215.00	1946.95	260.00	2030.62		
220.00	2090.62	265.00	2189.19		
225.00	2249.19	270.00	2364.21		
230.00	2424.21				

LICENSING REQUIREMENTS MANUAL

PRESSURE AND TEMPERATURE LIMITS REPORT

Table 4.2-2
Cooldown Curve Data Points for 14 EFPY (TS 3.4.9.1)

	0^{O}F/HR.	20^{O}F/HR.	40^{O}F/HR.	60^{O}F/HR.	100^{O}F/HR.
Temp. (^{o}F)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)	Press. (psig)
60.00	621.00	621.00	621.00	607.90	531.77
65.00	621.00	621.00	621.00	618.15	543.64
70.00	621.00	621.00	621.00	621.00	556.89
75.00	621.00	621.00	621.00	621.00	571.67
80.00	621.00	621.00	621.00	621.00	588.13
85.00	621.00	621.00	621.00	621.00	606.46
90.00	621.00	621.00	621.00	621.00	621.00
95.00	621.00	621.00	621.00	621.00	621.00
100.00	621.00	622.00	621.00	621.00	621.00
105.00	621.00	621.00	621.00	621.00	621.00
110.00	621.00	621.00	621.00	621.00	621.00
115.00	621.00	621.00	621.00	621.00	621.00
120.00	621.00	621.00	621.00	621.00	621.00
120.00	907.20	883.94	862.00	841.62	806.75
125.00	935.36	914.39	894.99	877.44	849.30
130.00	966.47	948.05	931.50	917.13	896.50
135.00	1000.87	985.31	971.92	961.10	948.86
140.00	1038.88	1026.49	1016.66	1009.79	1006.92
145.00	1080.88	1072.06	1066.18	1063.72	1071.29
150.00	1127.31	1122.43	1120.97	1123.42	1123.42
155.00	1178.62	1178.62	1178.62	1178.62	1178.62
160.00	1235.32	1235.32	1235.32	1235.32	1235.32
165.00	1297.99	1297.99	1297.99	1297.99	1297.99
170.00	1367.25	1367.25	1367.25	1367.25	1367.25
175.00	1443.79	1443.79	1443.79	1443.79	1443.79
180.00	1528.38	1528.38	1528.38	1528.38	1528.38
185.00	1621.87	1621.87	1621.87	1621.87	1621.87
190.00	1725.19	1725.19	1725.19	1725.19	1725.19
195.00	1839.38	1839.38	1839.38	1839.38	1839.38
200.00	1965.58	1965.58	1965.58	1965.58	1965.58
205.00	2105.05	2105.05	2105.05	2105.05	2105.05
210.00	2259.18	2259.18	2259.18	2259.18	2259.18
215.00	2429.53	2429.53	2429.53	2429.53	2429.53

Table 4.2-3

Overpressure Protection System (OPPS) Setpoints (TS 3.4.9.3).

FUNCTION	SETPOINT
OPPS Enable Temperature	350°F
PORV Setpoint	Figure 4.2-8

Table 4.2-4

Reactor Coolant Pump Restrictions

T_{RCS}	Running RCPs
< 190°F	0 – 2
≥ 190°F	3

Table 4.2-5
Calculation of Chemistry Factors Using Surveillance Capsule Data[(a)(e)]

Material	Capsule	Capsule f[(b)]	FF[(c)]	ΔRT_{NDT}[(d)]	FF *ΔRT_{NDT}	FF^2
Intermediate Shell Plate B9004-2 (Longitudinal)	U	0.601	0.857	24.26	20.8	0.735
	V	2.64	1.26	55.93	70.5	1.59
Intermediate Shell Plate B9004-2 (Transverse)	U	0.601	0.857	17.56	15.1	0.735
	V	2.64	1.26	46.27	58.3	1.59
				SUM:	164.7	4.66
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 35.3$					
Weld Metal	U	0.601	0.857	3.64	3.1	0.735
	V	2.64	1.26	25.47	32.1	1.59
				SUM:	35.2	2.32
	$CF = \Sigma(FF*RT_{NDT}) \div \Sigma(FF^2) = 15.2$					

Notes:

(a) Regulatory Guide 1.99, Revision 2, Position 2.1.
(b) f = fluence (10^{19} n/cm^2); Fluence values were taken from Capsule V analysis (Reference 4).
(c) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.
(d) ΔRT_{NDT} values obtained from CVGRAPH Version 4.0.
(e) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-6
Reactor Vessel Beltline Material Properties [c]

Material	Method Used To Calculate CF[a]	Average Cu wt %	Average Ni wt %	Chemistry Factor (°F)	Initial RT_{NDT}[b] (°F)
Closure Head Flange	N/A	N/A	0.74	N/A	-10
Vessel Flange	N/A	N/A	0.73	N/A	0
Intermediate Shell Plate B9004-1	Position 1.1	0.065	0.55	44.0	60
Intermediate Shell Plate B9004-2	Position 1.1	0.06	0.57	37.0	40
	Position 2.1	N/A	N/A	35.3	40
Lower Shell Plate B9005-1	Position 1.1	0.08	0.58	51.0	28
Lower Shell Plate B9005-2	Position 1.1	0.07	0.57	44.0	33
Weld Metal (Longitudinal & Circumferential Seams)	Position 1.1	0.05	0.07	34.1	-30
	Position 2.1	N/A	N/A	15.2	-30

Notes:

(a) Regulatory Guide 1.99, Revision 2, Position.

(b) Initial RT_{NDT} values of the base metal and weld metal materials are measured values.

(c) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-7

Summary of Adjusted Reference Temperature (ARTs) for 15 EFPY [b]

MATERIAL DESCRIPTION	Method Used To Calculate the CF[a]	15 EFPY ART	
		1/4T ART(°F)	3/4T ART(°F)
Intermediate Shell Plate B9004-1	Position 1.1	140	128
Intermediate Shell Plate B9004-2	Position 1.1	112	97
	Position 2.1	94	84
Lower Shell Plate B9005-1	Position 1.1	115	101
Lower Shell Plate B9005-2	Position 1.1	112	101
Longitudinal Welds (located at 45° azimuthal angle)	Position 1.1	19	3
	Position 2.1	-8	-15
Circumferential Weld	Position 1.1	41	23
	Position 2.1	1	-7

Notes:

(a) Regulatory Guide 1.99, Revision 2.

(b) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-8

Calculation of Adjusted Reference Temperatures (ARTs) for 15 EFPY [(b)]

PARAMETER	VALUES	
Operating Time	15 EFPY	
Material - Intermediate Shell Plate	B9004-1	B9004-1
Location	1/4T ART	3/4T ART
Chemistry Factor, CF (°F)	44.0	44.0
Fluence, (f), (10^{19} n/cm^2)[(a)]	1.13	0.439
Fluence Factor, FF	1.03	0.771
ΔRT_{NDT} = CF x FF(°F)	45.5	33.9
Initial RT_{NDT}, I(°F)	60	60
Margin, M(°F)	34	33.9
ART, per Regulatory Guide 1.99, Revision 2	140	128

Notes:

(a) Fluence (f), is based upon f_{surf} (10^{19} n/cm^2, E > 1.0 MeV) = 1.81 at 15 EFPY.

(b) See Section 4.2.1.1 for a discussion of EFPY.

The Beaver Valley Unit 2 reactor vessel wall thickness is 7.875 inches at the beltline region.

Table 4.2-9
Reactor Vessel Toughness Data (Unirradiated)

COMPONENT	CODE NO.	MATERIAL SPEC. NO.	Cu %	Ni %	P %	T_{NDT} °F	50 FT/LB 35 MIL TEMP °F	RT_{NDT} °F	USE FT-LBS.
Closure Head Dome	B9008-1	A533B, CL. 1	.13	.54	.013	-20	50	-10	137
Closure Head Flange	B9002-1	A508, CL. 2	---	.74	.012	-10	<40	-10	136
Vessel Flange	B9001-1	A508, CL. 2	---	.73	.010	0	<10	0	132.5
Inlet Nozzle	B9011-1	A508, CL. 2	---	.88	.006	0	<10	0	104
Inlet Nozzle	B9011-2	A508, CL. 2	---	.88	.010	10	<10	10	115
Inlet Nozzle	B9011-3	A508, CL. 2	---	.84	.009	20	<40	20	122
Outlet Nozzle	B9012-1	A508, CL. 2	---	.71	.007	-10	< 0	-10	137
Outlet Nozzle	B9012-2	A508, CL. 2	---	.74	.006	-10	< 0	-10	121
Outlet Nozzle	B9012-3	A508, CL. 2	---	.68	.008	-10	< 0	-10	112
Nozzle Shell	B9003-1	A533B, CL. 1	.13	.61	.008	-10	110	50	91
Nozzle Shell	B9003-2	A533B, CL. 1	.12	.58	.009	0	120	60	79.5
Nozzle Shell	B9003-3	A533B, CL. 1	.13	.61	.008	-10	110	50	97.5
Inter. Shell	B9004-1	A533B, CL. 1	.07	.53	.010	0	120	60	83
Inter. Shell	B9004-2	A533B, CL. 1	.07	.59	.007	-10	100	40	75.5
Lower Shell	B9005-1	A533B, CL. 1	.08	.59	.009	-50	88	28	82
Lower Shell	B9005-2	A533B, CL. 1	.07	.58	.009	-40	93	33	77.5
Bottom Head Torus	B9010-1	A533B, CL. 1	.15	.49	.007	-30	56	-4	97
Bottom Head Dome	B9009-1	A533B, CL. 1	.14	.53	.007	-30	35	-25	116
Weld (Inter. & Lower Shell Long. Seams & Girth Seam)*			.08	.07	.008	-30	<30	-30	144.5
HAZ (Plate B9004-2)			---	---	----	-80	40	-20	76

* Same heat of wire and lot of flux used in all seams including surveillance weldment.

(1) For evaluation of Inservice Reactor Vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter 92-01 or equivalent document are applicable.

(2) See Section 4.2.1.1 for a discussion of EFPY.

Table 4.2-10

RT_{PTS} Calculation for Beltline Region Materials at EOL (32 EFPY) [(d)]

Material	Method	f[(a)] Fluence	FF[(b)]	CF (°F)	ΔRT_{PTS} (°F)	Margin (°F)	$RT_{NDT(U)}$ [(c)] (°F)	RT_{PTS} (°F)
Intermediate Shell Plate B9004-1	RG 1.99, R2, P1.1	3.85	1.348	44.0	54.6	34	60	149
Intermediate Shell Plate B9004-2	RG 1.99, R2, P1.1	3.85	1.348	37.0	49.9	34	40	124
	RG 1.99, R2, P2.1	3.85	1.348	35.3	47.6	17	40	105
Lower Shell Plate B9005-1	RG 1.99, R2, P1.1	3.85	1.348	51.0	68.8	34	28	131
Lower Shell Plate B9005-2	RG 1.99, R2, P1.1	3.85	1.348	44.0	59.3	34	33	126
Circumferential Weld	RG 1.99, R2, P1.1	3.85	1.348	34.1	45.9	45.9	-30	62
	RG 1.99, R2, P2.1	3.85	1.348	15.2	20.5	20.5	-30	11
Longitudinal Weld	RG 1.99, R2, P1.1	1.21	1.053	34.1	35.9	35.9	-30	42
	RG 1.99, R2, P2.1	1.21	1.053	15.2	16.0	16.0	-30	2

Notes:

(a) f = peak clad/base metal interface fluence (10^{19} n/cm^2, E>1.0 MeV) at 32 EFPY (45° fluence for longitudinal welds)

(b) $FF = f^{(0.28 - 0.10 \log f)}$

(c) $RT_{NDT(U)}$ values are measured values.

(d) See Section 4.2.1.1 for a discussion of EFPY.

ATTACHMENT D-1

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 295

The following is a list of the affected Bases pages.
These pages are included for information only.

Affected Page	Pending LARs
B 3/4 1-2a	
B 3/4 4-1	292
B 3/4 4-5	292
B 3/4 4-6	292
B 3/4 4-6a	292
B 3/4 4-7	292
B 3/4 4-7a	292
B 3/4 4-7b	292
B 3/4 4-7c	292
B 3/4 4-8	292
B 3/4 4-8a	292
B 3/4 4-9	292
B 3/4 4-10	292
B 3/4 4-10a	292
B 3/4 4-10b	
B 3/4 4-10c	292
B 3/4 4-10d	292
B 3/4 4-10e	292
B 3/4 4-10g	292
B 3/4 4-10h	292
B 3/4 4-10j	292
B 3/4 4-11b	
B 3/4 5-1a	
B 3/4 5-2	

3/4.1.2 BORATION SYSTEMS (Continued)

The minimum required volume of water for the Refueling Water Storage Tank (RWST) provides: 1) a source of water and Net Positive Suction Head (NPSH) for High Head Safety Injection and Low Head Safety Injection (LHSI), 2) adequate sump water for LHSI and Recirculation Spray Pump NPSH, and 3) water for containment Quench Spray. Specifically, the limiting case for defining the minimum RWST volume is derived from the containment analysis for subatmospheric peak pressure during a Reactor Coolant Pump suction Large Break Loss of Coolant Accident. The minimum volume corresponds to 439,050 total gallons as contained in the RWST. From this total volume, the analysis value of 430,500 gallons is considered to be delivered to the respective systems.

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analysis.

The limitations for a maximum of one centrifugal charging pump to be OPERABLE and the surveillance requirement to verify all charging pumps except the required OPERABLE pump to be inoperable less than or equal to the enable temperature specified in the PTLR ~~set forth in Specification 3.4.9.3~~ provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Substituting a low head safety injection pump for a charging pump in MODES 5 and 6 will not increase the probability of an overpressure event since the shutoff head of the low head safety injection pumps is less than or equal to the setpoint of the overpressure protection system.

Isolation of the primary grade water flow path during MODES 4, 5 and 6 precludes an unplanned boron dilution at these conditions since the sole source of unborated water to the charging pumps is isolated. This eliminates the design basis boron dilution event in MODES 4, 5 and 6. During planned boron dilution events, operator attention will be focused on the boron dilution process and any inappropriate blender operation would be readily identified through various indications which includes the output from the source range nuclear instrumentation.

This page contains changes proposed by LAR 292.

3/4.4.1.1, 2, 3 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In Modes 1 and 2, with one reactor coolant loop not in operation, THERMAL POWER is restricted to less than or equal to 31 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the design DNBR limit. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more non-isolated RCS cold legs less than or equal to the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary side water temperature of each steam generator in a non-isolated loop is less than 50°F above each of the non-isolated RCS cold leg temperatures. The secondary side water temperature is to be verified by direct measurements of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping. This shall be determined within 10 minutes prior to starting the first reactor coolant pump.

3/4.4.8 SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 0.20 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.20 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limits shown on Figure 3.4-1 must be restricted to ensure that assumptions made in the UFSAR accident analyses are not exceeded.

Reducing Tavg to < 500°F minimizes the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. This action also reduces the pressure differential across the steam generator tubes reducing the probability and magnitude of main steam line break accident induced primary-to-secondary leakage. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak testing, and data for the maximum rate of change of reactor coolant temperature. The analytical methods used to determine the RCS P/T limits and the OPPS limits (PORV pressure relief setpoint and OPPS enable temperature) were developed in accordance with WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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

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

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

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185-82 and Appendix H of 10 CFR 50. The operating PTLR P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, Rev. 2.

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

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

The criticality limit curve includes the 10 CFR 50,Appendix G requirement that it be ≥ 40°F above the heatup curve and not less than the minimum permissible temperature for inservice hydrostatic testing. However, the PTLR criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.5, "RCS Minimum Temperature for Criticality."

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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

~~During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.~~

This page contains changes proposed by LAR 292.

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

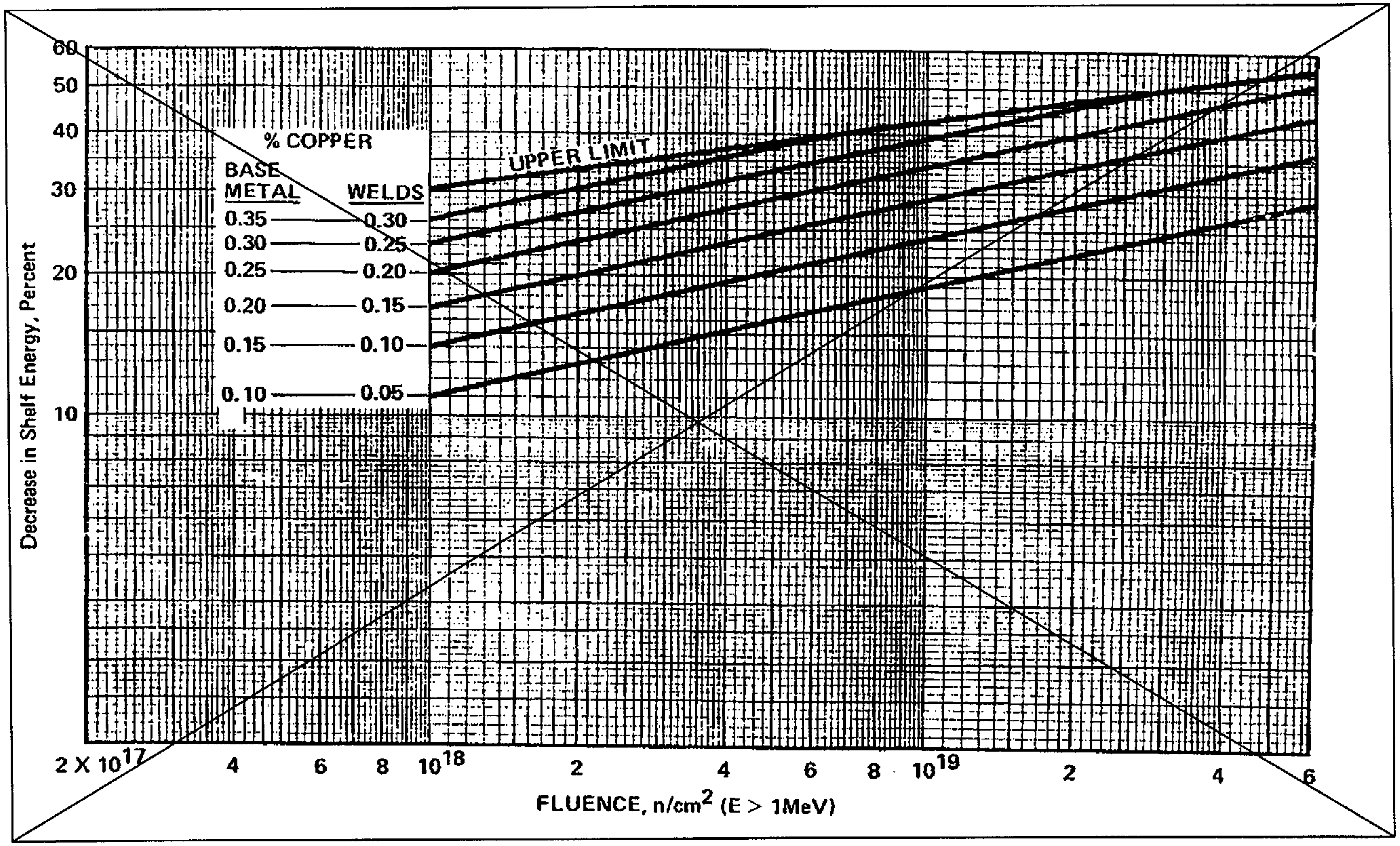
~~The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.~~

~~The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curves, Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the time in life indicated on the respective curves.~~

~~The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and nickel content of the material in question, can be predicted using WCAP-15570, Rev. 2 and Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT}.~~

~~The heatup and cooldown curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2 and no longer contain the additional margin of 10°F and 60 psig for instrument error previously incorporated in these curves.~~

This page contains changes proposed by LAR 292.



~~FIGURE B 3/4 4-1~~

~~PREDICTED DECREASE IN SHELF ENERGY AS A FUNCTION OF COPPER CONTENT AND FLUENCE~~

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Moved to PTLR.

~~TABLE B 3/4.4-1~~

~~REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)~~

~~Component~~	~~Heat No.~~	~~Code No.~~	~~Material Type~~	~~Cu (%)~~	~~Ni (%)~~	~~P (%)~~	~~T_{NDT} (°F)~~	~~RT_{NDT} (°F)~~	~~Upper Shelf Energy (Ft-lb) MWD~~	~~NMWD~~
~~Closure Head Dome~~	~~C6213-1B~~	~~B6610~~	~~A533B CL. 1~~	~~.15~~	—	~~.010~~	~~-40~~	~~0*~~	~~121~~	—
~~Closure Head Seg.~~	~~A5518-2~~	~~B6611~~	~~A533B CL. 1~~	~~.14~~	—	~~.015~~	~~-20~~	~~-20*~~	~~131~~	—
~~Closure Head Flange~~	~~ZV3758~~	—	~~A508 CL. 2~~	~~.08~~	—	~~.007~~	~~60*~~	~~60*~~	~~>100~~	—
~~Vessel Flange~~	~~ZV3661~~	—	~~A508 CL. 2~~	~~.12~~	—	~~.010~~	~~60*~~	~~60*~~	~~166~~	—
~~Inlet Nozzle~~	~~9-5443~~	—	~~A508 CL. 2~~	~~.10~~	—	~~.008~~	~~60*~~	~~60*~~	~~82.5~~	—
~~Inlet Nozzle~~	~~9-5460~~	—	~~A508 CL. 2~~	~~.10~~	—	~~.010~~	~~60*~~	~~60*~~	~~94~~	—
~~Inlet Nozzle~~	~~9-5712~~	—	~~A508 CL. 2~~	~~.08~~	—	~~.007~~	~~60*~~	~~60*~~	~~97~~	—
~~Outlet Nozzle~~	~~9-5415~~	—	~~A508 CL. 2~~	—	—	~~.008~~	~~60*~~	~~60*~~	~~97~~	—
~~Outlet Nozzle~~	~~9-5415~~	—	~~A508 CL. 2~~	—	—	~~.007~~	~~60*~~	~~60*~~	~~112.5~~	—
~~Outlet Nozzle~~	~~9-5444~~	—	~~A508 CL. 2~~	~~.09~~	—	~~.007~~	~~60*~~	~~60*~~	~~103~~	—
~~Upper Shell~~	~~123V339~~	—	~~A508 CL. 2~~	—	—	~~.010~~	~~40~~	~~40*~~	~~155~~	—
~~Inter Shell~~	~~C4381-2~~	~~B6607-2~~	~~A533B CL. 1~~	~~.14~~	~~.62~~	~~.015~~	~~-10~~	~~73~~	~~123~~	~~82.5~~
~~Inter Shell~~	~~C4381-1~~	~~B6607-1~~	~~A533B CL. 1~~	~~.14~~	~~.62~~	~~.015~~	~~-10~~	~~43~~	~~128.5~~	~~90~~
~~Lower Shell~~	~~C6317-1~~	~~B6903-1~~	~~A533B CL. 1~~	~~.20~~	~~.54~~	~~.010~~	~~-50~~	~~27~~	~~134~~	~~80~~
~~Lower Shell~~	~~C6293-2~~	~~B7203-2~~	~~A533B CL. 1~~	~~.14~~	~~.57~~	~~.015~~	~~-20~~	~~20~~	~~129.5~~	~~83.5~~
~~Trans Ring~~	~~123V223~~	—	~~A508 CL. 2~~	—	—	—	~~30~~	~~30*~~	~~143~~	—
~~Bottom Hd Seg~~	~~C4423-3~~	~~B6618~~	~~A533B CL. 1~~	~~.13~~	—	~~.008~~	~~-30~~	~~-29*~~	~~124~~	—
~~Bottom Hd Dome~~	~~C4482-1~~	~~B6619~~	~~A533B CL. 1~~	~~.13~~	—	~~.015~~	~~-50~~	~~-33*~~	~~125.5~~	—
~~Inter to Lower Shell Weld~~	~~90136~~	—	—	~~.27~~	~~.07~~	—	—	~~-56~~	—	~~>100~~
~~Inter Shell Long. Weld~~	~~305424~~	—	—	~~.28~~	~~.63~~	—	—	~~-56~~	—	~~>100~~
~~Lower Shell Long. Weld~~	~~305414~~	—	—	~~.34~~	~~.61~~	—	—	~~-56~~	—	~~>100~~
~~Weld HAZ~~				—	—	—	~~-40~~	~~-40~~	—	~~136.5~~

~~*Estimated Per NRC Standard Review Plan Branch Technical Position MTEB 5-2~~

~~MWD Major Working Direction~~

~~NMWD Normal to Major Working Direction~~

~~Note: For evaluation of Inservice Reactor Vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter 92-01 or equivalent document are applicable.~~

This page contains changes proposed by LAR 292.

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

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

~~RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Regulatory Guide 1.99 Revision 2 curve which shows the effect of fluence and copper content on upper shelf energy (USE) for reactor vessel steels are shown in Figure B 3/4.4-1.~~

~~Given the copper and nickel contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be predicted by the equation: $\Delta RT_{NDT} = (CF) f^{(0.28-0.1 \log f)}$ where f = fluence and CF = chemistry factor, a function of copper and nickel. Fast-neutron fluence (E > 1 Mev) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations can be generated as a function of full-power service life. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The preirradiation fracture-toughness properties of the Beaver Valley Unit 1 reactor vessel materials are presented in Table B 3/4.4-1. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review plan.[1] The postirradiation fracture toughness properties of the reactor vessel beltline material, determined in accordance with 1996 Addenda to ASME Section XI, Appendix G and ASME Code Case N-640, were obtained directly from the Beaver Valley Unit 1 Vessel Material Surveillance Program.~~

~~Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. Adjusted Reference Temperature (ART), defined as ART = initial~~ ~~RT_{NDT}~~ ~~+ Margins for uncertainties +~~ ~~ΔRT_{NDT}~~~~, is used to index the material to the~~ ~~K_{IC}~~ ~~curve and, in turn, to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.~~

The pressure-temperature limit curves contained in the PTLR, are developed using ASME Code Case N-640. One of the safety margins incorporated into the curves is the lower bound fracture toughness curve. The lower bound fracture toughness curves available in Appendix G to ASME Section XI use the reference stress intensity factor K_{IA}. The pressure-temperature limit curves based on Code Case N-640 use the reference stress intensity factor K_{IC}. K_{IA} is a fracture toughness curve which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC} is a fracture toughness curve which is a lower bound on static fracture toughness only. The only change that is made when generating the revised pressure-temperature limits curve with K_{IC} is the lower bound fracture toughness curve selected. All other margins involved in the generation process remain unchanged. Since the heatup and cooldown process is a very slow one, with the fastest rate allowed being 100°F per hour, the rate of change of pressure and temperature is considered constant so that the stress is essentially constant. Both heatup and cooldown correspond to

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~~1. "Fracture Toughness Requirements," Branch Technical Position MTEB No. 5-2, Section 5.3.2-14 in Standard Review Plan, NUREG-75/087, 1975.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

static loading, with regard to fracture toughness. The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a pressurized thermal shock event, but not during heatup and cooldown. Therefore, the static toughness K_{IC} lower bound toughness is used to generate the pressure-temperature limit curves contained in the PTLR.

~~The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup and cooldown cannot be greater than the reference stress intensity factor, K_{IC}, for the metal temperature at that time. K_{IC} is obtained from the reference fracture toughness curve, defined in Appendix G (Code Case N-640) to the ASME Code.[2] The K_{IC} curve is given by the equation:~~

$$\cancel{K_{IC} = 33.2 + 20.734 * e^{[0.02(T-RT_{NDT})]}} \qquad \cancel{(4-1)}$$

~~where K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nilductility temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G (Code Case N-640) to the ASME Code[2] as follows:~~

$$\cancel{C\ K_{IM} + K_{It} \leq K_{IC}} \qquad \cancel{(4-2)}$$

~~2. ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendices, Rules for Construction of Nuclear Vessels," Appendix G. "Protection Against Nonductile Failure," pp. 461-469, 1980 Edition, American Society of Mechanical Engineers, New York, 1980.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~where:~~

~~K_{IM} is the stress intensity factor caused by membrane (pressure) stress~~

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

~~K_{IC} is a function of temperature relative to the RT_{NDT} of the material~~

~~C = 2.0 for Level A and Level B service limits~~

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

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

~~Cooldown~~

~~For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.~~

~~The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~It follows that, at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IC} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IC} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.~~

~~The above procedures are needed because there is no direct control on temperature at the 1/4T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and insures conservative operation of the system for the entire cooldown period.~~

~~Heatup~~

~~Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IC} for the 1/4T crack during heatup is lower than the K_{IC} for the 1/4T crack during steady-state conditions at the same coolant temperature.~~

~~During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{IC}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.~~

This page contains changes proposed by LAR 292.

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.~~

~~Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves, as documented in WCAP-15570, Revision 2, are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.~~

~~The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.~~

~~The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing. These pressure-temperature limits lines on Figures 3.4-2 and 3.4-3 for boltup temperature are provided to ensure compliance with the minimum temperature requirements of Appendix G to ASME Section XI for vessel closure head flange boltup. It recommends that when the flange and adjacent shell region are stressed by the full intended bolt preload the minimum metal temperature in the stressed region is at least the initial RT_{NDT} temperature for the material in the stressed regions.~~

This page contains changes proposed by LAR 292.

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4.5-3 to assure compliance with the requirements of Appendix H to 10 CFR 50.~~

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

~~Pressure-temperature limit curves shown in Figure B 3/4 4-2 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and ASME Code Case N-640.~~

Moved to PTLR.

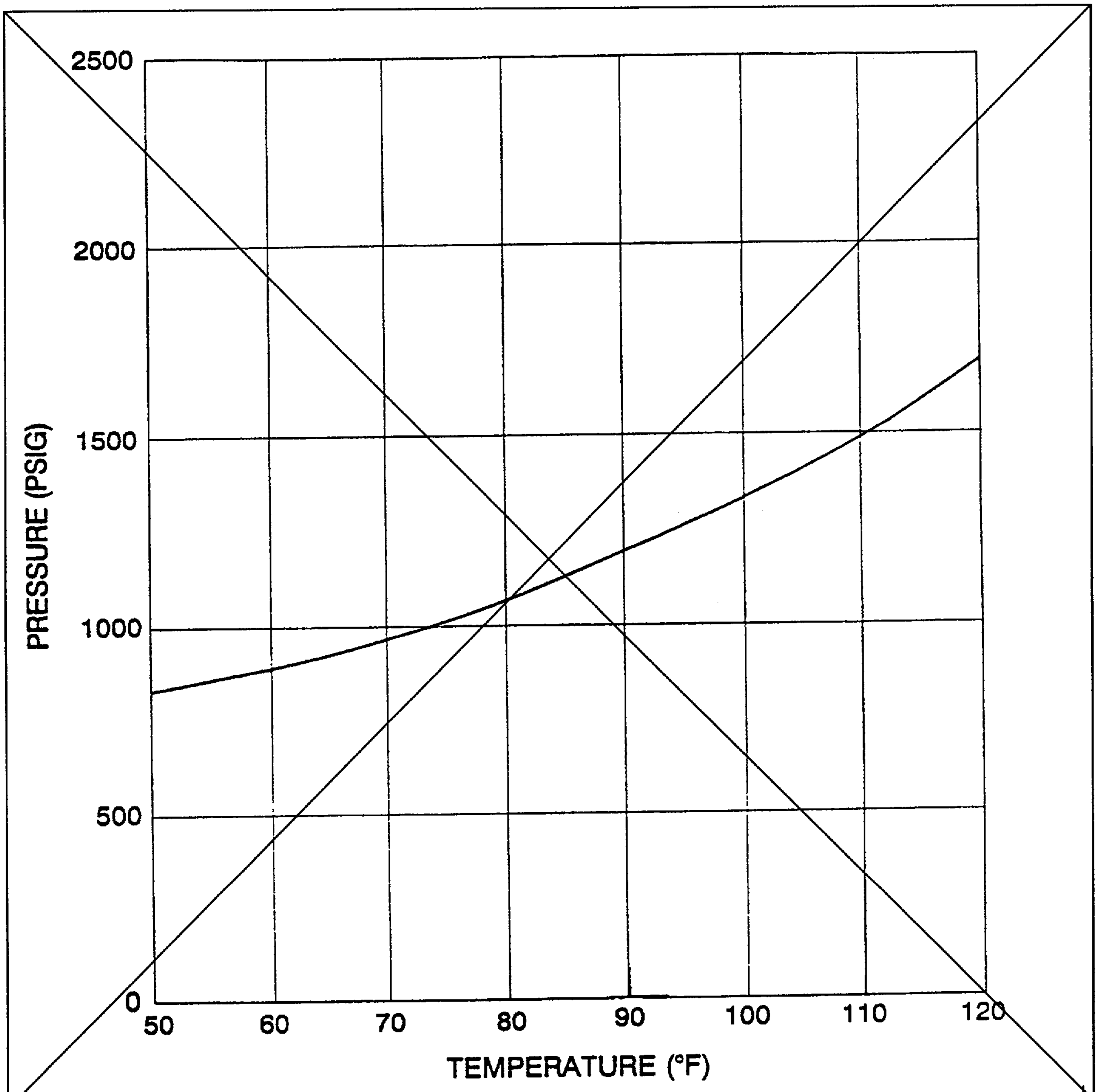
OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G (including ASME Code Case N-640) requirements during the OPPS MODES.

Moved to PTLR.

This page contains changes proposed by LAR 292.



~~FIGURE B 3/4 4-2~~

~~ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE~~

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

BACKGROUND (Continued)

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires deactivating all but one charging pump and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the OPPS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve and, if needed, until the charging pump is actuated by SI.

The OPPS for pressure relief consists of two PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

PORV REQUIREMENTS

As designed for the OPPS System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the OPPS actuation circuit. The OPPS actuation circuit monitors RCS pressure and determines when a condition not acceptable with respect to the PTLR is approached. If the indicated pressure meets or exceeds the OPPS actuation setpoint, a PORV is signaled to open. Having the setpoints of both valves within the limits ensures that the Appendix G limits will not be exceeded in any analyzed event. When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

The low limit on pressure during the transient is typically established based solely on an operational consideration for the Reactor Coolant Pump (RCP) No. 1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. The upper limit (based on the minimum of the steady-state 10 CFR 50 Appendix G requirement and the PORV piping limitations) and the RCP No. 1 seal performance criteria create a pressure range from which the setpoints for both PORVs are selected. When there is insufficient range between the upper and lower pressure limits to select the PORV setpoints to provide protection against violating both limits, setpoint selection to provide protection against the upper limit violation takes precedence.

RCS VENT REQUIREMENTS

Once the RCS is depressurized, a vent exposed to the pressurizer relief tank (PRT) or containment atmosphere will maintain the RCS pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting OPPS mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it may be satisfied by removing a pressurizer safety valve or establishing an opening between the RCS and the PRT or containment atmosphere of the required size through any positive means available which cannot be inadvertently defeated. The vent must be above the level of reactor coolant, so as not to drain the RCS when open.

This page contains changes proposed by LAR 292.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

APPLICABLE SAFETY ANALYSES

Safety analyses demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits when low RCS temperature conditions exist. At the enable temperature specified in the PTLR and below, overpressure prevention is provided by two OPERABLE RCS relief valves or a depressurized RCS and a sufficient sized RCS vent.

The actual temperature at which the pressure in the PTLR P/T limit curve falls below the OPPS setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR P/T limit~~heatup and cooldown~~ curves are revised, the OPPS must be re-evaluated to ensure its functional requirements can still be met.

The PTLR contains the acceptance limits that define the OPPS requirements.~~The heatup and cooldown curves represent the Appendix G (including ASME Code Case N-640) limits that define OPPS operation. Setpoint calculations correlated to RCS temperature define acceptable OPPS setpoints for steady-state pressure-temperature limits based on Revision 2 of NRC Regulatory Guide 1.99.~~ Any change to the RCS that may affect OPPS operation must be evaluated against the analyses to determine the impact of the change on the OPPS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

MASS INPUT TYPE TRANSIENTS

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

HEAT INPUT TYPE TRANSIENTS

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

This page contains changes proposed by LAR 292.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEAT INPUT TYPE TRANSIENTS (Continued)

The following are required during the OPPS MODES to ensure that mass and heat input transients do not occur, which either of the OPPS overpressure protection means cannot handle:

a. Deactivating all but one OPERABLE charging pump, except during pump swapping operations as addressed in the LCO;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Meeting the secondary side water to RCS cold leg temperature difference requirement specified in LCO 3.4.1.3, "Reactor Coolant System - Shutdown."

The analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain the RCS pressure below the limits when only one charging pump is actuated by SI. Thus, the LCO allows only one charging pump OPERABLE during the OPPS MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

The isolated accumulators must have their discharge valves closed with power removed. Fracture mechanics analyses established the temperature of OPPS Applicability at the enable temperature specified in the PTLR.

PORV PERFORMANCE

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit specified in the PTLR. The setpoint is derived by analyses that model the performance of the OPPS assuming the limiting OPPS transient of SI actuation of one charging pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the P/T limits will be met.

The Maximum Allowable Nominal PORV Setpoint for the OPPS is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum allowable nominal setpoint ensures that 10 CFR 50 Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

LCO (Continued)

To limit the coolant input capability, the LCO requires that a maximum of one charging pump be capable of injecting into the RCS and all accumulator discharge isolation valves be closed and immobilized. The LCO is qualified by a note that permits two pumps capable of RCS injection for less than or equal to 1 hour to allow for pump swaps.

The LCO is also qualified by a note stating that accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or at the maximum RCS pressure for the existing temperature, as allowed by the PTLR P/T limit curves. This note permits the accumulator discharge isolation valve surveillance to be performed only under these pressure and temperature conditions.

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

a. Two OPERABLE PORVs; a PORV is OPERABLE for OPPS when its block valve is open, its lift setpoint is set to the limit and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits; or

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of 2.07 square inches.

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

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. When the reactor vessel head is off, overpressurization cannot occur.

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

ACTION

a. With two or more charging pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

b. An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the PTLR P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, the ACTION provides two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to more than the enable temperature specified in the PTLR, the accumulator pressure cannot exceed the OPPS limits if the accumulators are fully injected. Depressurizing the accumulators below the OPPS limit specified in the PTLR also gives this protection.

The completion times are based on operating experience that these activities can be accomplished in these time periods indicating that an event requiring OPPS is not likely in the allowed times.

c. In MODE 4 when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a completion time of 7 days. Two RCS relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

The completion time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. If plant operation results in transitioning to MODE 5, the completion time to restore an inoperable PORV may not exceed 7 days as required by this ACTION.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

The frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 4.4.9.3.1.b allows opening the accumulator discharge isolation valves to perform accumulator discharge check valve testing.

SR 4.4.9.3.2

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

The SR is required to be performed prior to entering the condition for the OPPS to be OPERABLE. This assures low temperature overpressure protection is available when the RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR. Performing the surveillance every 31 days on each required PORV permits verification and adjustment, if necessary, of its lift setpoint, and considers instrumentation reliability which has been shown through operating experience to be acceptable. The CHANNEL FUNCTIONAL TEST will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required.

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

3/4.4.11 RELIEF VALVES (Continued)

APPLICABILITY

In MODES 1, 2, and 3, the PORV and its block valve are required to be OPERABLE to limit the potential for a small break LOCA through the flow path. The most likely cause for a PORV small break LOCA is a result of a pressure increase transient that causes the PORV to open. Imbalances in the energy output of the core and heat removal by the secondary system can cause the RCS pressure to increase to the PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for OPPS in MODES 4 (below the enable temperature specified in the PTLR), 5, and 6 with the reactor vessel head in place. LCO 3.4.9.3 addresses the PORV requirements in these MODES.

ACTION

A General Note provides clarification that all pressurizer PORVs and block valves are treated as separate entities, each with separate completion times (i.e., the completion time is on a component basis).

a. With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, it may be able to be manually opened and closed, and therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems related to PORV accident monitoring instruments identified in LCO 3.3.3.8, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. If the position indication is inoperable, then the PORVs are inoperable. For these reasons, the block valve shall be closed but the ACTION requires power be maintained to the valve. Automatic control problems and related instrumentation problems would not render the PORVs inoperable. Accident analyses assume manual operation of

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point on the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the ECCS Flow Analysis. The term "required developed head" refers to the pump performance at a given flow point that is assumed in the ECCS Flow Analysis. This is possible since the analysis assumes the pump delivers different flows at different times during accident mitigation. These multiple points are represented by a curve. The values at various flow points are defined by the Minimum Operating Point (MOP) curve in the Inservice Testing (IST) Program. The verification that the pump's developed head at the flow test point is greater than or equal to the required developed head is performed by using the MOP curve. Surveillance requirements are specified in the IST Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

The limitation for a maximum of one charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable ≤ the enable temperature specified in the PTLR ~~set forth in Specification 3.4.9.3~~ provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

3/4.5.4 BORON INJECTION SYSTEM

The OPERABILITY of the boron injection system as part of the ECCS ensures that sufficient negative reactivity is injected into the core to limit any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident or a steam line rupture.

The boron injection tank is required to be isolated when RCS temperature is ≤ the enable temperature specified in the PTLR ~~set forth in Specification 3.4.9.3~~ to prevent a potential overpressurization due to an inadvertent safety injection signal.

The analysis of a main steam pipe rupture is performed to demonstrate that the following criteria are satisfied:

1. Assuming a stuck rod cluster control assembly, with or without offsite power, and assuming a single failure in the engineered safeguards, there is no consequential damage to the primary system and the core remains in place and intact.

2. Energy release to containment from the worst steam pipe break does not cause failure of the containment structure.

3. Radiation doses are not expected to exceed the guidelines of the 10 CFR 100.

The limits on injection tank minimum volume and boron concentration ensure that the assumptions used in the steam line break analysis are met.

ATTACHMENT D-2

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 167

The following is a list of the affected Bases pages.
These pages are included for information only.

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3/4.1.2 BORATION SYSTEMS (Continued)

The OPERABILITY of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST minimum volume and boron concentration ensure that: 1) sufficient water is available within containment to permit recirculation cooling flow to the core, 2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area > 3.0 ft^2) assuming complete mixing of the RWST, RCS, ECCS, chemical addition tank, containment spray system piping, and other water volumes that may eventually reside in the sump Post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and to preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The limitations for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below the enable temperature specified in the PTLR~~350°F~~ provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV. Substituting a Low Head Safety Injection pump for a charging pump in MODES 5 and 6 will not increase the probability of an overpressure event since the shutoff head of the Low Head Safety Injection pumps is below the setpoint of the overpressure protection system.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.77% Δk/k after xenon decay and cooldown to 200°F. The maximum boration capability requirements occur at BOL from full power peak xenon conditions and requires 13,390 gallons of 7000 ppm borated water from the boric acid storage tanks or 100,000 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 350°F, one boron injection flow path is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

This page contains changes proposed by LAR 157.

3/4.4.1.1, 2, 3 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In MODES 1 and 2, with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event when the rod control system is capable of control bank rod withdrawal.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more non-isolated RCS cold legs less than or equal to the enable temperature specified in the PTLR~~set forth in Specification 3.4.9.3~~ are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary side water temperature of each steam generator in a non-isolated loop is less than 50°F above each of the non-isolated RCS cold leg temperatures. The secondary side water temperature is to be verified by direct measurements of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping. This shall be determined within 10 minutes prior to starting a reactor coolant pump.

3/4.4.8 SPECIFIC ACTIVITY (Continued)

relief valves. This action also reduces the pressure differential across the steam generator tubes reducing the probability and magnitude of main steam line break accident induced primary-to-secondary leakage. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak testing, and data for the maximum rate of change of reactor coolant temperature. The analytical methods used to determine the RCS P/T limits and the OPPS limits (PORV pressure relief setpoint and OPPS enable temperature) were developed in accordance with WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

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

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

10 CFR 50, Appendix G, requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. It also requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and

system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

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

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185-82 and Appendix H of 10 CFR 50. The operating PTLR P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, Rev. 2.

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

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

The criticality limit curve includes the 10 CFR 50,Appendix G requirement that it be ≥ 40°F above the heatup curve and not less than the minimum permissible temperature for inservice hydrostatic testing. However, the PTLR criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.5, "RCS Minimum Temperature for Criticality."

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

~~During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.~~

~~The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall~~

~~of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figures 3.4-3 (Sheets 1 through 5), are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 15 EFPY.~~

~~The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content and nickel content of the material in question, can be predicted using WCAP-15139 and Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 (Sheets 1 through 5), include predicted adjustments for this shift in RT_{NDT}.~~

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

~~RT_{NDT} increases as the material is exposed to fast-neutron radiation. Thus, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Regulatory Guide 1.99 Revision 2 curves which show the effect of fluence and copper content on upper shelf energy (USE) for reactor vessel steels are shown in Figure B 3/4 4-1.~~

~~TABLE B 3/4.4-1~~

~~REACTOR VESSEL TOUGHNESS DATA~~~~[(1)]~~

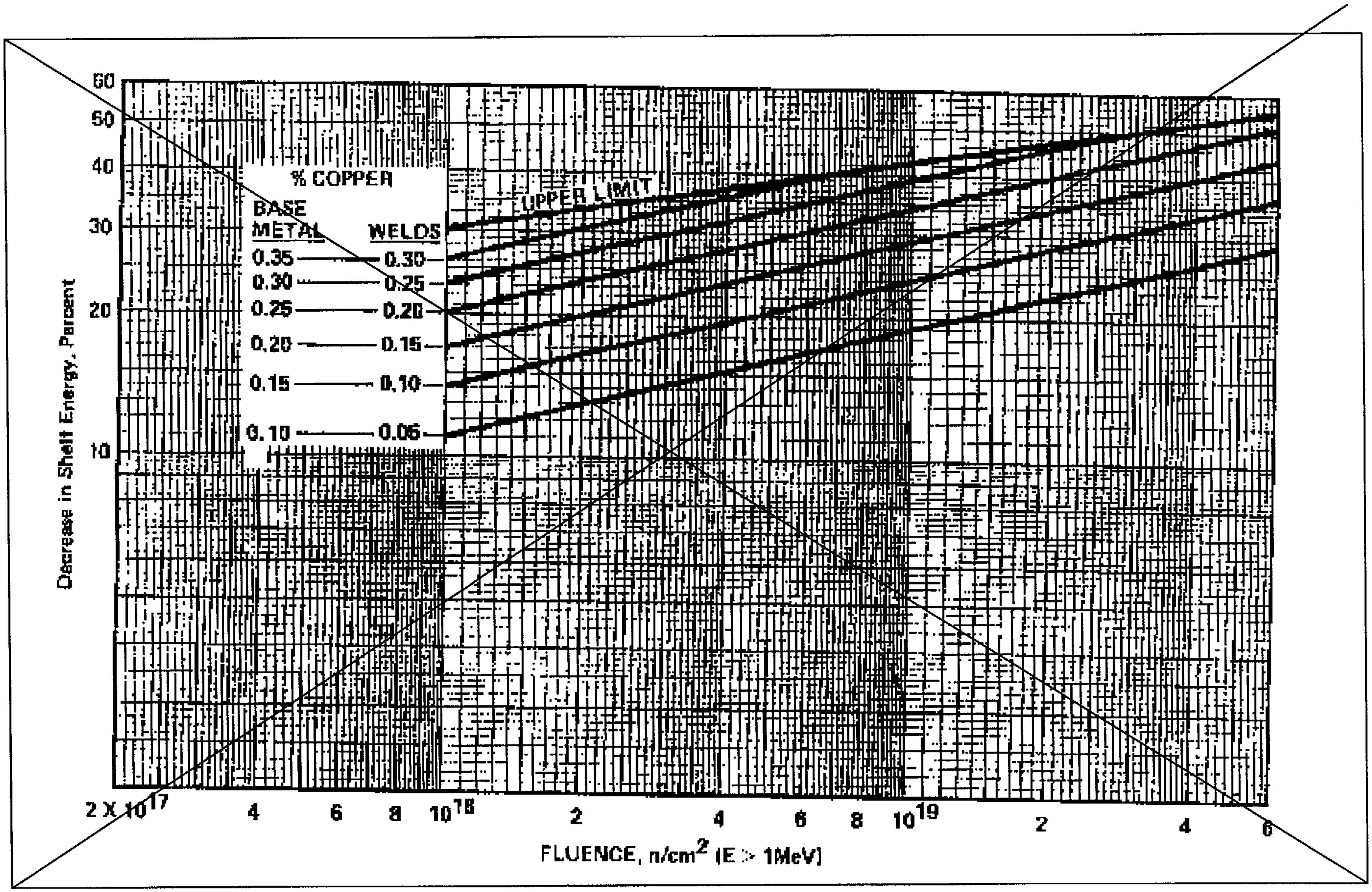
Moved to PTLR.

~~COMPONENT~~	~~CODE NO.~~	~~MATERIAL SPEC. NO.~~	~~Cu %~~	~~Ni %~~	~~P %~~	~~T_{NDT} °F~~	~~50 FT/LB 35 MIL TEMP °F~~	~~RT_{NDT} °F~~	~~USE FT-LBS.~~
~~Closure Head Dome~~	~~B9008-1~~	~~A533B, CL. 1~~	~~.13~~	~~.54~~	~~.013~~	~~20~~	~~50~~	~~10~~	~~137~~
~~Closure Head Flange~~	~~B9002-1~~	~~A508, CL. 2~~	—	~~.74~~	~~.012~~	~~10~~	~~<40~~	~~10~~	~~136~~
~~Vessel Flange~~	~~B9001-1~~	~~A508, CL. 2~~	—	~~.73~~	~~.010~~	~~0~~	~~<10~~	~~0~~	~~132.5~~
~~Inlet Nozzle~~	~~B9011-1~~	~~A508, CL. 2~~	—	~~.88~~	~~.006~~	~~0~~	~~<10~~	~~0~~	~~104~~
~~Inlet Nozzle~~	~~B9011-2~~	~~A508, CL. 2~~	—	~~.88~~	~~.010~~	~~10~~	~~<10~~	~~10~~	~~115~~
~~Inlet Nozzle~~	~~B9011-3~~	~~A508, CL. 2~~	—	~~.84~~	~~.009~~	~~20~~	~~<40~~	~~20~~	~~122~~
~~Outlet Nozzle~~	~~B9012-1~~	~~A508, CL. 2~~	—	~~.71~~	~~.007~~	~~10~~	~~<0~~	~~10~~	~~137~~
~~Outlet Nozzle~~	~~B9012-2~~	~~A508, CL. 2~~	—	~~.74~~	~~.006~~	~~10~~	~~<0~~	~~10~~	~~121~~
~~Outlet Nozzle~~	~~B9012-3~~	~~A508, CL. 2~~	—	~~.68~~	~~.008~~	~~10~~	~~<0~~	~~10~~	~~112~~
~~Nozzle Shell~~	~~B9003-1~~	~~A533B, CL. 1~~	~~.13~~	~~.61~~	~~.008~~	~~10~~	~~110~~	~~50~~	~~91~~
~~Nozzle Shell~~	~~B9003-2~~	~~A533B, CL. 1~~	~~.12~~	~~.58~~	~~.009~~	~~0~~	~~120~~	~~60~~	~~79.5~~
~~Nozzle Shell~~	~~B9003-3~~	~~A533B, CL. 1~~	~~.13~~	~~.61~~	~~.008~~	~~10~~	~~110~~	~~50~~	~~97.5~~
~~Inter. Shell~~	~~B9004-1~~	~~A533B, CL. 1~~	~~.07~~	~~.53~~	~~.010~~	~~0~~	~~120~~	~~60~~	~~83~~
~~Inter. Shell~~	~~B9004-2~~	~~A533B, CL. 1~~	~~.07~~	~~.59~~	~~.007~~	~~10~~	~~100~~	~~40~~	~~75.5~~
~~Lower Shell~~	~~B9005-1~~	~~A533B, CL. 1~~	~~.08~~	~~.59~~	~~.009~~	~~50~~	~~88~~	~~28~~	~~82~~
~~Lower Shell~~	~~B9005-2~~	~~A533B, CL. 1~~	~~.07~~	~~.58~~	~~.009~~	~~40~~	~~93~~	~~33~~	~~77.5~~
~~Bottom Head Torus~~	~~B9010-1~~	~~A533B, CL. 1~~	~~.15~~	~~.49~~	~~.007~~	~~30~~	~~56~~	~~4~~	~~97~~
~~Bottom Head Dome~~	~~B9009-1~~	~~A533B, CL. 1~~	~~.14~~	~~.53~~	~~.007~~	~~30~~	~~35~~	~~25~~	~~116~~
~~Weld (Inter. & Lower Shell Long. Seams & Girth Seam)*~~			~~.08~~	~~.07~~	~~.008~~	~~30~~	~~<30~~	~~30~~	~~144.5~~
~~HAZ (Plate B9004-2)~~			—	—	—	~~80~~	~~40~~	~~20~~	~~76~~

~~* Same heat of wire and lot of flux used in all seams including surveillance weldment.~~

~~(1) For evaluation of Inservice Reactor Vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter 92-01 or equivalent document are applicable.~~

(Proposed Wording)



~~FIGURE B 3/4 4-1~~

~~PREDICTED DECREASE IN SHELF ENERGY AS A FUNCTION OF COPPER CONTENT AND FLUENCE~~

(Proposed Wording)

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~Given the copper and nickel contents of the most limiting material, the radiation-induced ΔRT_{NDT} can be predicted by the equation: ΔRT_{NDT} = $(CF) f^{(0.28 - 0.1 \log f)}$. Fast-neutron fluence (E > 1 Mev) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations can be generated as a function of full-power service life. The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to RT_{NDT}.~~

~~The preirradiation fracture-toughness properties of the Beaver Valley Unit 2 reactor vessel materials are presented in Table B 3/4.4-1. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the 1972 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code.~~

The pressure-temperature limit curves contained in the PTLR, are developed using Code Case N-640. One of the safety margins incorporated into the curves is the lower bound fracture toughness curve. The lower bound fracture toughness curves available in Appendix G to Section XI use the reference stress intensity factor K_{IA}. The pressure-temperature limit curves based on Code Case N-640 use the reference stress intensity factor K_{IC}. K_{IA} is a fracture toughness curve which is a lower bound on all static, dynamic and arrest fracture toughness, and K_{IC} is a fracture toughness curve which is a lower bound on static fracture toughness only. The only change that is made when generating the revised pressure-temperature limits curve with K_{IC} is the lower bound fracture toughness curve selected. All other margins involved in the generation process remain unchanged. Since the heatup and cooldown process is a very slow one, with the fastest rate allowed being 100°F per hour, the rate of change of pressure and temperature is considered constant so that the stress is essentially constant. Both heatup and cooldown correspond to static loading, with regard to fracture toughness. The only time when dynamic loading can occur and where the dynamic/arrest toughness K_{IA} should be used for the reactor pressure vessel is when a crack is running. This might happen during a pressurized thermal shock event, but not during heatup and cooldown. Therefore, the static toughness K_{IC} lower bound toughness is used to generate the pressure-temperature limit curves contained in the PTLR.

~~The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup and cooldown cannot be greater than the reference stress intensity factor, K_{IC}, for the metal temperature at~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~that time. K_{IC} is obtained from the reference fracture toughness curve, defined by Code Case N-640. The K_{IC} curve is given by the equation:~~

$$\cancel{K_{IC} = 33.2 + 20.734 * e^{[0.02(T-RT_{NDT})]}} \qquad \cancel{(4-1)}$$

~~where K_{IC} is the reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature, RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined by Code Case N-640 as follows:~~

$$\cancel{C\,K_{IM} + K_{It} \leq K_{IC}} \qquad \cancel{(4-2)}$$

~~where~~

- ~~K_{IM} is the stress intensity factor caused by membrane (pressure) stress~~
- ~~K_{It} is the stress intensity factor caused by the thermal gradients~~
- ~~K_{IC} is a function of temperature relative to the RT_{NDT} of the material~~
- ~~C = 2.0 for Level A and Level B service limits~~
- ~~C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical~~

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

~~Cooldown~~

~~For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increases with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4 T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady-state situation. It follows that, at any given reactor coolant temperature, the~~ $\cancel{\Delta T}$ ~~developed during cooldown results in a higher value of~~ $\cancel{K_{IC}}$ ~~at the 1/4 T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in~~ $\cancel{K_{IC}}$ ~~exceeds~~ $\cancel{K_{IT}}$~~, the calculated allowable pressure during cooldown will be greater than the steady-state value.~~

~~The above procedures are needed because there is no direct control on temperature at the 1/4 T location and, therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and insures conservative operation of the system for the entire cooldown period.~~

~~Heatup~~

~~Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the~~ $\cancel{K_{IC}}$ ~~for the 1/4 T crack during heatup is lower than the~~ $\cancel{K_{IC}}$ ~~for the 1/4 T crack during steady-state conditions at the same coolant temperature.~~

~~During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower~~ $\cancel{K_{IC}}$~~'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.~~

~~Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves, as documented in WCAP-15139, are produced as follows: A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.~~

~~The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.~~

~~The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing. These pressure-temperature limit lines on Figures 3.4-2 and 3.4-3 (Sheets 1 through 5) for boltup temperature are provided to ensure compliance with the minimum temperature requirements of Appendix G to ASME Section XI for~~

~~3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)~~

~~vessel closure head flange boltup. It recommends that when the flange and adjacent shell region are stressed by the full intended bolt preload the minimum metal temperature in the stressed region is at least the initial RT_{NDT} temperature for the material in the stressed regions.~~

~~The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 5.3-6 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.~~

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

~~Pressure temperature limit curves shown in Figure B 3/4 4-2 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N 640.~~

Moved to PTLR.

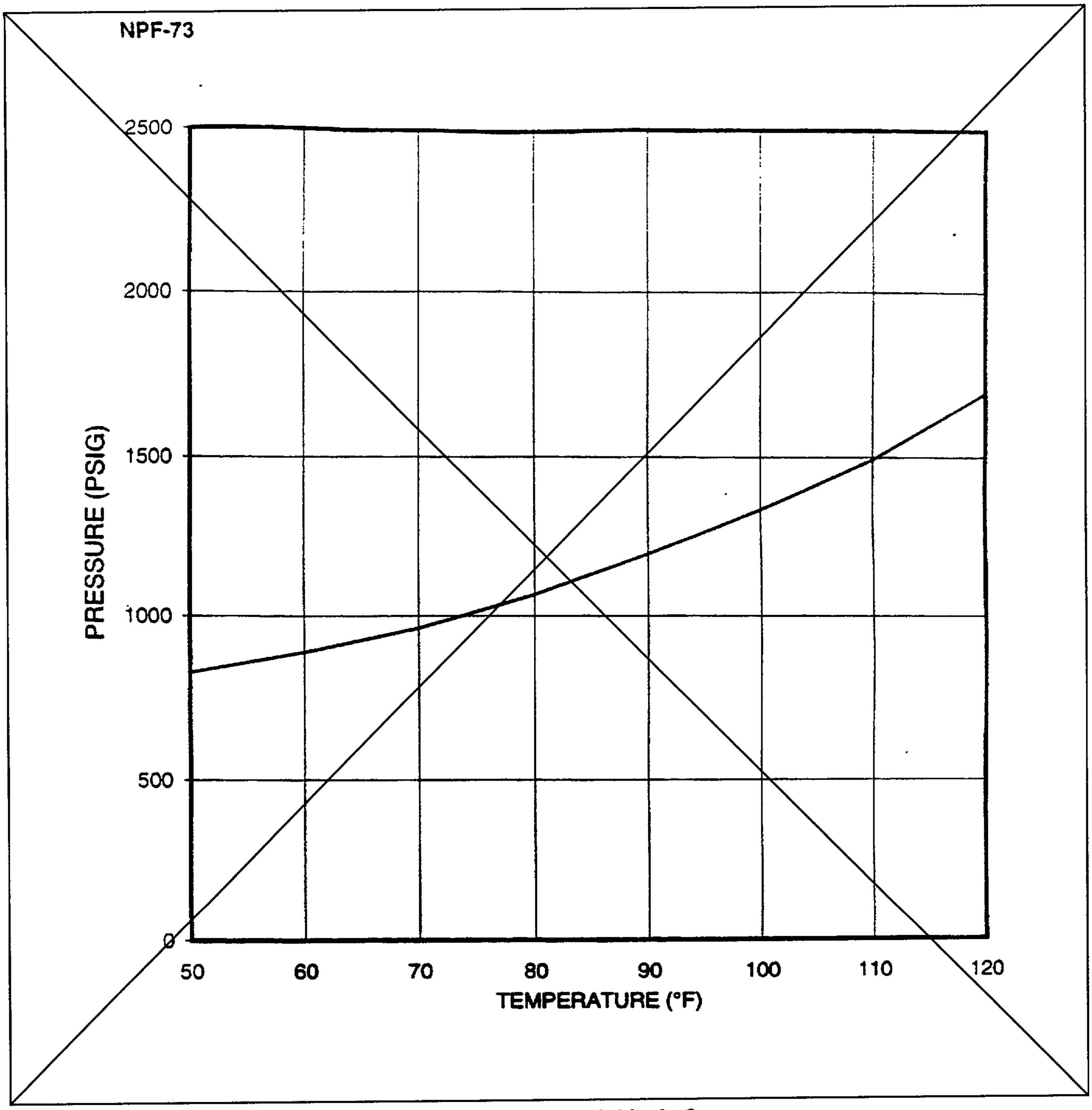
OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G requirements during the OPPS MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

Moved to PTLR.



~~FIGURE B 3/4 4-2~~

~~ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE~~

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

BACKGROUND (Continued)

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits specified in the PTLR by a significant amount could cause brittle cracking of the reactor vessel.

LCO 3.4.9.3, "Overpressure Protection Systems," provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires deactivating all but one charging pump and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the OPPS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve and, if needed, until the charging pump is actuated by SI.

The OPPS for pressure relief consists of two PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV REQUIREMENTS

As designed for the OPPS System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the OPPS actuation logic. The OPPS actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable, with respect to the PTLR ~~in the~~ limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal. The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open. Having the setpoints of both valves within the limits ensures that the Appendix G limits will not be exceeded in any analyzed event. When a

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

PORV REQUIREMENTS (Continued)

PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

The low limit on pressure during the transient is typically established based solely on an operational consideration for the Reactor Coolant Pump (RCP) No. 1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. The upper limit (based on the minimum of the steady-state 10 CFR 50 Appendix G requirement and the PORV piping limitations) and the RCP No. 1 seal performance criteria create a pressure range from which the setpoints for both PORVs are selected. When there is insufficient range between the upper and lower pressure limits to select the PORV setpoints to provide protection against violating both limits, setpoint selection to provide protection against the upper limit violation takes precedence.

RCS VENT REQUIREMENTS

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting OPPS mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it may be satisfied by removing a pressurizer safety valve or establishing an opening between the RCS and the containment atmosphere of the required size through any positive means available which cannot be inadvertently defeated. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits when low RCS temperature conditions exist. At the enable temperature specified in the PTLR and below, overpressure prevention is provided by two OPERABLE RCS relief valves or a depressurized RCS and a sufficient sized RCS vent.

This page contains changes proposed by LAR 157.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

The actual temperature at which the pressure in the PTLR P/T limit curve falls below the OPPS setpoint, as specified in the PTLR, increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR P/T limit curves are revised, the OPPS must be re-evaluated to ensure its functional requirements can still be met.

The PTLR contains the acceptance limits that define the OPPS requirements. ~~The heatup and cooldown curves represent the Appendix G limits that define OPPS operation.~~ Any change to the RCS that may affect OPPS operation must be evaluated against the analyses to determine the impact of the change on the OPPS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

MASS INPUT TYPE TRANSIENTS

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

HEAT INPUT TYPE TRANSIENTS

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the OPPS MODES to ensure that mass and heat input transients do not occur, which either of the OPPS overpressure protection means cannot handle:

a. Deactivating all but one charging pump OPERABLE;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if the secondary side water temperature of each steam generator in a non-isolated loop is greater than or equal to 50°F above the non-isolated RCS cold leg temperature in any non-isolated loop. LCO 3.4.1.2, "Reactor Coolant System - Hot Standby," and LCO 3.4.1.3, "Reactor Coolant System - Shutdown," provide this protection.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEAT INPUT TYPE TRANSIENTS (Continued)

The analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain the RCS pressure below the limits when only one charging pump is actuated by SI. Thus, the LCO allows only one charging pump OPERABLE during the OPPS MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

The isolated accumulators must have their discharge valves closed with power removed. Fracture mechanics analyses established the temperature of OPPS Applicability at the enable temperature specified in the PTLR.

PORV PERFORMANCE

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit specified in the PTLR. The setpoint is derived by analyses that model the performance of the OPPS assuming the limiting transient of SI actuation of one charging pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensure the P/T limits will be met.

The Maximum Allowed Nominal PORV Setpoint for the OPPS, specified in the PTLR, is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum allowable nominal setpoint ensures that Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the reactor vessel and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; (4) single failure; and (5) the pressure difference between the wide range pressure transmitter and the reactor vessel limiting beltline region.

The PTLR PORV setpoint will be updated when the revised PTLR P/T limits conflict with the OPPS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.9.1, "Pressure/Temperature Limits," discuss these examinations.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

PORV PERFORMANCE (Continued)

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

RCS VENT PERFORMANCE

With the RCS depressurized, analyses show that a PORV or equivalent opening with a vent size of 3.14 square inches is capable of mitigating the allowed OPPS overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the OPPS configuration, SI actuation with one charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size is based on the PORV size, therefore, the vent is bounded by the PORV analyses.

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

LCO

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

The Maximum Allowable Nominal Setpoint Curve of the PTLR defines the maximum nominal setpoint at which the PORVs can be set which will ensure that Appendix G limits are not exceeded. To maximize operating margin, the setpoint for the higher PORV is set at the Maximum Allowable Nominal Operating Curve within the respective instrumentation loop calibration tolerance band. The PORV setpoint uncertainty is calculated with reference to the methodology in ISA 67.04-1994 for performing instrumentation uncertainty calculations. The instrumentation calibration tolerances are provided in plant procedures. The overall setpoint calculation accounts for the instrumentation calibration tolerances in the uncertainty calculation.

Since actuation of both PORVs can result in excessive undershoot below the PORV setpoint, the lower PORV setpoints are staggered by an amount greater than or equal to the limiting overshoot (from either the mass injection or heat addition events). The staggered setpoints are provided in plant procedures.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)
LCO (Continued)

To limit the coolant input capability, the LCO requires one charging pump capable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized. The LCO is qualified by a note that permits two pumps capable of RCS injection for less than or equal to 15 minutes to allow for pump swaps. This note also allows all charging pumps capable of injecting into the RCS during a change from MODE 3 to MODE 4 to be OPERABLE for a limited period of time.

The LCO is also qualified by a note stating that accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or at the maximum RCS pressure for the existing temperature, as allowed by the PTLR P/T limit curves. This note permits the accumulator discharge isolation valve surveillance to be performed only under these pressure and temperature conditions.

Operation above the enable temperature specified in the PTLR, ~~350°F~~ but less than the enable temperature specified in the PTLR plus 25°F~~375°F~~ with only one centrifugal charging pump OPERABLE is allowed for up to 4 hours. As shown by analysis, LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single LHSI pump with no credit for accumulator injection. Given the short time duration that the condition of having only one centrifugal charging pump OPERABLE is allowed and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below the enable temperature specified in the PTLR, ~~350°F~~ but greater than the enable temperature specified in the PTLR minus 25°F~~325°F~~ with all centrifugal charging pumps OPERABLE is allowed for up to 4 hours immediately following a change from MODE 3 to MODE 4. This provides a reasonable period of time for the operators to secure an OPERABLE pump following entry into MODE 4. Since the charging pump is required to be OPERABLE in MODE 3, but is not required in MODE 4 due to OPPS limitations, some time constraints for making the transition must be identified. During low pressure, low temperature operation, all automatic Safety Injection actuation signals are blocked. In normal conditions, a single failure of the ESF actuation circuitry will result in the starting of at most one train of Safety Injection (one centrifugal charging pump, and one LHSI pump). For temperatures above the enable temperature specified in the PTLR minus 25°F~~325°F~~, an overpressure event occurring as a result of starting these two pumps can be successfully mitigated by operation of both PORVs without exceeding Appendix G limits. Given the short time duration that this condition is allowed and the low probability of a single failure causing an overpressure event during this time, the single failure of a PORV is not assumed. Initiation of both trains

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

LCO (Continued)

of Safety Injection during this 4-hour time frame due to operator error or a single failure occurring during testing of a redundant channel are not considered to be credible accidents.

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

a. Two OPERABLE PORVS; a PORV is OPERABLE for OPPS when its block valve is open, its lift setpoint is set to the limit and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits; or

b. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with an area of 3.14 square inches.

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

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. When the reactor vessel head is off, overpressurization cannot occur.

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

ACTION

a. With two or more charging pumps capable of injecting into the RCS, RCS overpressurization is possible.

   To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

b. An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the PTLR P/T limit curves.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

ACTION (Continued)

If isolation is needed and cannot be accomplished in 1 hour, the ACTION provides two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to more than the enable temperature specified in the PTLR, the accumulator pressure cannot exceed the OPPS limits if the accumulators are fully injected. Depressurizing the accumulators below the OPPS limit specified in the PTLR also gives this protection.

The completion times are based on operating experience that these activities can be accomplished in these time periods indicating that an event requiring OPPS is not likely in the allowed times.

c. In MODE 4 when any RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR, with one required RCS relief valve inoperable, the RCS relief valve must be restored to OPERABLE status within a completion time of 7 days. Two RCS relief valves are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component. The exception to Specification 3.0.4 will permit plant heatup with one inoperable PORV. Continued operation is permitted with one PORV inoperable.

The completion time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low. If plant operation results in transitioning to MODE 5, the completion time to restore an inoperable PORV may not exceed 7 days as required by this ACTION.

d. The consequences of operational events that will overpressurize the RCS are more severe at lower temperature. Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the completion time to restore two valves to OPERABLE status is 24 hours.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

SR 4.4.9.3.2

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. This surveillance is performed if the PORV satisfies the LCO.

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

The SR is required to be performed prior to entering the condition for the OPPS to be OPERABLE. This assures low temperature overpressure protection is available when the RCS cold leg temperature is less than or equal to the enable temperature specified in the PTLR. Performing the surveillance every 31 days on each required PORV permits verification and adjustment, if necessary, of its lift setpoint, and considers instrumentation reliability which has been shown through operating experience to be acceptable. The CHANNEL FUNCTIONAL TEST will verify the setpoint is within the allowed maximum limits. PORV actuation could depressurize the RCS and is not required.

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

3/4.4.11 REACTOR COOLANT SYSTEM RELIEF VALVES (Continued)

APPLICABILITY (Continued)

PORV opening setpoint. The most rapid increases will occur at the higher operating power and pressure conditions of MODES 1 and 2. The PORVs are also required to be OPERABLE in MODES 1, 2, and 3 to minimize challenges to the pressurizer safety valves.

Pressure increases are less prominent in MODE 3 because the core input energy is reduced, but the RCS pressure is high. Therefore, the LCO is applicable in MODES 1, 2, and 3. The LCO is not applicable in MODE 4 when both pressure and core energy are decreased and the pressure surges become much less significant. The PORV setpoint is reduced for OPPS in MODES 4 (below the enable temperature specified in the PTLR), 5, and 6 with the reactor vessel head in place. LCO 3.4.9.3 addresses the PORV requirements in these MODES.

ACTION

A General Note provides clarification that all pressurizer PORVs and block valves are treated as separate entities, each with separate completion times (i.e., the completion time is on a component basis).

a. With the PORVs inoperable and capable of being manually cycled, either the PORVs must be restored or the flow path isolated within 1 hour. The block valves should be closed but power must be maintained to the associated block valves, since removal of power would render the block valve inoperable. Although a PORV may be designated inoperable, the associated vent path may be manually opened and closed, and the PORV therefore, able to perform its function. PORV inoperability may be due to seat leakage, instrumentation problems related to PORV accident monitoring instruments identified in LCO 3.3.3.8, or other causes that do not prevent manual use and do not create a possibility for a small break LOCA. If the position indication is inoperable, then the PORVs are inoperable. For these reasons, the block valve shall be closed but the ACTION requires power be maintained to the valve. Automatic control problems and related instrumentation problems would not render the PORVs inoperable. Accident analyses assume manual operation of the PORVs and do not take credit for automatic actuation. This condition is only intended to

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point on the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the ECCS Flow Analysis. The term "required developed head" refers to the pump performance at a given flow point that is assumed in the ECCS Flow Analysis. This is possible since the analysis assumes the pump delivers different flows at different times during accident mitigation. These multiple points are represented by a curve. The values at various flow points are defined by the Minimum Operating Point (MOP) curve in the Inservice Testing (IST) Program. The verification that the pump's developed head at the flow test point is greater than or equal to the required developed head is performed by using the MOP curve. Surveillance requirements are specified in the IST Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and frequencies necessary to satisfy the requirements.

The 18-month surveillance interval is consistent with expected length of fuel cycles and allows for component testing to be performed during plant shutdown conditions if necessary to avoid a plant transient that could occur if the component were tested at power. However, for those components that may be safely tested at power, the 18-month surveillance may be met by performing the required testing at power.

The limitation for a maximum of one charging pump to be OPERABLE and the surveillance requirement to verify all charging pumps except the required OPERABLE pump to be inoperable below the enable temperature specified in the PTLR~~350°F~~, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

ATTACHMENT E

Beaver Valley Power Station
PTLR Requirement Location

As per Generic Letter 96-03 relocation of the Pressure/Temperature (P/T) limit curves and Overpressure Protection System (OPPS) setpoints to a licensee-controlled document requires three separate licensee actions. The licensee must

(1) have a methodology approved by the NRC to reference in its Technical Specifications (TS);

(2) develop a report such as a Pressure and Temperature Limits Report (PTLR) or a similar document to contain the figures, values, parameters, and any explanation necessary; and

(3) modify the applicable sections of the Technical Specifications accordingly.

The first two of the three requirements for relocating the P/T curves and OPPS setpoints are an NRC-approved methodology and the associated reporting requirements in the PTLR. The methodology will consist of only those methods used for calculation, not the calculations themselves. The PTLR will consist of the explanations, figures, values, and parameters derived from the calculations. Since the PTLR will be provided to the NRC upon issuance after each fluence period or Effective Full Power Years (EFPYs) and after approval of the methodology, a PTLR should be provided when the methodology is submitted so that questions regarding the content and format of the PTLR can be addressed prior to its formal completion.

The following table shows the PTLR location of the requirements recommended by Generic Letter 96-03 to be included in PTLR.

The table shows the relationship between the provisions, if applicable, specified in the Technical Specifications for the approved methodology and the requirements to be included in the methodology and the PTLR. The provisions for the methodology are those shown in Section 6.9.6 of the Technical Specifications.

REQUIREMENTS FOR METHODOLOGY AND PTLR

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN TS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	UNIT 1 PTLR LOCATION	UNIT 2 PTLR LOCATION
1. The methodology shall describe how the neutron fluence is calculated (reference new regulatory guide when it is issued).	Describe transport calculation methods including computer codes and formulas used to calculate neutron fluence. Provide references.	Provide the values of neutron fluences that are used in the adjusted reference temperature (ART) calculation.	Results are shown in Table 4.2-7. Methodology is described in WCAP-15571, Rev. 0 for Capsule Y. See Section 4.2.4 of the PTLR.	Results are shown in Table 4.2-8. Methodology is described in WCAP-14484, Rev. o, for Capsule Y. See Section 4.2.4 of the PTLR.
2. The Reactor Vessel Material Surveillance Program shall comply with Appendix H to 10 CFR Part 50. The reactor vessel material irradiation surveillance specimen removal schedule shall be provided, along with how the specimen examinations shall be used to update the PTLR curves.	Briefly describe the surveillance program. Licensee transmittal letter should identify by title and number report containing the Reactor Vessel Surveillance Program and surveillance capsule reports. Topical/generic report contains placeholder only. Reference Appendix H to 10 CFR Part 50.	Provide the surveillance capsule withdrawal schedule, or reference by title and number the documents in which the schedule is located.	BVPS Unit 1 UFSAR Table 4.5-3. See Section 4.2.2 of the PTLR.	BVPS Unit 2 UFSAR Table 5.3-6. See Section 4.2.2 of the PTLR.
		Reference the surveillance capsule reports by title and number if ARTs are calculated using surveillance data.	WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," C. Brown, et. al., November 2000. See Section 4.2.4 of the PTLR.	WCAP-14484, Revision 0, "Analysis of Capsule V from the Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Grendys, S. L. Anderson, J. F. Williams, February 1996. See Section 4.2.4 of the PTLR.

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN TS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	UNIT 1 PTLR LOCATION	UNIT 2 PTLR LOCATION
3. Low temperature overpressure protection (LTOP) system limits developed using NRC-approved methodologies may be included in the PTLR.	Describe how the LTOP system limits are calculated applying system/thermal hydraulics and fracture mechanics. Reference SRP Section 5.2.2; ASME Code Case N-514; ASME Code, Appendix G, Section XI as applied in accordance with 10 CFR 50.55.	Provide setpoint curves or setpoint values.	Table 4.2-3. Methodology described in Westinghouse Report, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule", Revision 1, April 2001. See Section 4.2.4 of the PTLR.	Figure 4.2-8. Table 4.2-3. Methodology described in Westinghouse Report NPD-OPES(99)-055. See Section 4.2.4 of the PTLR.
4. The adjusted reference temperature (ART) for each reactor beltline material shall be calculated, accounting for irradiation embrittlement, in accordance with Regulatory Guide 1.99, Revision 2.	Describe the method for calculating the ART using Regulatory Guide 1.99, Revision 2.	Identify both the limiting ART values and limiting materials at the 1/4t and 3/4t locations (t = vessel beltline thickness).	Table 4.2-6. Methodology described in WCAP-15569. See Section 4.2.4 of the PTLR.	Table 4.2-7. Methodology described in WCAP-15139. See Section 4.2.4 of the PTLR.
		PWRs - identify RT_{PTS} value in accordance with 10 CFR 50.61.	Tables 4.2-9 & 4.2-10. Methodology described in WCAP-15569. See Section 4.2.4 of the PTLR	Table 4.2-10. Methodology described in WCAP-15139. See Section 4.2.4 of the PTLR
5. The limiting ART shall be incorporated into the calculation of the pressure and temperature limit curves in accordance with NUREG-0800, SRP Section 5.3.2, Pressure-Temperature Limits.	Describe the application of fracture mechanics in constructing P/T curves based on ASME Code, Appendix G, Section XI, and SRP Section 5.3.2.	Provide the P/T curves for heatup, cooldown, criticality, and hydrostatic and leak tests.	Figures 4.2-1 & 4.2-2. Methodology described in WCAP-15569. See Section 4.2.4 of the PTLR	Figures 4.2-1 through 4.2-6. Methodology described in WCAP-15139. See Section 4.2.4 of the PTLR

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN TS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	UNIT 1 PTLR LOCATION	UNIT 2 PTLR LOCATION
6. The minimum temperature requirements of Appendix G to 10 CFR Part 50 shall be incorporated into the pressure and temperature limit curves.	Describe how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T curves.	Identify minimum temperatures on the P/T curves such as minimum boltup temperature and hydrotest temperature.	Figure 4.2-1. Methodology described in WCAP-15569. See Section 4.2.4 of the PTLR	Figure 4.2-1. Methodology described in WCAP-15139. See Section 4.2.4 of the PTLR
7. Licensees who have removed two or more capsules should compare for each surveillance material the measured increase in reference temperature (RT_{NDT}) to the predicted increase in RT_{NDT}; where the predicted increase in RT_{NDT} is based on the mean shift in RT_{NDT} plus the two standard deviation value ($2\sigma_{\Delta}$) specified in Regulatory Guide 1.99, Revision 2. If the measured value exceeds the predicted value (increase in $RT_{NDT} + 2\sigma_{\Delta}$), the licensee should provide a supplement to the PTLR to demonstrate how the results affect the approved methodology.	Describe how the data from multiple surveillance capsules are used in the ART calculation. Describe procedure if measured value exceeds predicted value. <u>WHEN OTHER PLANT DATA ARE USED</u> 1. Identify the source(s) of data when other plant data are used. 2.a Identify by title and number the safety evaluation report that approved the use of data for the plant. Justify applicability. OR	Provide supplemental data and calculations of the chemistry factor in the PTLR if the surveillance data are used in the ART calculation.	Tables 4.2-4, 4.2-6 & 4.2-7. See WCAP-15570 & 15569. See Section 4.2.4 of the PTLR	Tables 4.2-5, 4.2-7 & 4.2-8. See WCAP-15139. See Section 4.2.4 of the PTLR

PROVISIONS FOR METHODOLOGY FROM ADMINISTRATIVE CONTROLS SECTION IN TS	MINIMUM REQUIREMENTS TO BE INCLUDED IN METHODOLOGY	MINIMUM REQUIREMENTS TO BE INCLUDED IN PTLR	UNIT 1 PTLR LOCATION	UNIT 2 PTLR LOCATION
	2.b Compare licensee data with other plant data for both the radiation environments (e.g., neutron spectrum, irradiation temperature) and the surveillance test results.	Evaluate the surveillance data to determine if they meet the credibility criteria in Regulatory Guide 1.99, Revision 2. Provide the results.	Tables 4.2-4 & 4.2-6. See WCAP-15570 & 15569. See Section 4.2.4 of the PTLR	Tables 4.2-5 & 4.2-7. See WCAP-15139. See Section 4.2.4 of the PTLR