February 28, 1997

Ms. Irene Johnson, Acting Manager
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M96898, M96899, M96900 AND M96901)

Dear Ms. Johnson:

The Commission has issued the enclosed Amendment No. 153 to Facility Operating License No. DPR-19 and Amendment No. 148 to Facility Operating License No. DPR-25 for Dresden, Units 2 and 3, and Amendment No. 172 to Facility Operating License No. DPR-29 and Amendment No. 168 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated September 20, 1996. By letter dated December 9, 1996, the staff requested additional information concerning the application. By letter dated January 21, 1997, the licensee provided the additional information.

The amendments would update the pressure-temperature (P-T) curves contained in the Dresden and Quad Cities Technical Specifications to 22 Effective Full Power Years (EFPYs).

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249, 50-254, 50-265

Enclosures:
1. Amendment No. 153 to DPR-19
2. Amendment No. 148 to DPR-25
3. Amendment No. 172 to DPR-29
4. Amendment No. 168 to DPR-30
5. Safety Evaluation

cc w/encl: See next page

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*concurrence provided by memo dated 2/5/97; no major changes

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NAME	JSTANG		CMOORE		RPULSIFER		Marco		*BELLIOT		RCAPRA	
DATE	02/21/97		02/21/97		02/21/97		02/26/97		02/05/97		03/01/97	

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I. Johnson
Commonwealth Edison Company

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Unit Nos. 2 and 3
Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 20, 1996, as supplemented January 21, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 153 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 28, 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 148
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 20, 1996, as supplemented January 21, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 148, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 28, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 153 AND 148

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
VIII	VIII
3/4.6-19	3/4.6-19
3/4.6-20	3/4.6-20
3/4.6-21	3/4.6-21
--	3/4.6-21a
--	3/4.6-21b
--	3/4.6-21c
--	3/4.6-21d
B3/4.6-6	B3/4.6-6
B3/4.6-7	B3/4.6-7
B3/4.6-8	B3/4.6-8
B3/4.6-9	--

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The primary system coolant system temperature and reactor vessel metal temperature and pressure shall be limited as specified below:

1. Pressure Testing:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figures 3.6.K-1 through 3.6.K-3 with the rate of change of the primary system coolant temperature ≤20°F per hour, or

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour when reactor vessel metal temperature and pressure is maintained within the Acceptable Regions as shown on Figure 3.6.K-4.

2. Non-Nuclear Heatup and Cooldown and low power PHYSICS TESTS:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-4, and

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During non-nuclear heatup or cooldown, and pressure testing operations, at least once per 30 minutes,

   a. The rate of change of the primary system coolant temperature shall be determined to be within the heatup and cooldown rate limits, and

   b. The reactor vessel metal temperature and pressure shall be determined to be within the Acceptable Regions on Figures 3.6.K-1 through 3.6.K-4.

2. For reactor critical operation, determine within 15 minutes prior to the withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown,

   a. The rate of change of the primary system coolant temperature to be within the limits, and

   b. The reactor vessel metal temperature and pressure to be within the Acceptable Region on Figure 3.6.K-5.

3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-5, and

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4. The reactor vessel flange and head flange temperature ≥83°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

ACTION:

With any of the above limits exceeded,

1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes without exceeding the applicable primary system coolant temperature rate of change limit, and

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations within 72 hours, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

4. The reactor vessel flange and head flange temperature shall be verified to be ≥83°F:

   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:

      1) ≤113°F, at least once per 12 hours.

      2) ≤93°F, at least once per 30 minutes.

   b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

FIGURE 3.6.K-1

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 18 EFPY

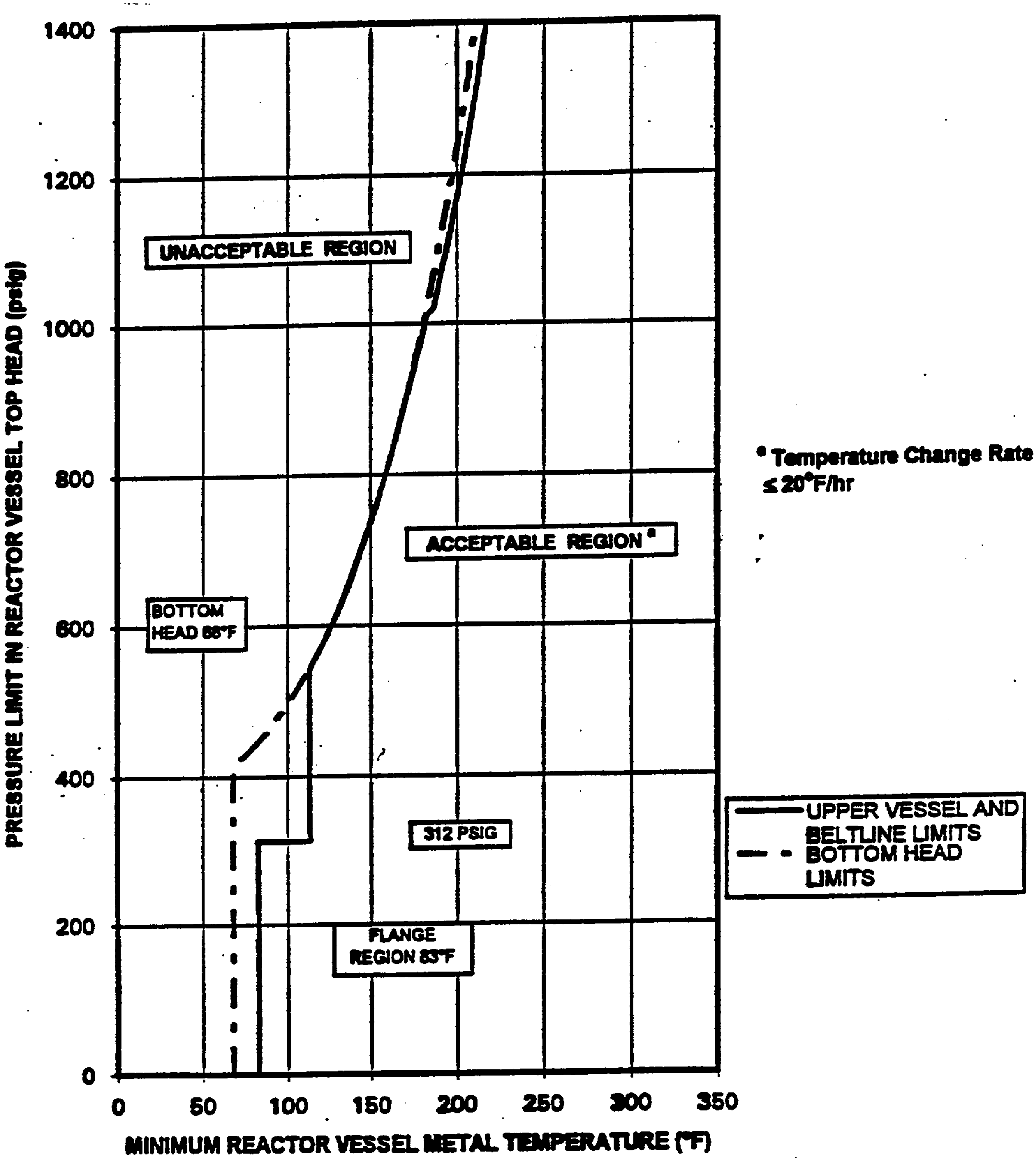


FIGURE 3.6.K-2

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 20 EFPY

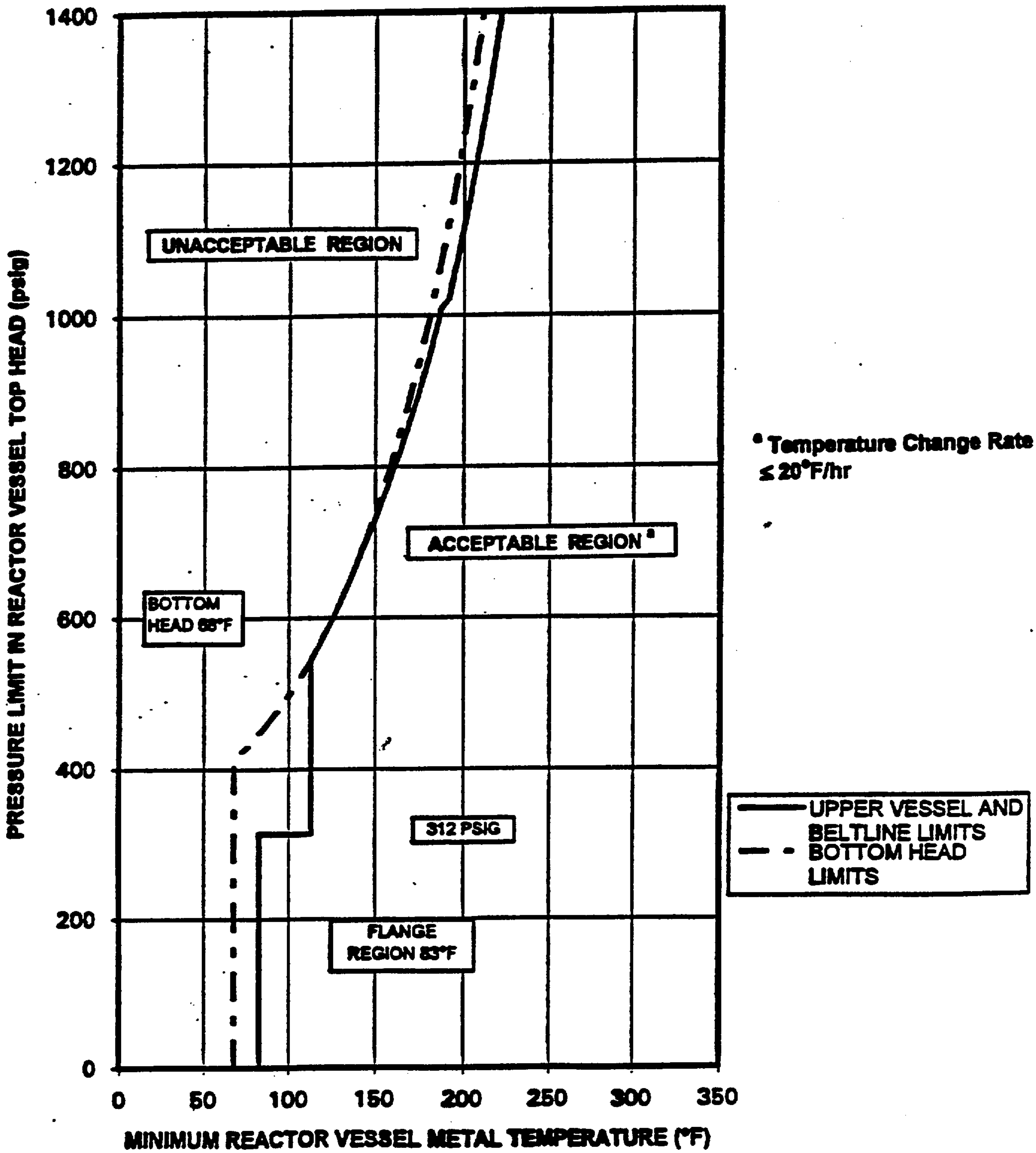


FIGURE 3.6.K-3

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 22 EFPY

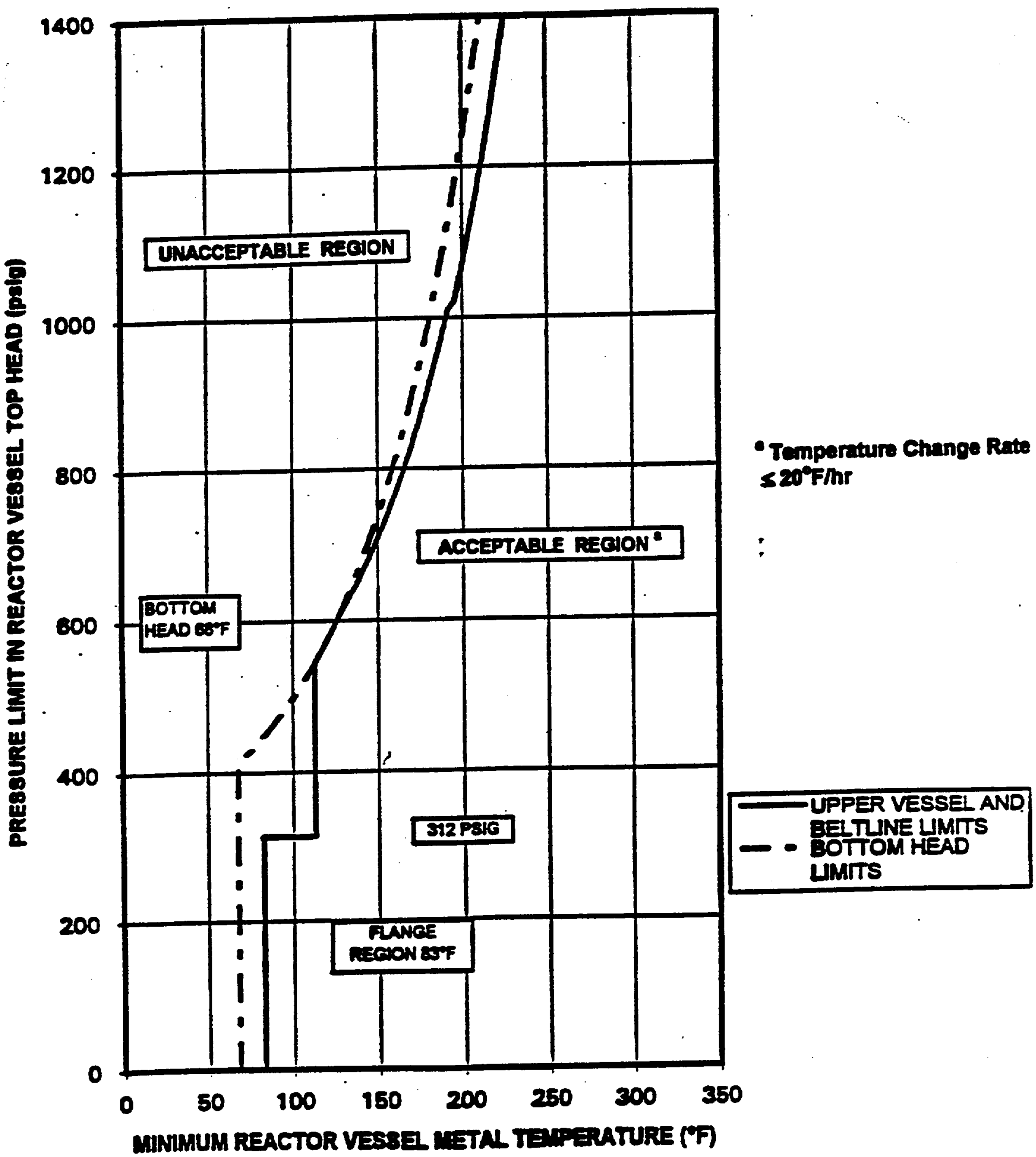


FIGURE 3.6.K-4

PRESSURE - TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN - VALID TO 22 EFPY

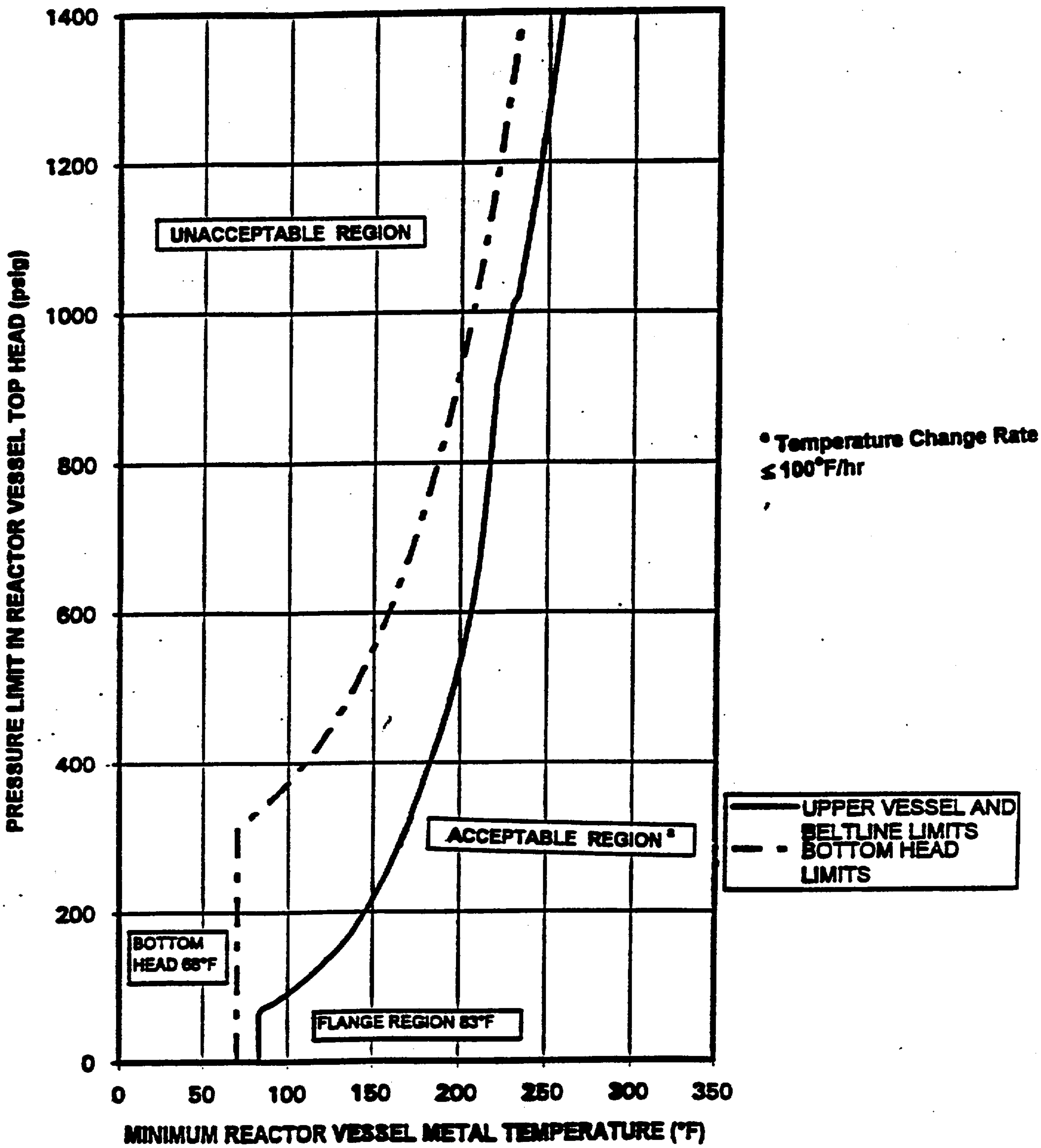
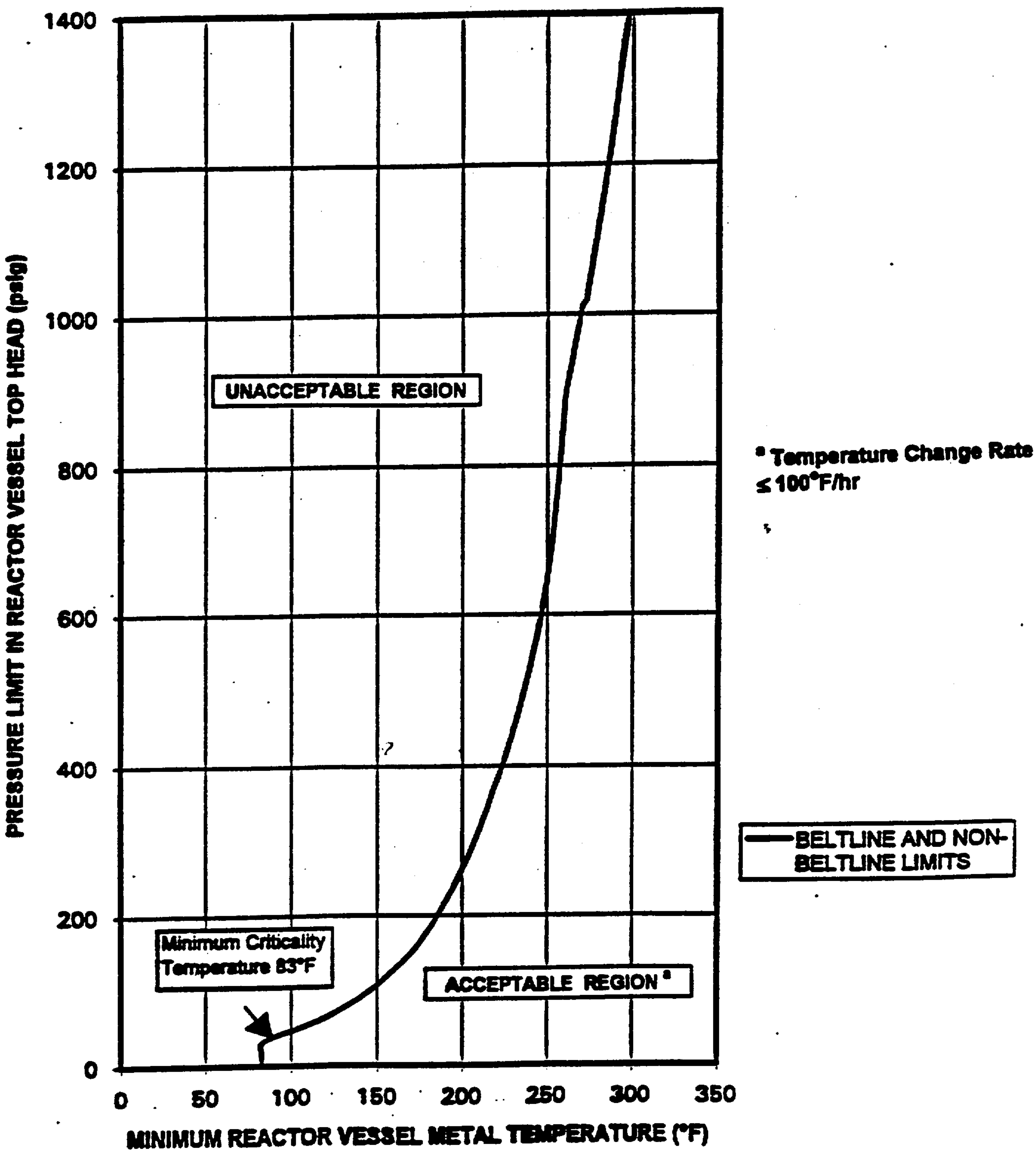


FIGURE 3.6.K-5

PRESSURE - TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 22 EFPY



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 pressure-temperature limit lines are shown, for operating conditions; Pressure Testing, Figures 3.6.K-1 through 3.6.K-3, Non-Nuclear Heatup/Cooldown, Figure 3.6.K-4, and Core Critical Operation Figure 3.6.K-5. The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Four vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region and the bottom head region); 3) the closure flange region and 4) the bottom head region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline, closure flange, and bottom head regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange and bottom head regions are non-beltline regions, they are treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, connecting welds and the vertical electroslag welds which terminate immediately below the vessel flange is 23°F. Therefore, the minimum allowable boltup temperature is established as 83°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Figures 3.6.K-1 through 3.6.K-3 - Pressure Testing

As indicated in Figure 3.6.K-1 through 3.6.K-3 for pressure testing, the minimum metal temperature of the reactor vessel shell is 83°F for reactor pressures less than 312 psig. This 83°F minimum boltup temperature is based on a RT_{NDT} of 23°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. The bottom head region limit is established as 68°F, based on moderator temperature assumptions for shutdown margin analyses. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 113°F. The 113°F minimum temperature is based on a closure flange region RT_{NDT} of 23°F and a 90°F conservatism required by 10CFR Part 50 Appendix G. Beltline curves as a function of vessel exposure for 18, 20 and 22 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 22 EFPY of operation.

BASES

Figures 3.6.K-1 through 3.6.K-3 are governing for applicable pressure testing with a maximum heatup/cooldown rate of 20°F/hour.

Figure 3.6.K-4 - Non-Nuclear Heatup/Cooldown

Figure 3.6.K-4 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. The maximum heatup/cooldown rate of 100°F/hour is applicable.

Figure 3.6.K-5 - Core Critical Operation

The core critical operation curve shown in Figure 3.6.K-5, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any pressure testing or non-nuclear heatup/cooldown limits. Since Figure 3.6.K-4 is more limiting, Figure 3.6.K-5 is Figure 3.6.K-4 plus 40°F. The maximum heatup/cooldown rate of 100°F/hour is applicable.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens are used in predicting reactor vessel material embrittlement. The operating limit curves of Figures 3.6.K-1 through 3.6.K-5 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two

BASES

valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 172
License No. DPR-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 20, 1996, as supplemented January 21, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 172 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 28, 1997

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 20, 1996, as supplemented January 21, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

For

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 28, 1997

ATTACHMENT TO LICENSE AMENDMENT NOS. 172 AND 168

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
VIII	VIII
3/4.6-19	3/4.6-19
3/4.6-20	3/4.6-20
3/4.6-21	3/4.6-21
--	3/4.6-21a
--	3/4.6-21b
--	3/4.6-21c
--	3/4.6-21d
B3/4.6-5	B3/4.6-5
B3/4.6-6	B3/4.6-6
B3/4.6-7	B3/4.6-7
B3/4.6-8	B3/4.6-8
B3/4.6-9	--

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The primary system coolant system temperature and reactor vessel metal temperature and pressure shall be limited as specified below:

1. Pressure Testing:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figures 3.6.K-1 through 3.6.K-3 with the rate of change of the primary system coolant temperature ≤20°F per hour, or

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour when reactor vessel metal temperature and pressure is maintained within the Acceptable Regions as shown on Figure 3.6.K-4.

2. Non-Nuclear Heatup and Cooldown and low power PHYSICS TESTS:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-4, and

   b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During non-nuclear heatup or cooldown, and pressure testing operations, at least once per 30 minutes,

   a. The rate of change of the primary system coolant temperature shall be determined to be within the heatup and cooldown rate limits, and

   b. The reactor vessel metal temperature and pressure shall be determined to be within the Acceptable Regions on Figures 3.6.K-1 through 3.6.K-4.

2. For reactor critical operation, determine within 15 minutes prior to the withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown,

   a. The rate of change of the primary system coolant temperature to be within the limits, and

   b. The reactor vessel metal temperature and pressure to be within the Acceptable Region on Figure 3.6.K-5.

3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

    a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-5, and

    b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4. The reactor vessel flange and head flange temperature ≥83°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

ACTION:

With any of the above limits exceeded,

1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes without exceeding the applicable primary system coolant temperature rate of change limit, and

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations within 72 hours, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

4. The reactor vessel flange and head flange temperature shall be verified to be ≥83°F:

    a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:

        1) ≤113°F, at least once per 12 hours.

        2) ≤93°F, at least once per 30 minutes.

    b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

FIGURE 3.6.K-1

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 18 EFPY

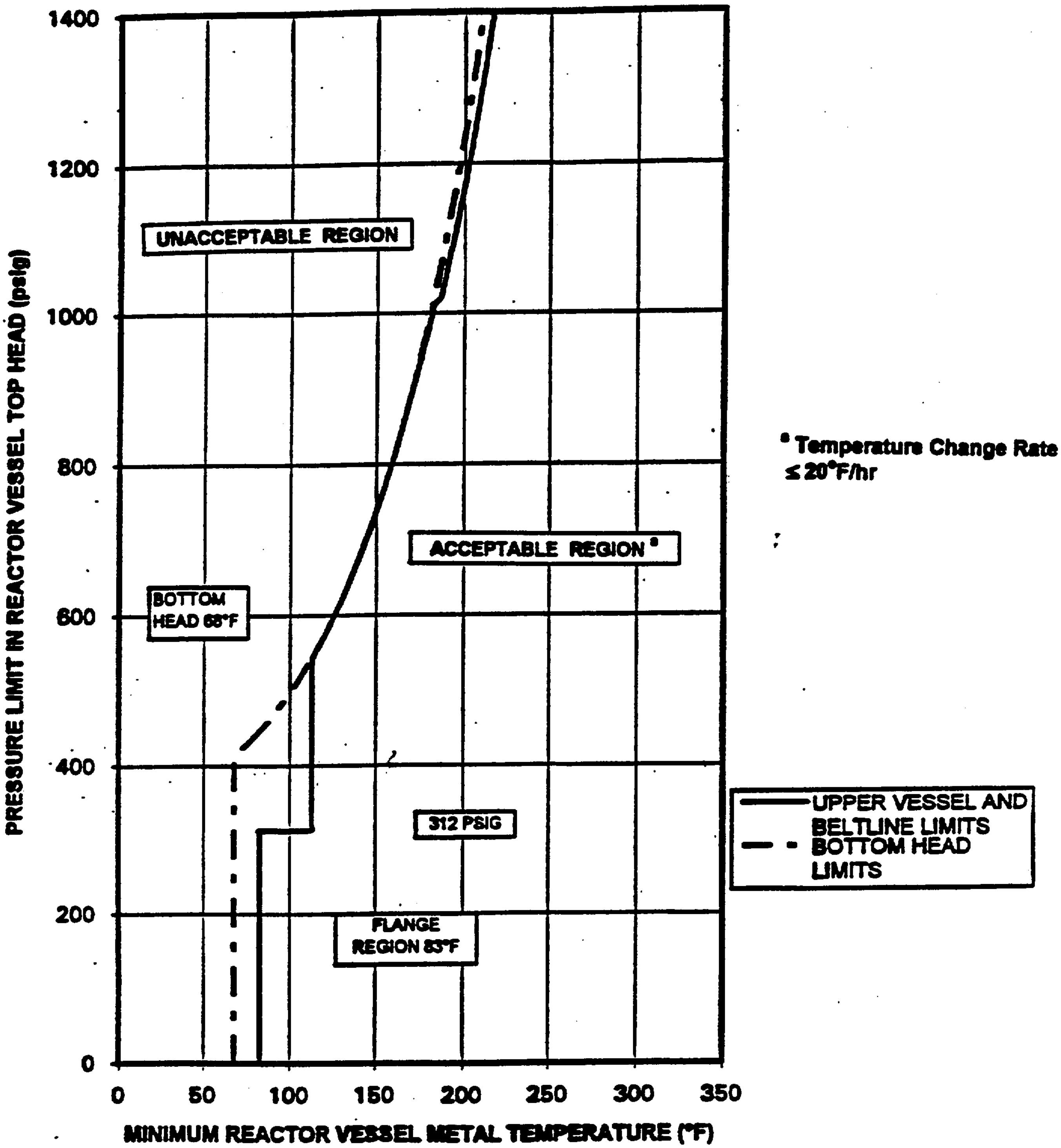


FIGURE 3.6.K-2

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 20 EFPY

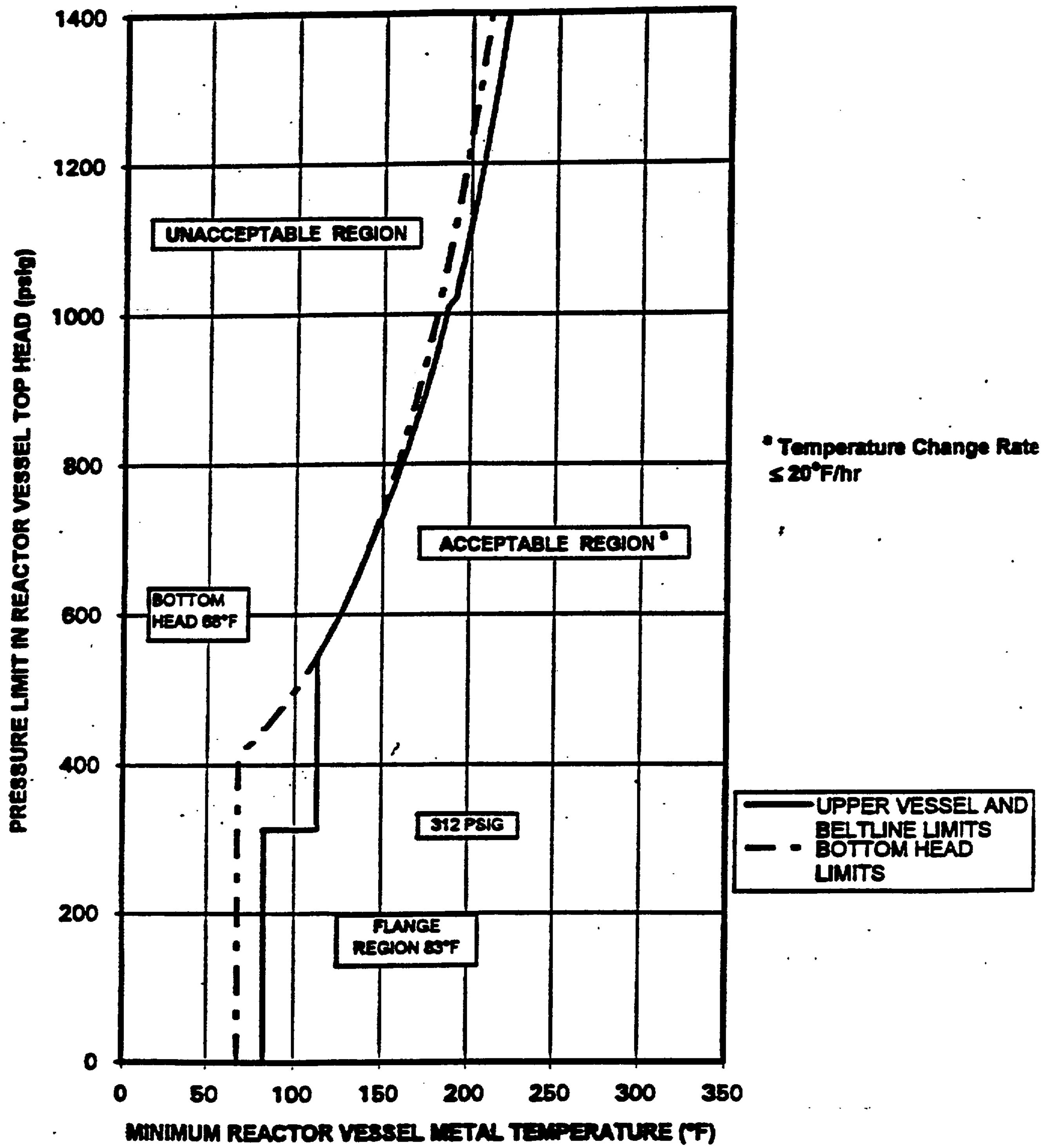


FIGURE 3.6.K-3

PRESSURE - TEMPERATURE LIMITS FOR PRESSURE TESTING - VALID TO 22 EFPY

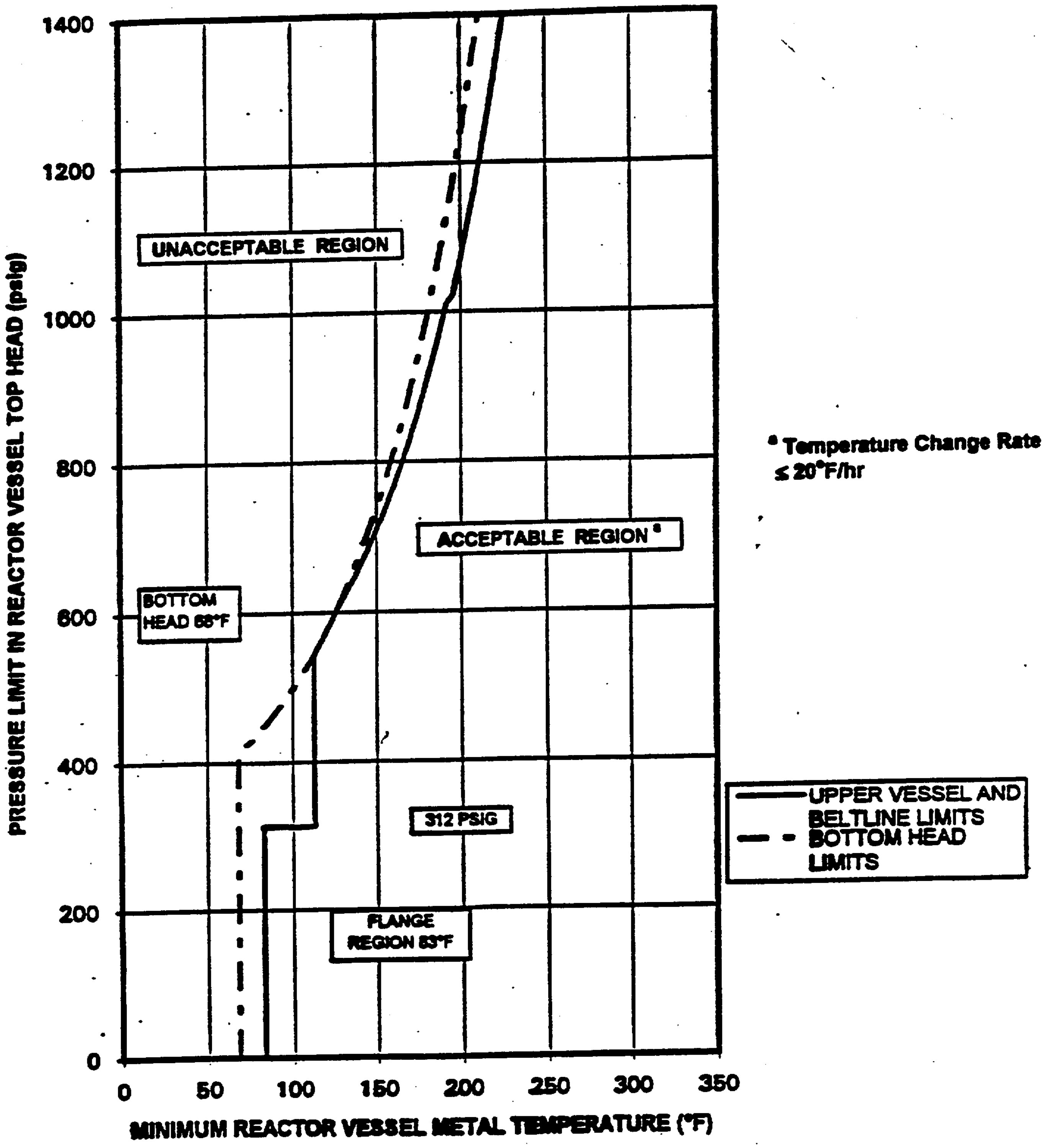


FIGURE 3.6.K-4

PRESSURE - TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN - VALID TO 22 EFPY

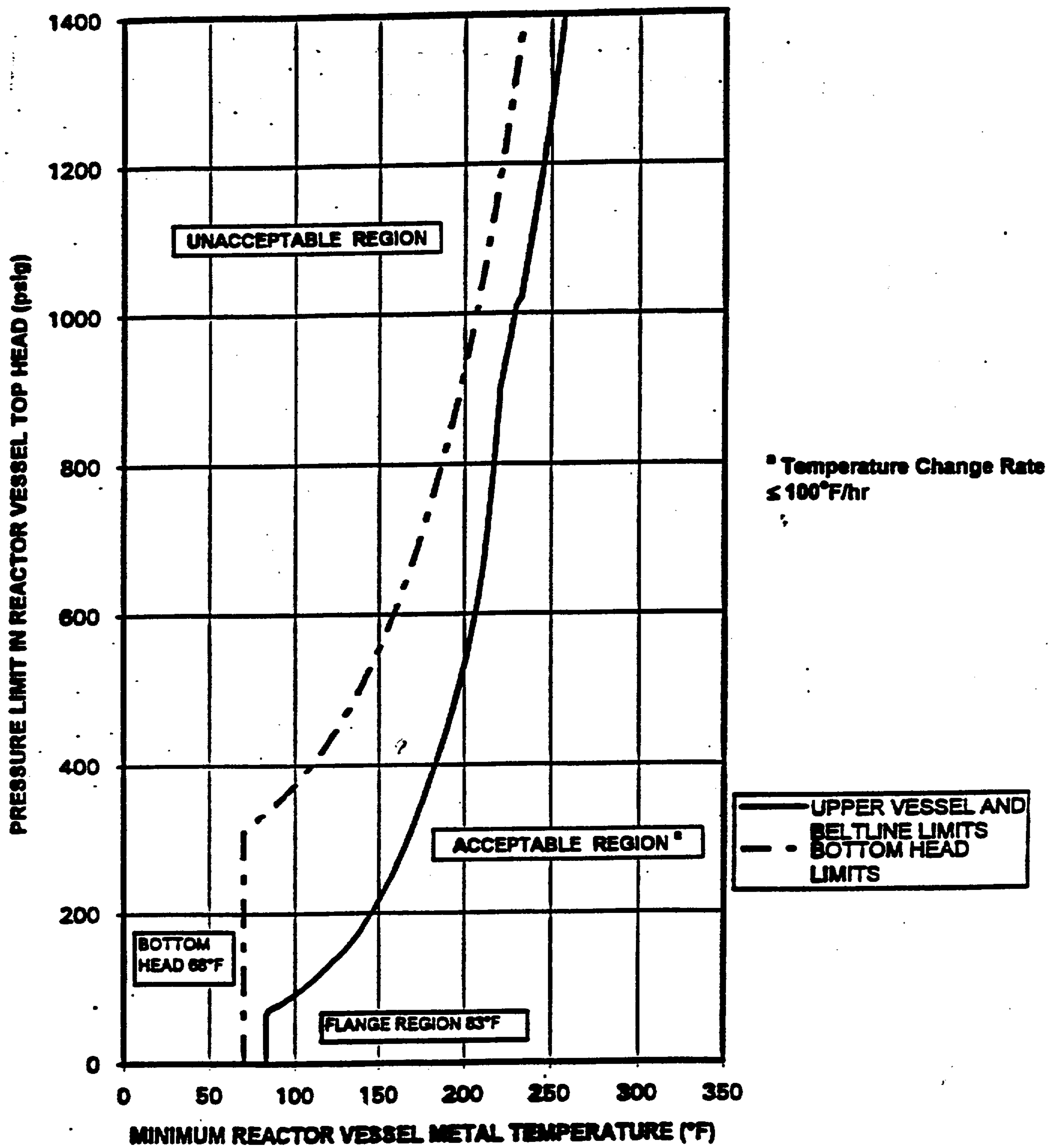
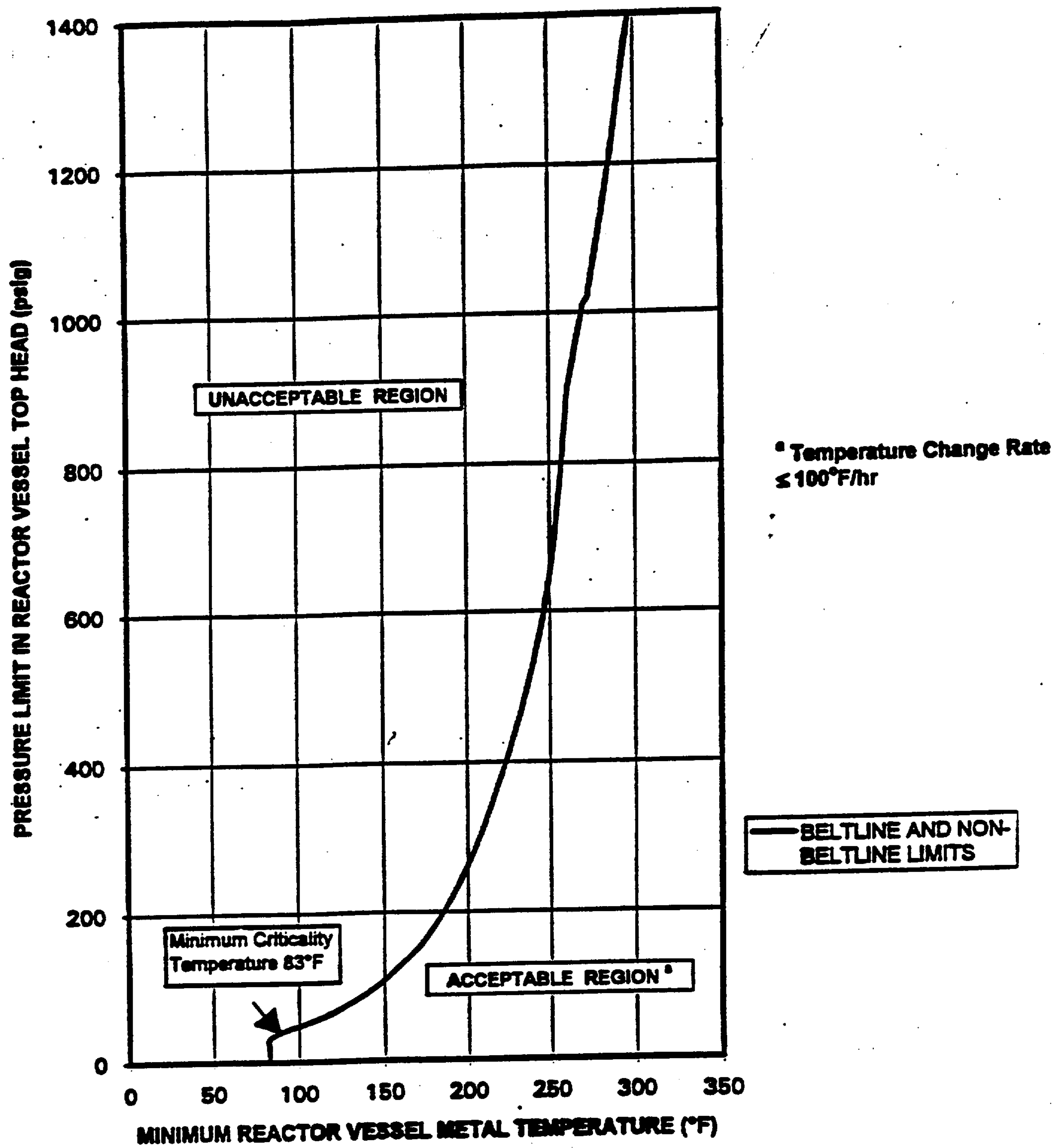


FIGURE 3.6.K-5

PRESSURE - TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS - VALID TO 22 EFPY



3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K 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 3.9.1.1.1 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 pressure-temperature limit lines are shown, for operating conditions; Pressure Testing, Figures 3.6.K-1 through 3.6.K-3 Non-Nuclear Heatup/Cooldown, Figure 3.6.K-4 and Core Critical Operation Figure 3.6.K-5. The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Four vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region and the bottom head region); 3) the closure flange region, and 4) the bottom head region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline, closure flange, and bottom head regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange and bottom head regions are non-beltline regions, they are treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

Boltup Temperature

The limiting initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, connecting welds and the vertical electroslag welds which terminate immediately below the vessel flange is 23°F. Therefore, the minimum allowable boltup temperature is established as 83°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Figures 3.6.K-1 through 3.6.K-3 Pressure Testing

As indicated in Figure 3.6.K-1 through 3.6.K-3 for pressure testing, the minimum metal temperature of the reactor vessel shell is 83°F for reactor pressures less than 312 psig. This 83°F minimum boltup temperature is based on a RT_{NDT} of 23°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. The bottom head region limit is established as 68°F, based on moderator temperature assumptions for shutdown margin analyses. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 113°F. The 113°F minimum temperature is based on a closure flange region RT_{NDT} of 23°F and a 90°F conservatism required by 10CFR Part 50 Appendix G. Beltline curves as a function of vessel exposure for 18, 20 and 22 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 22 EFPY of operation.

Figures 3.6.K-1 through 3.6.K-3 are governing for applicable pressure testing with a maximum heatup/cooldown rate of 20°F/hour.

Figure 3.6.K-4 - Non-Nuclear Heatup/Cooldown

Figure 3.6.K-4 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. The maximum heatup/cooldown rate of 100°F/hour is applicable.

BASES

Figure 3.6.K-5 - Core Critical Operation

The core critical operation curve shown in Figure 3.6.K-5, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Pressure testing or non-nuclear heatup/cooldown limits. Since Figure 3.6.K-4 is more limiting, Figure 3.6.K-5 is Figure 3.6.K-4 plus 40°F. The maximum heatup/cooldown rate of 100°F/hour is applicable.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-82 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens are used in predicting reactor vessel material embrittlement. The operating limit curves of Figures 3.6.K-1 through 3.6.K-5 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

BASES

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Residual Heat Removal - HOT SHUTDOWN

3/4.6.P Residual Heat Removal - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

A single shutdown cooling mode subsystem provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two subsystems be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associtated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Therefore, to meet the Limiting Condition for Operation, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems (the ability to take credit for a common heat exchanger and discharge piping only applies to the SDC mode of RHR).

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 148 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 172 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 168 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated September 20, 1996, Commonwealth Edison Company (ComEd, the licensee) submitted changes related to the pressure-temperature (P-T) limits in the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). These changes include reformatting of TS 3/4.6.K from a single, three part paragraph to three separate paragraphs for clarity as well as specification of a time limitation in the ACTION statement for TS 3.6. The licensee revised the P-T limits to provide new limits that are valid to 22 effective full power years (EFPY). The curves are based on the most limiting material among the group of four vessels. In addition, the licensee requested to amend the TSs by incorporation of separate P-T limits for the vessel bottom head. The NRC staff issued a Request for Additional Information (RAI) on December 9, 1996, subsequent to a teleconference that occurred on November 26, 1996. Additional clarifying information that did not change the original proposed no significant hazards consideration determination was supplied by ComEd in a letter dated January 21, 1997.

In the response to the RAI, ComEd committed to provide a revision to the supporting analysis, "General Electric Company (GE) report, GE-NE-B11-00707-01R1, 'Pressure Temperature Curves for Dresden and Quad Cities Stations,' dated July 1996," to reflect the addition of one standard deviation to the electroslag (ES) weld chemistry mean values. The licensee also committed to revise the adjusted reference temperature (ART) tables for 18, 20 and 22 EFPY as a result of the change in the chemistry mean values of the ES welds. Both revisions are expected to be provided by April 18, 1997, at which time the staff will verify all commitments.

The staff evaluates the P-T limits based on the following NRC regulations and guidance: Appendix G to 10 CFR Part 50; Generic Letters (GL) 88-11 and 92-01; Regulatory Guide (RG) 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that P-T limits for the reactor vessel must be at least as conservative as those obtained by Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). GL 88-11 recommends that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on the ART of reactor vessel materials. The ART is defined as the sum of initial nil-ductility transition reference temperature (RT_{NDT}) of the material, the increase in RT_{NDT} caused by neutron irradiation, and a margin to account for uncertainties in the prediction method. The increase in RT_{NDT} is calculated from the product of a chemistry factor and a fluence factor. The chemistry factor may be calculated using credible surveillance data, obtained by the licensee's surveillance program, as directed by Position 2 of RG 1.99, Revision 2. If credible surveillance data are not available, the chemistry factor is calculated dependent upon the amount of copper and nickel in the vessel material as specified in Table 1 of RG 1.99, Revision 2. GL 92-01 requires licensees to submit reactor vessel materials data, which the staff uses in the review of the P-T limits submittals.

SRP 5.3.2 provides guidance on calculation of the P-T limits using linear elastic fracture mechanics methodology specified in Appendix G to Section III of the ASME Code. The linear elastic fracture mechanics methodology postulates sharp surface defects that are normal to the direction of maximum stress and have a depth of one-fourth of the reactor vessel beltline thickness (1/4T) and a length of 1-1/2 times the beltline thickness. The critical locations in the vessel for this methodology are the 1/4T and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

2.0 EVALUATION

For the Dresden, Units 2 and 3, and the Quad Cities, Units 1 and 2, reactor vessels, the licensee determined that the most limiting material among the group of four vessels at the 1/4T and 3/4T locations is the lower intermediate to lower shell circumferential welds in the Dresden, Unit 3, vessel. This weld was fabricated using weld wire heat 299L44. The licensee calculated an ART of 97.4 degrees Fahrenheit at the 1/4T location and an ART of 81.2 degrees Fahrenheit at the 3/4T location at 22 EFPY. The neutron fluence used in the ART calculation was 2.43×10^{17} n/cm^2 at the 1/4T location and 1.17×10^{17} n/cm^2 at the 3/4T location. The initial RT_{NDT} for the limiting weld was -5 degrees Fahrenheit. The margin term used in calculating the ART for the limiting weld was 59 degrees Fahrenheit.

The staff performed an independent calculation of the ART values for the limiting materials using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for the Dresden, Units 2 and 3, and the Quad Cities, Units 1 and 2, reactor vessels is the lower intermediate to lower shell circumferential welds in the

Dresden, Unit 3, vessel (heat 299L44). The staff's calculated ART values for the limiting materials agreed with the licensee's calculated ART values.

Substituting the ART values for Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2, into equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraphs IV.A.2 and IV.A.3 of Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120 degrees Fahrenheit for normal operation and by 90 degrees Fahrenheit for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 23 degrees Fahrenheit for Dresden and Quad Cities provided by the licensee, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation, hydrostatic pressure test and leak test.

GE developed non-beltline P-T curves for a conservatively large BWR/6 vessel. These curves may be applied generically to other vessels by adding the highest initial RT_{NDT} value of the applicable vessel component in the subject vessel to the pressure vs. temperature minus RT_{NDT} ($T-RT_{NDT}$) values from the GE analysis. Clinton Power Station used this approach to develop separate vessel bottom head curves, and this methodology was accepted by the staff in a letter from D. Pickett to P. Telthorst, dated October 23, 1996. The Dresden and Quad Cities P-T limits for the vessel bottom head were also based on the GE analysis. The highest initial RT_{NDT} for the Dresden and Quad Cities bottom head materials was 60 degrees Fahrenheit. The staff's calculations verified that the shift was consistent with the licensee's values, and that the P-T curves are at least as conservative as that which would be generated using the methodologies of SRP Section 5.3.2 and Appendix G of the ASME Code.

The ES welds are not the most limiting material among the group of four vessels and, therefore, were not used to develop the P-T curves. However, BAW 2258 and 2259, "Evaluation of RT_{NDT}, USE, and Chemical Composition of Core Region Electroslag Welds for Dresden, Units 2 and 3, and Quad Cities, Units 1 and 2," January 1996, were referenced by the licensee, and reviewed by the staff in support of the submittal. These documents contain the licensee's assessment of the best estimate chemistry and the initial RT_{NDT} of the ES welds. The licensee's best estimate of the initial RT_{NDT} was the mean value of surveillance weld data as well as drop weight data reported in the weld procedure qualifications (PQs) of ES welds fabricated in the same time frame as the Dresden and Quad Cities ES welds. The licensee determined the mean value of the initial RT_{NDT} to be 23.1 degrees Fahrenheit with a standard deviation of 13.0 degrees Fahrenheit. The licensee's best estimate of the chemistry was also the mean of surveillance weld data and weld PQs of ES welds fabricated in the same time frame as the Dresden and Quad Cities ES welds.

The licensee determined the mean values of the copper (Cu) and nickel (Ni) contents to be 0.19 percent and 0.31 percent, respectively, with standard deviations of 0.048 percent for Cu and 0.051 percent for Ni.

The staff considers the licensee's proposed method of determining the initial RT_{NDT} acceptable. The staff verified the initial RT_{NDT} and standard deviation values proposed by the licensee. The uncertainty in this estimate is accounted for by using the standard deviation of the initial RT_{NDT} in the margin term of the ART calculation.

The staff does not consider the initial proposal by the licensee for determining the best-estimate chemistry acceptable because it did not meet the criteria in RG 1.99, Revision 2, nor was an alternative provided to the NRC. This RG indicates that the best-estimate values of copper and nickel for the material will normally be the mean of the measured values for weld samples made with the same weld wire heat number as the vessel weld. If these data are not available, a conservative estimate (mean plus one standard deviation) based on generic data may be used if justified. The licensee could not identify the particular heats of weld wire that were used to fabricate the vessel ES welds. The licensee proposed generic best-estimate values based on the average of all the available data. However, RG 1.99, Revision 2, recommends the best-estimate copper and nickel values should be the mean plus one standard deviation values, when generic values are being used to determine the amount of embrittlement. The staff informed the licensee of its position in an RAI dated December 9, 1996.

In response to the staff RAI, the licensee reevaluated its position and will treat the data as generic. The revised best estimate chemistry will result from the addition of one standard deviation to the mean of the data. This methodology is consistent with RG 1.99, Revision 2, Position 1.1. As mentioned, the licensee has committed to provide a revision to the supporting GE analysis and revised ART tables for 18, 20 and 22 EFPY by April 18, 1997. However, since the revisions do not affect the limiting material, they will not affect the P-T limits.

3.0 CONCLUSION

The staff has performed an independent analysis to verify the licensee's proposed P-T limits. The staff concludes that the proposed P-T limits, including the separate P-T limits for the vessel bottom head, are valid to 22 EFPY since the limits conform to the requirements of Appendix G of 10 CFR Part 50 and GL 88-11. Hence, the proposed P-T limits may be incorporated in the Dresden, Units 2 and 3, and the Quad Cities, Units 1 and 2, TSs. In addition, the proposed changes in the Bases section of the TSs are consistent with the P-T limits change, therefore, they are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve 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. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 66703). Accordingly, the amendments meet 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 the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: A. Lee

Date: February 28, 1997

7.0 REFERENCES

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

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits.

3. Code of Federal Regulations, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

4. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and its Impact on Plant Operations," July 12, 1988.

5. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Appendix G for Nuclear Power Plant Components, Division 1, "Protection Against Nonductile Failure."

6. Letter from P. Piet (ComEd) to USNRC Document Control Desk, Subject: Dresden Nuclear Power Station Units 2 and 3 Quad Cities Nuclear Power Station Units 1 and 2 Changes to Pressure-Temperature (P-T) Curves, dated September 20, 1996.

7. Letter from J. S. Perry (ComEd) to USNRC Document Control Desk, Subject: Dresden Station Units 2 and 3 Quad Cities Units 1 and 2 ComEd Response to NRC Request for Additional Information: P-T Curves, dated January 21, 1997.

8. Letter from D. Pickett to P. Telthorst, Subject: Issuance of Amendment No. 109 to Facility Operating License No. NPF-62 - Clinton Power Station, Unit 1 (TAC No. M94887), dated October 23, 1996.