February 4, 2000 Template No. NRR-058

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: QUAD CITIES - ISSUANCE OF AMENDMENTS - REVISED PRESSURE-TEMPERATURE LIMITS (TAC NOS. MA7138 AND MA7139)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 195 to Facility Operating License No. DPR-29 and Amendment No. 191 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 November 12, 1999, as supplemented by letter dated January 10, 2000.

The amendments change Technical Specification (TS) 3/4.6.K to revise the reactor pressure boundary pressure-temperature limits, change TS 3/4.12.C to delete a special test exception which allows performance of the hydrostatic test above 212 degrees Fahrenheit while in Mode 4, and change TS 3/4.6.P to clarify the operability requirements for the residual heat removal system during the hydrostatic test.

Your letter of November 12, 1999, also requested an exemption from the requirements of 10 CFR 50.60. That request is being handled concurrently with your amendment request, but as a separate action.

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

Sincerely,

/RA/

Stewart N. Bailey, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-254 and 50-265

Enclosures: 1. Amendment No. 195 to DPR-29
2. Amendment No. 191 to DPR-30
3. Safety Evaluation

cc w/encls: See next page

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O. Kingsley
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Units 1 and 2

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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. 195
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 November 12, 1999, as supplemented by letter dated January 10, 2000, 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. <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 195 , 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

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: February 4, 2000

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. 191
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 November 12, 1999, as supplemented by letter dated January 10, 2000, 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. 191 , 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

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 4, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 195 AND 191

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
XIII	XIII
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	3/4.6-21a
3/4.6-21b	3/4.6-21b
3/4.6-21c	--
3/4.6-21d	--
3/4.6-27	3/4.6-27
B 3/4.6-5	B 3/4.6-5
B 3/4.6-6	B 3/4.6-6
B 3/4.6-7	B 3/4.6-7
3/4.12-3	3/4.12-3
3/4.12-4	--
B 3/4.12-1	B 3/4.12-1
B 3/4.12-2	--
B 3/4.12-3	--

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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 Figure 3.6.K-1 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-2.

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-2, 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-2.

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-3.

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-3, 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 32 EFPY

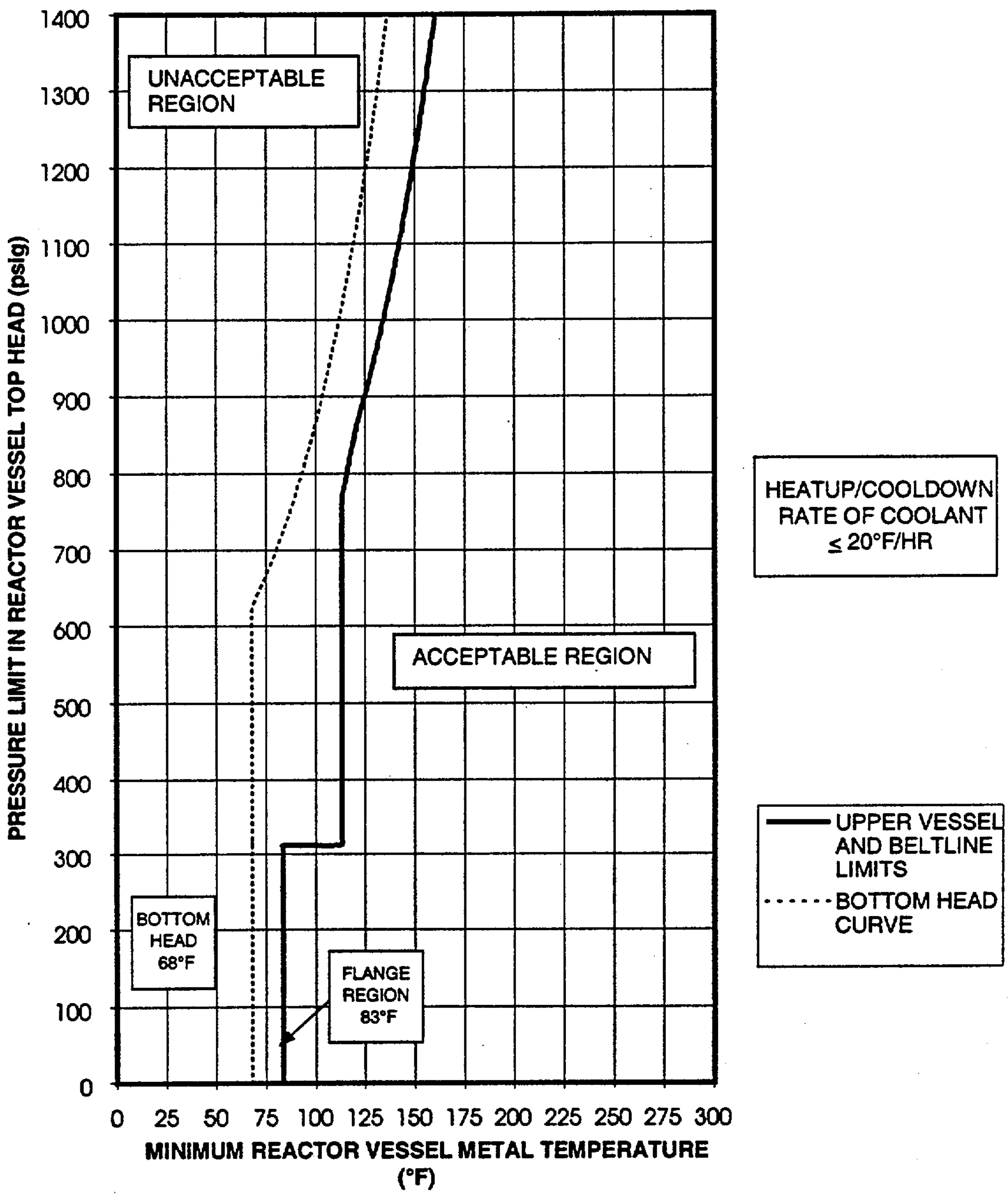


FIGURE 3.6.K-2
PRESSURE – TEMPERATURE LIMITS FOR NON-NUCLEAR
HEATUP/COOLDOWN – VALID TO 32 EFPY

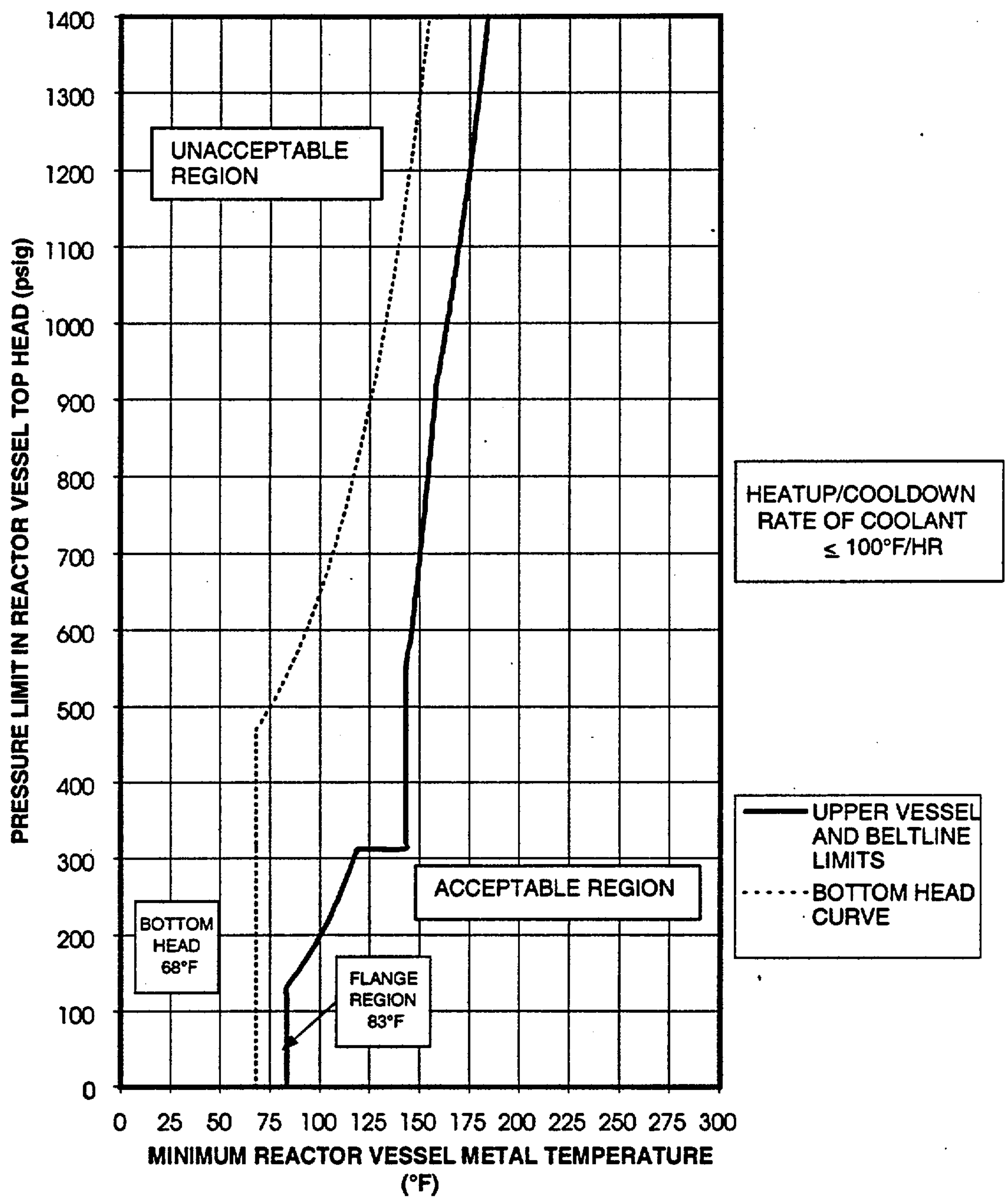
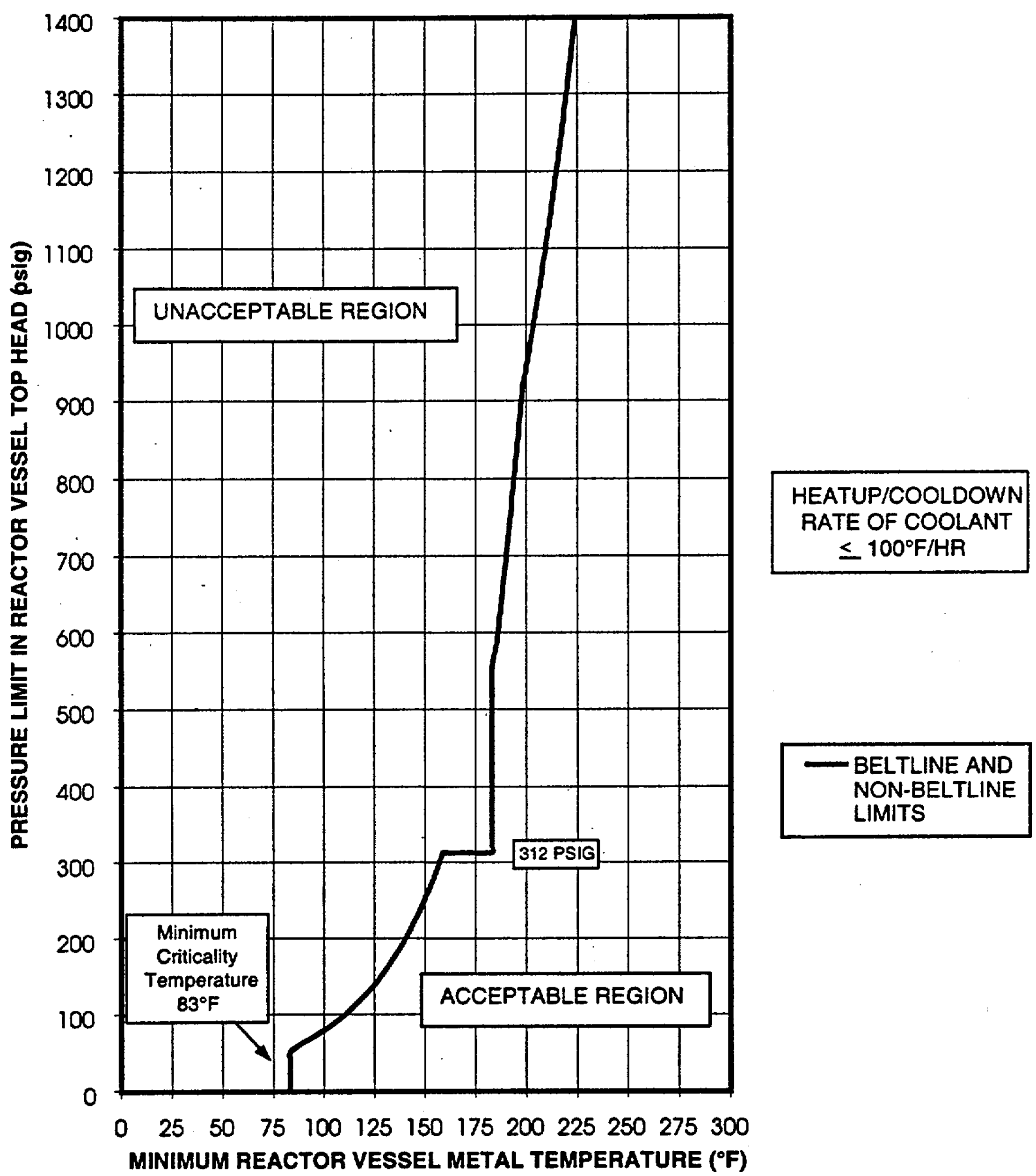


FIGURE 3.6.K-3
PRESSURE – TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS – VALID TO 32 EFPY



3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a][b] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

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a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat.

b The RHR shutdown cooling subsystem(s) is not required to be OPERABLE and may be removed from operation during hydrostatic testing.

BASES

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, Figure 3.6.K-1 Non-Nuclear Heatup/Cooldown, Figures 3.6.K-2 and Core Critical Operation Figure 3.6.K-3. 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

BASES

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.

Figure 3.6.K-1 Pressure Testing

As indicated in Figure 3.3.6.K-1 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 32 effective full power years (EFPY) are presented.

Figure 3.6.K-1 is governing for applicable pressure testing with a maximum heatup/cooldown rate of 20°F/hour.

Figure 3.6.K-2 - Non-Nuclear Heatup/Cooldown

Figure 3.6.K-2 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-3 - Core Critical Operation

The core critical operation curve shown in Figure 3.6.K-3, 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.

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-3 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.

3.12 - LIMITING CONDITIONS FOR OPERATION | 4.12 - SURVEILLANCE REQUIREMENTS

[DELETED]

BASES

3/4.12.A PRIMARY CONTAINMENT INTEGRITY

The requirement for PRIMARY CONTAINMENT INTEGRITY is not applicable during the period when open vessel tests are being performed during the low power PHYSICS TESTS. Low power PHYSICS TESTS during OPERATIONAL MODE 2 may be required to be performed while still maintaining access to the primary containment and reactor pressure vessel. Additional requirements during these tests to restrict reactor power and reactor coolant temperature provide protection against potential conditions which could require primary containment or reactor coolant pressure boundary integrity.

3/4.12.B SHUTDOWN MARGIN Demonstrations

Performance of SHUTDOWN MARGIN demonstrations with the vessel head removed requires additional restrictions in order to ensure that criticality does not occur. These additional restrictions are specified in this LCO. SHUTDOWN MARGIN tests may be performed while in OPERATIONAL MODE 2 in accordance with Table 1-2 without meeting this Special Test Exception. For SHUTDOWN MARGIN demonstrations performed while in OPERATIONAL MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM, or must be verified by a second licensed operator or other technically qualified individual. To provide additional protection against inadvertent criticality, control rod withdrawals that are "out-of-sequence", i.e., do not conform to the Banked Position Withdrawal Sequence, must be made in individual notched withdrawal mode to minimize the potential reactivity insertion associated with each movement. Because the reactor vessel head may be removed during these tests, no other CORE ALTERATION(s) may be in progress. This Special Test Exception then allows changing the Table 1-2 reactor mode switch position requirements to include the Startup or Hot Standby position such that the SHUTDOWN MARGIN demonstrations may be performed while in OPERATIONAL MODE 5.

3/4.12.C Deleted

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 191 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated November 12, 1999, as supplemented by letter dated January 10, 2000, Commonwealth Edison Company (ComEd, or the licensee) submitted changes related to the pressure-temperature (P-T) limits in the Quad Cities Nuclear Power Station, Units 1 and 2, (Quad Cities) Technical Specifications (TSs). The licensee revised the P-T limits to provide new limits that are valid to 32 effective full power years (EFPY). The proposed changes are based, in part, on the use of Code Cases N-588 and N-640, which were reviewed by the staff. Code Case N-588 permits the postulation of a circumferentially-oriented flaw (in lieu of an axially-oriented flaw) for the evaluation of the circumferential welds in reactor pressure vessel (RPV) P-T limit curves. Code Case N-640 permits the use of an alternate reference fracture toughness for reactor vessel materials in determining the P-T limits. Approval of Code Cases N-588 and N-640 for Quad Cities is covered by an exemption to 10 CFR 50.60, which was issued under a separate cover. The January 10, 2000, letter did not change the original proposed no significant hazards consideration determination.

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P-T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1; GL 92-01, Revision 1, Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Revision 2); and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in nil-ductility transition temperature and the decrease in upper-shelf energy (USE) resulting from neutron radiation. GL 92-01, Revision 1, requested that licensees submit their RPV data for their plants to the staff for review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves and as the basis for the staff's review of pressurized thermal shock (PTS) assessments

(10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code).

SRP Section 5.3.2 provides an acceptable method of determining the P-T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

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

2.0 EVALUATION

2.1 Pressure Temperature Limits

2.1.a Licensee Evaluation

The licensee submitted detailed calculations for ART and P-T limit curves valid for up to 32 EFPYs. The licensee provided bounding analyses for Quad Cities, Units 1 and 2, reactor vessels by (1) assuming the fluence is 5.1×10^{17} n/cm^2, which bounds the fluence of 3.5×10^{17} n/cm^2 for Unit 1 and the fluence of 4.9×10^{17} n/cm^2 for Unit 2, and (2) manipulating the thermal gradient stress at the 1/4T for heatup so that P-T limits at 3/4T can be conservatively evaluated at 1/4T using the ART of 1/4T. The licensee determined that the most limiting material at the 1/4T is the axial welds that were fabricated using electro-slag welding process. The licensee calculated an ART of 100 degrees Fahrenheit for the limiting material at the 1/4 T location at 32 EFPYs based on the ΔRT_{NDT} value of 34 degrees Fahrenheit at the 1/4T, the initial RT_{NDT} of 23.1 degrees Fahrenheit, and a plant-specific margin term of 42.8 degrees Fahrenheit (σ_I = 13 degrees Fahrenheit and σ_Δ = 17 degrees Fahrenheit).

Based on the ART of 100 degrees Fahrenheit for the limiting beltline material, the licensee used the methodology of Appendix G, as modified by Code Cases N-588 and N-640, to calculate the P-T limits. For the bottom head, the licensee used the results from a generic analysis, as modified by Code Cases N-588 and N-640, to calculate the P-T limits. The ART for the bottom head is 60 degrees Fahrenheit, the same as its initial RT_{NDT}, because the bottom head is a non-beltline material. The licensee's evaluation applied to both units and the proposed P-T limits are for both units.

2.1.b Staff Evaluation

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Revision 2. Based on these calculations, the staff verified that the licensee's limiting material for Quad Cities Nuclear Power Station, Units 1 and 2, reactor vessel is the electro-slag axial welds. The staff's calculated ART value for the limiting material agrees with the licensee's calculated ART value. Further, except for the bounding fluence value assumed by the licensee, the material data for all beltline materials reported in this submittal is consistent with that in the NRC's reactor vessel integrity database (RVID).

For the beltline P-T limits, the licensee provided detailed calculations for a sample point on each P-T curve. The staff examined the sample calculations, and verified that the licensee's proposed beltline P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50 as modified by Code Cases N-588 and N-640. It should be noted that the staff did not approve the simplified approach of manipulating the thermal gradient stress at the 1/4T for heatup so that P-T limits at 3/4T can be evaluated at 1/4T using the ART of 1/4T. The qualitative analysis for using this approach is not sufficient to demonstrate the validness of this simplified approach for all vessels. However, the staff has performed conventional calculations for the heatup P-T limit curve not using this assumption to ensure that the heatup P-T limits are bounded by the proposed P-T limit curve. Further, since the limiting material is axial welds and the use of the modified stress intensity factor K_{Im} due to membrane tension for axial flaws are not significantly different from that in the 1989 ASME Code, the benefit from applying Code Case N-588 in this application is not significant.

For the bottom head P-T limits, the licensee used the results from a generic analysis, as modified by Code Cases N-588 and N-640, to calculate the P-T limits. Since the bottom head P-T limits were derived from the P-T limits of a generic vessel, calculations were provided in the submittal for a series of P-T points on the P-T limit curve. The staff determined that the bottom head P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50 as modified by Code Cases N-588 and N-640 because (1) the licensee has performed analysis to demonstrate that the generic analysis is bounding, and (2) the licensee has provided detailed calculations to demonstrate that the bottom head P-T limits were developed according to the requirements and bound all other vessel discontinuities.

Thus, the staff determined that the licensee's proposed P-T limit curves were acceptable, since they meet the intent of the requirements of 10 CFR 50.60 and 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 Quad Cities, Units 1 and 2, the staff has determined that the proposed P-T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

2.2 Special Test Exception

The licensee proposed to delete the special test exception TS 3/4.12.C, "Inservice Leak and Hydrostatic Testing Operation," which allows the temperature to exceed 212 degrees Fahrenheit in Mode 4 to perform the test. A review of the revised P-T limits reveals that the licensee will no longer be required to exceed 212 degrees Fahrenheit in order to perform the system hydrostatic test. Therefore, the staff finds the proposed change to be acceptable.

The licensee also proposed to change footnote (b) to TS 3/4.6.P, "Residual Heat Removal - COLD SHUTDOWN," to clarify that the residual heat removal system was not required to be operable during the hydrostatic test. This change is necessary due to the deletion of TS 3/4.12.C, which allowed suspension of the requirements of TS 3/4.6.P during the hydrostatic test. The staff finds the change to TS 3/4.6.P to be acceptable.

2.3 Summary

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, hydrotest, and criticality satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Cases N-588 and N-640, and Appendix G of 10 CFR Part 50 for 32 EFPYs. The proposed P-T limits also satisfy GL 88-11, because the method in RG 1.99, Revision 2, was used to calculate the ART. Hence, the proposed P-T limit curves may be incorporated into the Quad Cities, Units 1 and 2, TSs. Further, the staff finds the proposed deletion of the special test exception and the proposed clarification of the residual heat removal system operability requirements to be acceptable.

3.0 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.

4.0 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 (64 FR 70081). 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.

5.0 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.

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