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

September 19, 2000

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: DRESDEN - ISSUANCE OF AMENDMENTS - REVISED PRESSURE-TEMPERATURE LIMITS (TAC NOS. MA8346 AND MA8347)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 179 to Facility Operating License No. DPR-19 and Amendment No. 174 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3. The amendments are in response to your application dated February 23, 2000, as supplemented by letters dated June 19 and July 17, 2000.

The amendments change Technical Specification (TS) 3/4.6.K to revise the reactor pressure-temperature (P-T) limits; change TSs 1.0 and 3/4.12.C to delete a special test exception that allowed the hydrostatic test to be performed above 212 degrees Fahrenheit while in Mode 4; and add a condition to the Unit 2 and 3 licenses to specify expiration dates for the P-T limits.

The NRC staff had technical issues with the dosimetry and methodology used to derive the fluence values used by Commonwealth Edison Company (ComEd) in the proposed licensing action. These issues were summarized in a telefaxsimile sent to ComEd on June 26, 2000. The staff believes that these issues must be resolved in order to justify applying the fluence values for a full 32 effective full power years. As an interim solution, ComEd proposed that NRC approve the P-T limits for a shorter, more defensible period. The staff concluded, as discussed in the enclosed Safety Evaluation, that several conservatisms provide a reasonable assurance of safety and an acceptable justification for applying the fluence values for an interim period. The license conditions included with this amendment allow the use of the P-T limits during the interim period suggested by ComEd. These conditions were agreed to by ComEd. Please note that contrary to a statement in a letter from ComEd dated July 17, 2000, the NRC staff was not withholding approval because ComEd's neutron fluence did not reflect the guidance contained in Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." Regulatory Guides describe methods acceptable to the NRC staff and provide guidance. However, compliance with Regulatory Guides is not required. Licensees may propose other methods with their basis.

The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
1. Amendment No. 179 to DPR-19
2. Amendment No. 174 to DPR-25
3. Safety Evaluation

cc w/encls: See next page

The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/RA/

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
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Docket Nos. 50-237 and 50-249

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*See previous concurrence

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DATE	09/14/00	09/19/00	09/07/00	09/15/00	09/14/00	09/19/0

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The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
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*See previous concurrence
** input provided by safety evaluation dated June 22, 2000
DOCUMENT NAME: G:\PDIII-2\dresden\AMDa8346.wpd

OFFICE	PM:LPD3	LA:LPD3	SC:SRXB	SC:EMCB	OGC	SC:LPD3
NAME	LROSSBACH	CMOORE	FAUKSTEWICZ*	KWICHMAN**	C Marco	AMENDIOLA
DATE	09/15/00	09/ /00	09/07/00	06/22/00	09/14/00	09/ /0

OFFICIAL RECORD COPY

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,

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

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NAME	LROSSBACH		CMOORE		KWICHMAN		C Marco		AMENDIOLA	
DATE	09/ 6 /00		09/ /00		09/ /00		09/14 /00		09/ /0	

Subject to change noted.

SC:SRXB	
FAUKSTEWICZ	
09/ 7 /00	

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The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Lawrence W. Rossbach, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures: 1. Amendment No. to DPR-19
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NAME	LROSSBACH	CMOORE	FAUKSTEWICZ*	KWICHMAN**	C Marco	AMENDIOLA
DATE	09/ 8 /00	09/ 8 /00	09/07/00	06/22/00	09/ 14 /00	09/ /0

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O. Kingsley
Commonwealth Edison Company

Dresden Nuclear Power Station
Units 2 and 3

cc:

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Morris, Illinois 60450-9765

Commonwealth Edison Company
Dresden Station Manager
6500 N. Dresden Road
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Dresden Resident Inspectors Office
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Regional Administrator
U.S. NRC, Region III
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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. 179
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 February 23, 2000, as supplemented by letters dated June 19 and July 17, 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 Operating License and 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. 179 , 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
and Operating License

Date of Issuance: September 19, 2000

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. 174
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 February 23, 2000, as supplemented by letters dated June 19 and July 17, 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 Operating License and 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. 174 , 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
and Operating License

Date of Issuance: September 19, 2000

ATTACHMENT TO LICENSE AMENDMENT NOS. 179 AND 174

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

DOCKET NOS. 50-237 AND 50-249

Replace the following page of Operating Licenses Nos. DPR-19 and DPR-25 with the attached revised page. The revised page is identified by amendment number and contains a vertical line in the margin indicating the area of change.

REMOVE	INSERT
Unit 2 - page 3a	Unit 2 - page 3a
Unit 3 - page 6	Unit 3 - page 6

Revise the Appendix "A" Technical Specifications by replacing the pages identified below with the attached pages. The revised pages are identified by amendment number and contain a vertical line in the margin indicating the area of change.

REMOVE	INSERT
VIII	VIII
XIII	XIII
XXVI	XXVI
1-9	1-9
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	--
B3/4.6-6	B3/4.6-6
B3/4.6-7	B3/4.6-7
3/4.12-3	--
3/4.12-4	--
B3/4.12-1	B3/4.12-1
B3/4.12-2	--
B3/4.12-3	--

a.	Surveillance Requirement 4.1.A.2 -	RPS Logic System Functional Test
b.	Surveillance Requirement 4.2.A.2 -	Primary & Secondary Containment Logic System Functional Test
c.	Surveillance Requirement 4.2.J.2 -	Feedwater Pump Trip Logic System Functional Test
d.	Surveillance Requirement 4.6.F.1.b -	Relief Valve Logic System Functional Test
e.	Surveillance Requirement 4.9.A.9 -	Simultaneous Diesel Generator Start
f.	Surveillance Requirement 4.9.A.10 -	Diesel Storage Tank Cleaning (Unit 3 and Unit 2/3 only)

Each of the above Surveillance Requirements shall be successfully demonstrated prior to entering into MODE 2 on the first plant startup following the fifteenth refueling outage (D2R15).

(7) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 163, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

(8) Pressure-Temperature Limit Curves

The pressure-temperature (P-T) limit curves issued by Amendment No.179 are approved for use until November 30, 2001, unless Commonwealth Edison Company, the licensee, obtains approval from the Nuclear Regulatory Commission staff for use beyond November 30, 2001.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated February 2, 1983, September 28, 1987, July 6, 1989, and August 15, 1989.

In addition, the facility has been granted certain exemptions from Sections II and III of Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to the licensee in a letter dated June 25, 1982.

L. Deleted. [Amdt. 87, 7-24-86]

M. Deleted. [Amdt. 85, 12-12-85]

N. By Amendment No. 144, the license is amended to allow, on a one time temporary basis, operation of Dresden, Unit 3, with the corner room structural steel members in the Low Pressure Coolant Injection Corner Rooms outside the Updated Final Safety Analysis Report (UFSAR) design parameters. Operation under these conditions is allowed up to and including the next scheduled refueling outage (D3R14).

The repairs to Dresden, Unit 3, corner room structural steel shall restore the steel design margins to the current UFSAR (updated through Revision 1A) design criteria. The design of the modifications to the Dresden, Unit 3, corner room structural steel members will be based on use of elastic section modules and the structural steel stresses will be limited to 1.6 of the American Institute of Steel Construction (AISC allowables). The modifications to Dresden, Unit 3, corner room structural steel will be implemented during the upcoming D3R14 refueling outage.

During this interim period of operation, should vibratory ground motion exceeding the UFSAR Operating Basis Earthquake (OBE) design parameters, Dresden, Unit 3, will be shut down for inspection and will not start up without prior NRC approval.

O. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 158, are hereby incorporated into this license. Commonwealth Edison Company shall operate the facility in accordance with the Additional Conditions.

P. Pressure-Temperature Limit Curves

The pressure-temperature (P-T) limit curves issued by Amendment No174are approved for use until October 30, 2002, unless Commonwealth Edison Company, the licensee, obtains approval from the Nuclear Regulatory Commission staff for use beyond October 30, 2002.

4. This license is effective as of the date of issuance and shall expire at Mid-night January 12, 2001.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by:

Peter A. Morris, Director
Division of Licensing

Enclosures:
Appendix A - Technical Specifications
Appendix B - Additional Conditions

Date of Issuance: January 12, 1971

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

TABLE 1-2

OPERATIONAL MODES

MODE	MODE SWITCH POSITION[f]	AVERAGE REACTOR COOLANT TEMPERATURE
1. POWER OPERATION	Run	Any temperature
2. STARTUP	Startup/Hot Standby	Any temperature
3. HOT SHUTDOWN	Shutdown[a,e]	> 212°F[d]
4. COLD SHUTDOWN	Shutdown[a,b,e]	≤ 212°F
5. REFUELING[c]	Shutdown or Refuel[a,d]	≤ 140°F

TABLE NOTATIONS

(a) The reactor mode switch may be placed in the Run, Startup/Hot Standby or Refuel position to test the switch interlock functions provided the control rods are verified to remain fully inserted by a second licensed operator or other technically qualified individual.

(b) The reactor mode switch may be placed in the Refuel position while a single control rod drive is being removed from the reactor pressure vessel per Specification 3.10.I.

(c) Fuel in the reactor vessel with one or more vessel head closure bolts less than fully tensioned or with the head removed.

(d) See Special Test Exceptions 3.12.A and 3.12.B.

(e) The reactor mode switch may be placed in the Refuel position while a single control rod is being moved provided the one-rod-out interlock is OPERABLE.

(f) When there is no fuel in the reactor vessel, the reactor is considered not to be in any OPERATIONAL MODE. The reactor mode switch may then be in any position or may be inoperable.

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

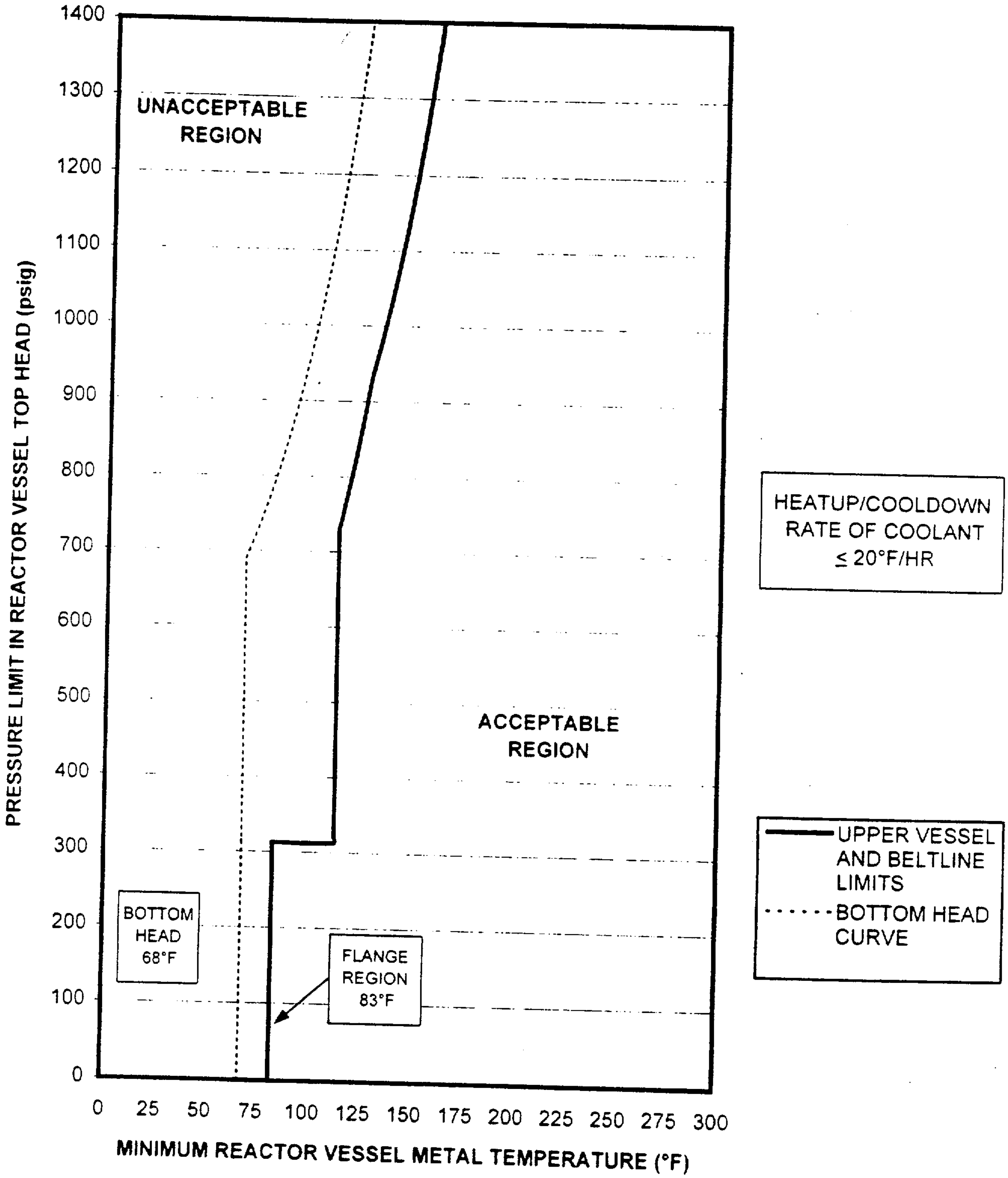


FIGURE 3.6.K-2

PRESSURE – TEMPERATURE LIMITS FOR NON-NUCLEAR HEATUP/COOLDOWN – VALID TO 32 EFPY

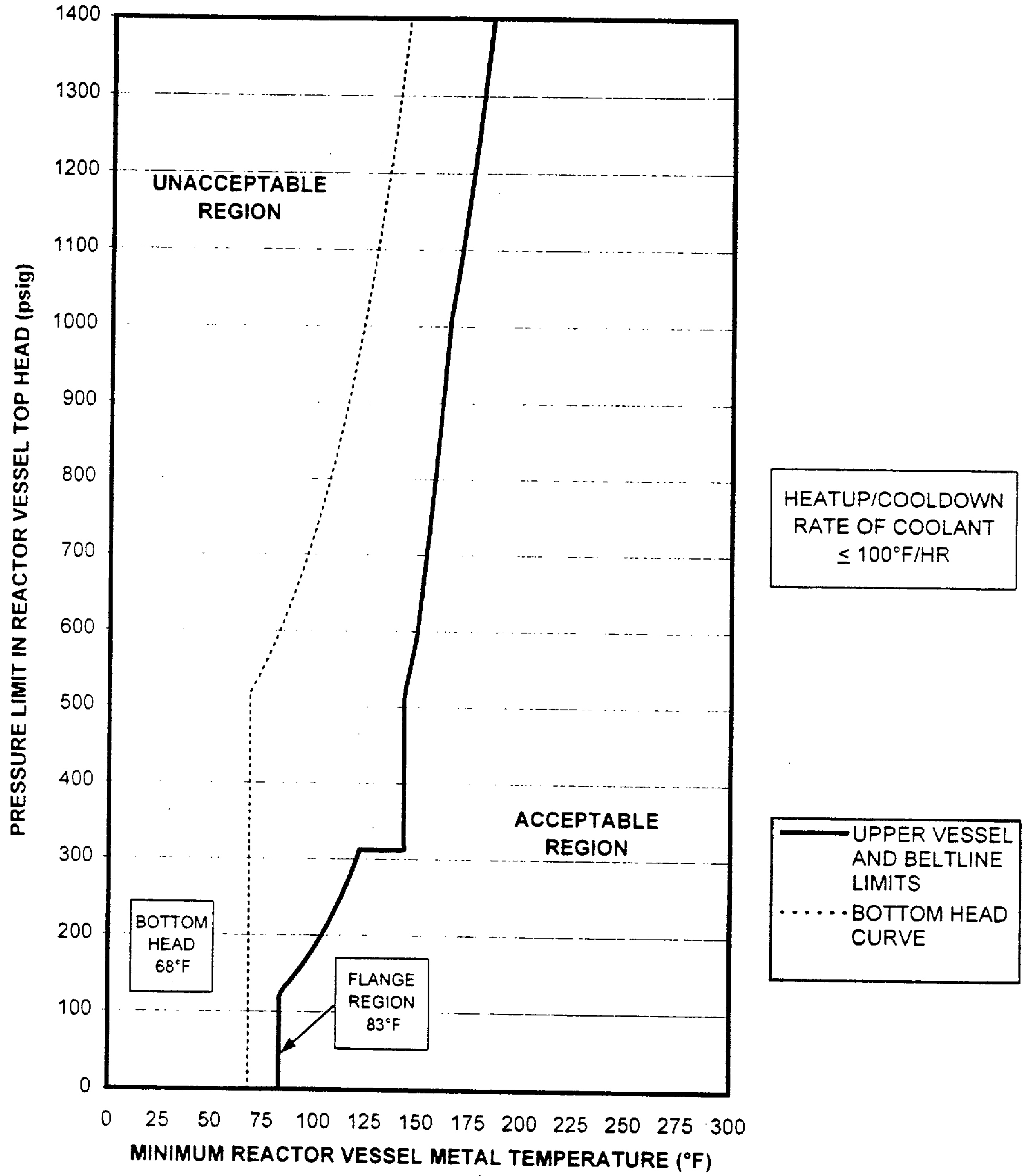
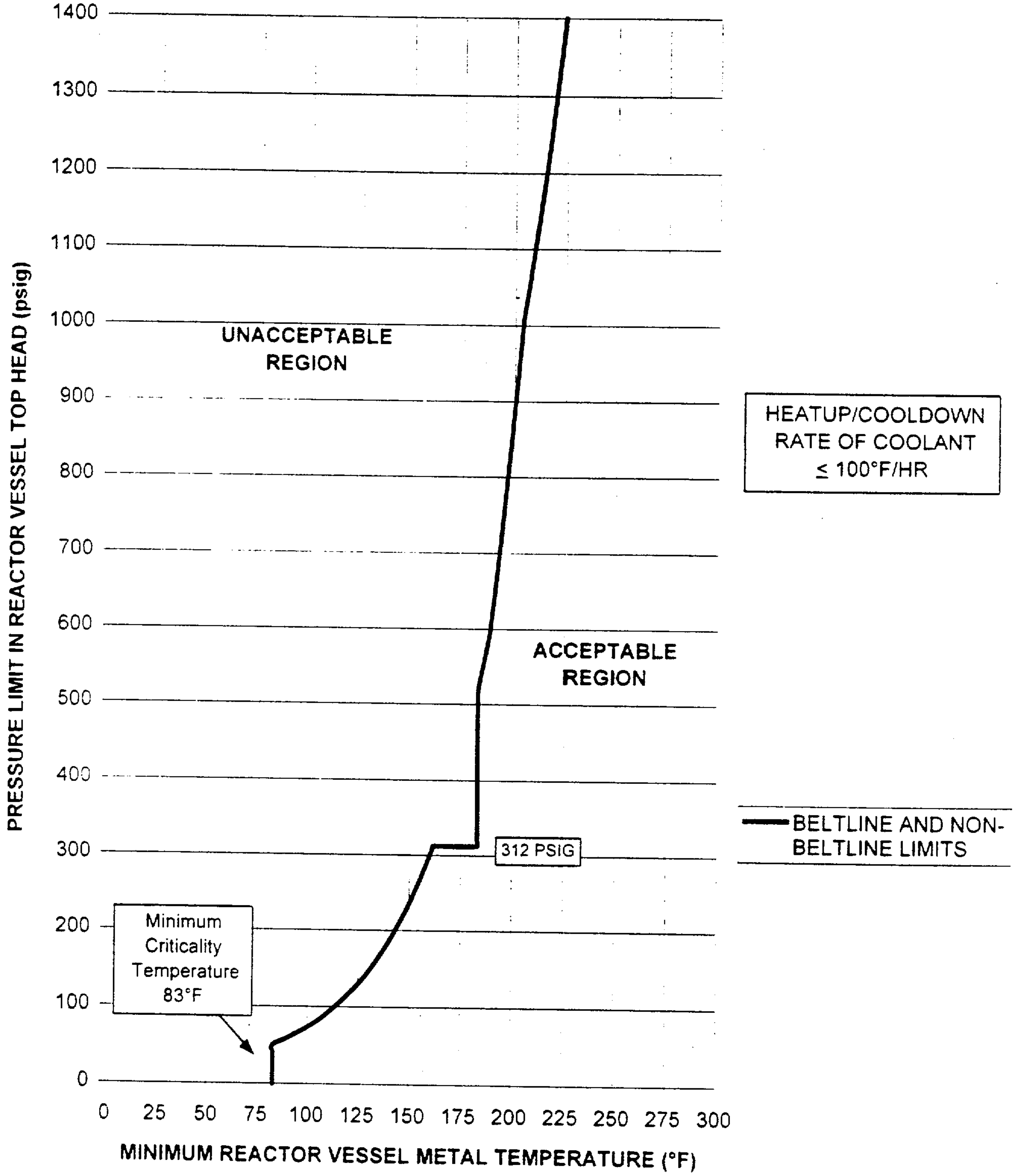


FIGURE 3.6.K-3

PRESSURE – TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS – VALID TO 32 EFPY



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, Figure 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 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 3.6.K-1 for pressure testing, the minimum metal temperature of the reactor vessel shell is 83°F for reactor pressure 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 10 CFR Part 50 Appendix G. Beltline curves as a function of vessel exposure for 32 effective full power years (EFPY) are presented.

BASES

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.

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

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 179 TO FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 174 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

On February 23, 2000, Commonwealth Edison Company (ComEd, the licensee) submitted a license amendment request to update the pressure-temperature (P-T) limit curves for the Dresden Nuclear Power Station, Units 2 and 3. The specific changes to the P-T limits for Dresden, Units 2 and 3, amend Technical Specification (TS) Section 3/4.6.K, "Primary System Boundary," to revise the heatup, cooldown, and inservice test limitations for the reactor pressure vessel (RPV) of each unit to a maximum of 32 effective full power years (EFPY). The proposed changes also amend TS Section 3/4.12, "Special Test Exceptions," to delete Special Test Exception 3/4.12.C, "Inservice Leak and Hydrostatic Testing Operation," which provides for pressure testing at temperatures greater than 212 degrees Fahrenheit when the plants are placed into operational Mode 4, "Cold Shutdown." The proposed changes also include appropriate changes to the TS Bases. By letter dated June 19, 2000, ComEd requested that reference to Special Test Exception 3/4.12.C be deleted from TS Section 1.0, "Definitions." By letter dated July 17, 2000, ComEd proposed that the staff grant an interim approval of the P-T limits to allow ComEd time to complete new neutron fluence calculations. The June 19 and July 17, 2000, letters are within the scope of the original notice and do not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in Appendix G of Part 50 to Title 10 of the *Code of Federal Regulations* (10 CFR Part 50, Appendix G), to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The Appendix to Part 50 requires the P-T limits for an operating plant to be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Appendix G to the Code) were applied. The methodology of Appendix G to the Code postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum applied stress. For materials in the beltline and upper and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. For the case of evaluating RPV nozzles, the surface flaw is postulated to propagate parallel to the axis of the nozzle's corner radius. The basic parameter in Appendix G to the Code for calculating P-T limit curves is the stress intensity factor, K_1, which is a function of the stress state and flaw configuration. The methodology requires that licensees determine the K_{1a} factors, which vary as a function of temperature, from the reactor coolant system (RCS) operating temperatures, and from the adjusted reference temperatures (ARTs) for the limiting materials in the RPV. Thus, the critical locations in the RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the points of the crack tips if the flaws are initiated and grown from the inside and outside surfaces of the vessel, respectively. Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (May 1988), provides an acceptable method of calculating ARTs for ferritic RPV materials; the methods of RG 1.99, Revision 2, include methods for adjusting the ARTs of materials in the beltline region of the RPV, where the effects of neutron irradiation may induce an increased level of embrittlement in the materials.

The methodology of Appendix G to the Code requires that P-T curves must satisfy a safety factor of 2.0 on primary membrane and bending stresses during normal plant operations (including heatups, cooldowns, and transient operating conditions), and a safety factor of 1.5 on primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 to 10 CFR Part 50, Appendix G, provides the staff's criteria for meeting the P-T limit requirements of Appendix G to the Code and 10 CFR Part 50, Appendix G. Table 1 to 10 CFR Part 50, Appendix G, also establishes minimum temperature requirements for operating the RCS at nuclear power facilities. P-T limit curves are generated by using the more conservative of the P-T combinations established by the Appendix G and minimum temperature requirements.

On August 25, 2000, pursuant to 10 CFR 50.12, the NRC granted two exemptions to allow ComEd to deviate from the requirements of 10 CFR Part 50 Appendix G, and to use Code Cases N-588 and N-640 as the bases for generating the Dresden P-T limit curves effective to 32 EFPY.* The staff's assessment of ComEd's proposed P-T limit curves effective to 32 EFPY is, in part, based on the these exemptions and is given in Section 3.0 of this SE.

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* Approval to use Code Case N-588 allows licensees to evaluate a circumferential weld based on the tensile stresses associated with a postulated circumferential flaw in the weld, and approval to use Code Case N-640 allows licensees to use the lower bound static initiation fracture toughness value equation (K_{1c} equation) as the basis for establishing the P-T limits in lieu of using the lower bound crack arrest fracture toughness value equation (K_{1a} equation), which is the method invoked by Appendix G to the Code. The staff's bases for approving these exemptions is given in the SE of August 25, 2000.

3.0 EVALUATION

3.1 Request to Update the P-T Limit Curves for Dresden, Units 2 and 3, Effective to 32 EFPY

For the Dresden RPVs, the licensee provided the P-T limit curves for normal operating conditions and pressure testing conditions effective to 32 EFPY. For the normal operating conditions with the core not critical, and for pressure testing conditions, individual P-T curves were proposed for the lower head in addition to the composite curves proposed for the beltline and nozzles regions of the RPVs. To test the validity of the licensee's proposed curves, the staff performed an independent assessment of the licensee's submittal. The staff applied the methodologies of the 1995 Edition of Appendix G to the Code and 10 CFR Part 50, Appendix G, as modified by the methodologies of ASME Code Cases N-588 and N-640, as the bases for its independent assessment. For the evaluation of the RPV nozzles, the staff also modified the methods of Appendix G to the Code by the nozzle evaluation methods proposed in Appendix 5 of Welding Research Council Bulletin WRC-175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials" (August 1972).

The staff's assessment also included an independent calculation of the ART values for both the 1/4T and 3/4T locations of the Dresden RPV beltline regions based on ComEd's neutron fluences for Dresden effective to 32 EFPY. For the evaluation of the limiting beltline materials, the staff confirmed that the ARTs and P-T limit curves were based on ComEd's projected neutron fluences for 32 EFPY, and on the methodology of RG 1.99, Revision 2. For the evaluation of the limiting material in the limiting nozzle and lower head evaluations, the staff applied the plant specific design basis data provided by the licensee.

ComEd developed the neutron fluence values used for the 32 EFPY P-T curves by extrapolation from earlier fluence values obtained from vessel surveillance capsules removed from Units 2 and 3 in 1981 and 1982. Staff review identified concerns with both the dosimetry measurements and the fluence calculations. Due to these concerns, the staff did not believe that the licensee had provided a defensible basis for applying these fluence values for the full 32 EFPY. In particular, the staff did not have confidence in the application of the calculated fluence values to the more remote years. These concerns were discussed with the licensee on June 28, 2000, and July 6, 2000. Because of the difficulty of quickly resolving the staff's concerns with the fluence values, ComEd proposed in their letter dated July 17, 2000, that NRC grant interim approval of the 32 EFPY P-T limits until November 30, 2001, for Unit 2, and October 31, 2002, for Unit 3. This would allow ComEd sufficient time to complete the new fluence calculations that they are currently working on in support of a future power uprate amendment request. To apply their proposal, ComEd agreed to the following license conditions. For Unit 2, "The pressure-temperature (P-T) limit curves issued by Amendment No. 179 are approved for use until November 30, 2001, unless Commonwealth Edison Company, the licensee, obtains approval from the Nuclear Regulatory Commission staff for use beyond November 30, 2001." For Unit 3, "The pressure-temperature (P-T) limit curves issued by Amendment No. 174 are approved for use until October 30, 2002, unless Commonwealth Edison Company, the licensee, obtains approval from the Nuclear Regulatory Commission staff for use beyond October 30, 2002." The staff finds these license conditions acceptable.

In reviewing ComEd's proposed interim resolution of the fluence issue, the staff considered that: (1) both Dresden units adopted low leakage loadings early in their operating life which provides a fifteen to twenty percent conservatism for the fluence value; (2) both units are at approximately 18 EFPY while the proposed P-T limit curves are estimated for 32 EFPY, thus, providing a forty percent conservatism; and (3) the licensee stated that the values for Dresden compare favorably with the results of similar plants operated in a similar fashion. The staff concluded that these conservatisms provide a reasonable assurance of safety and provide an acceptable justification for applying the fluence values for an interim period.

The staff determined that ComEd's P-T limit methods, in combination with the license condition, were based on conservative assumptions that made ComEd's proposed P-T limit curves as conservative or slightly more conservative than the P-T limit curves generated by the staff. The staff also confirmed that the ComEd's P-T limit curves included appropriate minimum temperature requirements that were at least as conservative as those required in Table 1 to 10 CFR Part 50, Appendix G, as exempted and modified by the Code Case methods. The staff finds it acceptable to use the proposed 32 EFPY P-T curves for the proposed time period.

3.2 Special Test Exception 3/4.12.C

The licensee proposed to delete TS Special Test Exception 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 inservice leak and hydrostatic test. A review of the revised P-T limits reveals that the revised P-T limits no longer require the licensee to exceed 212 degrees Fahrenheit in order to perform these tests. Therefore, this special test exception is no longer needed. The staff used Quad Cities Amendment Nos. 191/195, dated February 4, 2000, as a precedence for removing this special test exception. The staff finds the proposed change to TS 3/4.12.C to be acceptable.

The licensee also proposed to change footnote (d) to TS TABLE 1-2, "OPERATIONAL MODES," in TS Section 1.0, "DEFINITIONS," by deleting reference to Special Test Exception 3/4.12.C. This change is consistent with the removal of TS 3/4.12.C. The staff finds the proposed change to footnote (d) to TS TABLE 1-2 to be acceptable.

4.0 SUMMARY

Based on the staff's review and evaluation of ComEd's proposed P-T limit curves for Dresden, Units 2 and 3, the staff has determined that the proposed P-T limit curves are consistent with the alternate criteria of Code Cases N-588 and N-640, and satisfy the requirements of 10 CFR 50.60(a), "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation;" Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements;" and Appendix G to the 1995 Edition of Section XI of the ASME Code, as exempted by the methods of analyses in Code Cases N-588 and N-640. The staff, therefore, concludes that the updated P-T limit curves proposed by ComEd will continue to provide an acceptable level of margin and safety, and provide sufficient assurance that the Dresden reactors will be operated in a manner that will protect the Dresden RPVs against brittle fracture. The proposed curves are, therefore, approved for incorporation into the Dresden TS, subject to the license conditions discussed above. Based on the staff's review and evaluation of ComEd's proposed deletion of Special Test Exception 3/4.12.C, the staff concludes that deletion of this special test exception and references to it are acceptable. Also, the staff has no comments on the TS Basis changes.

5.0 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.

6.0 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. 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 (65 FR 17911). 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.

7.0 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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Date: September 19, 2000