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

September 5, 1991

Docket Nos. 50-237
and 50-249

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 71488 AND 71489)

In response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," you submitted letters dated November 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991.

As a result of utilizing the methodology described in Revision 2 of Regulatory Guide 1.99, which was recommended in the GL, your October 23, 1989, response proposed revisions to the Technical Specifications pressure/temperature operating limits for both Dresden units. Additional clarifying information was provided in your March 23, 1990 submittal and a revised Bases section was provided in your July 26, 1991 submittal to be consistent with Quad Cities.

The NRC staff has reviewed your application and the Commission has issued the enclosed Amendment No. 114 to Facility Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 111 to Facility Operating License No. DPR-25 for Dresden Unit 3, respectively. Your application also proposed to remove the reactor vessel venting requirement, however, the staff, in a letter dated March 1, 1990, rejected this change because of the uncertainty regarding the basis for this requirement.

The staff identified two open items as a result of its review of your response to GL 88-11 which require resolution.

1. Section IV.B of Appendix G requires that the predicted Charpy upper shelf energy (USE) at end-of-life (EOL) be above 50 ft-lb. Since you could not provide unirradiated USE values for beltline materials for Dresden, Units 2 and 3, the staff could not determine whether the limiting USE at end-of-life satisfies the minimum of 50 ft-lb requirement. The surveillance capsule reports provided a few USE data, however, the data cold not be used in calculating the limiting USE because it could not be correlated to a specific vessel material. The staff needs to know the unirradiated USE of all beltline materials in order to determine the limiting USE.

DF1

DF01
11

2. The reactor vessel materials surveillance program for Dresden, Units 2 and 3, does not comply with ASTM E 185 and 10 CFR 50, Appendix H, because the surveillance specimens could not be correlated with the reactor vessel beltline materials.

You are requested to provide a plan for resolving these two open items within 6 months of the issuance of this letter.

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,

Original Signed By:
L.N. Olshan for

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 114 to DPR-19
2. Amendment No. 111 to DPR-25
3. Safety Evaluation

cc w/enclosures:
See next page

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Mr. Thomas J. Kovach
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
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 November 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991, 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-19 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 114 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 114

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

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
3/4.6-2	3/4.6-2
3/4.6-23	3/4.6-23
B 3/4.6-26	B 3/4.6-26
	B 3/4.6-26a

List of Tables (continued)

List of Figures

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when reactor vessel metal temperature is equal to or above that shown in the appropriate curve of Figure 3.6.1. Figure 3.6.1 is effective through 16 effective full power years. At least six months prior to 16 effective full power years new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 80°F.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Pressurization Temperature

1. Reactor Vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

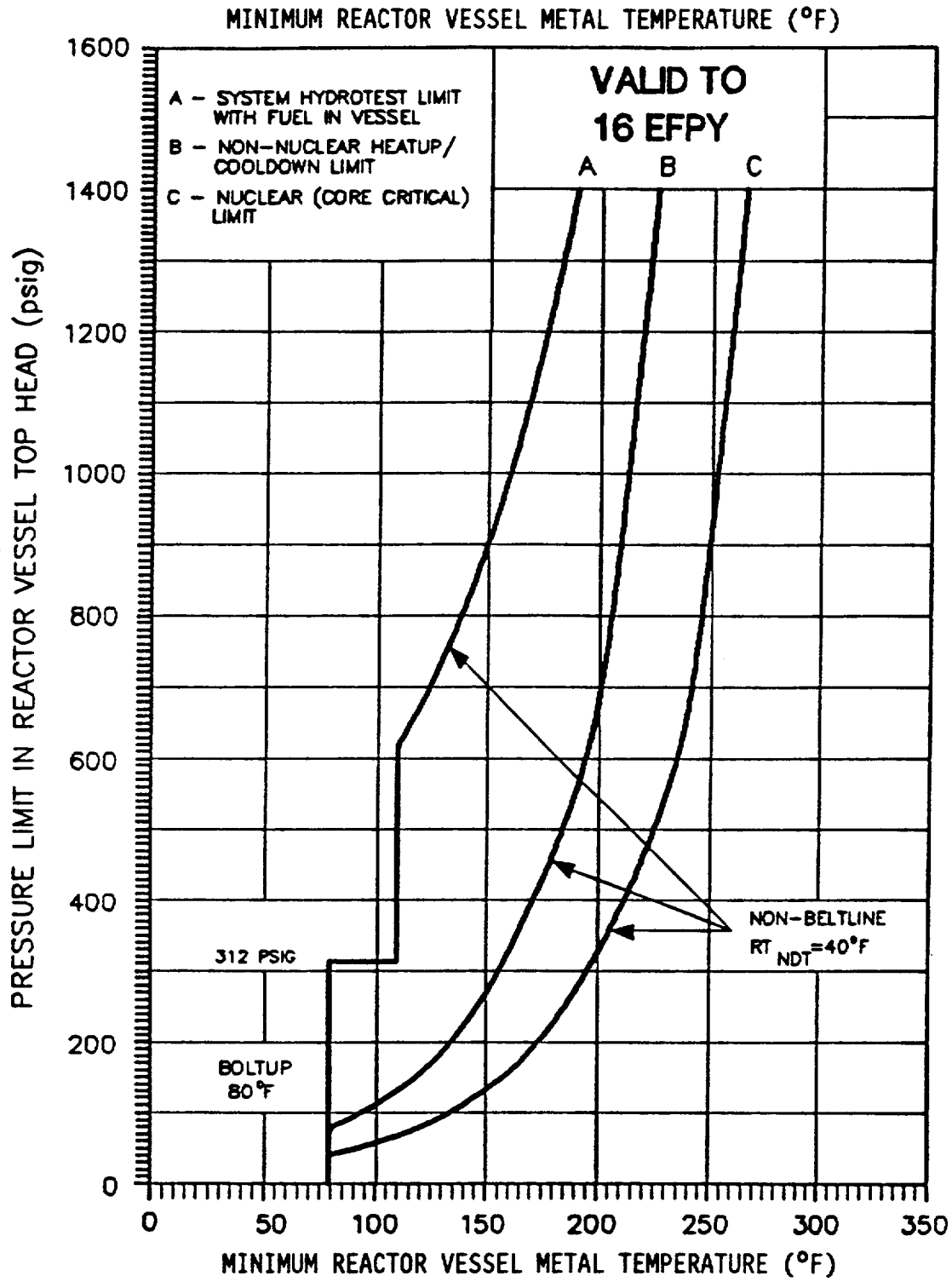
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)
VALID TO
16 EFPY
A
B
C
A – SYSTEM HYDROTEST LIMIT WITH FUEL IN VESSEL
B – NON-NUCLEAR HEATUP/ COOLDOWN LIMIT
C – NUCLEAR (CORE CRITICAL) LIMIT
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
1600
1400
1200
1000
800
600
400
200
0
312 PSIG
BOLTUP
80°F
NON-BELTLINE
RT NDT=40°F
0
50
100
150
200
250
300
350
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

FIGURE 3.6.1.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

B. Pressurization Temperature - The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature limits have been established for the operating conditions to which the reactor vessel can be subjected. Figure 3.6.1 presents the pressure-temperature curves for those operating conditions; Inservice Hydrostatic Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance with Appendix G to 10 CFR 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.

Three 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 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core (between the bottom and the top of active fuel), and is subject to an RT_{NDT} adjustment to account for irradiation embrittlement. The non-beltline and closure flange 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 region is a non-beltline region, it (the closure flange region) is treated separately for the development of the pressure-temperature curves to address 10 CFR 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, the connecting welds and the vertical electroslag welds which terminate immediately below the vessel flange are all 20°F or lower. Therefore, the minimum allowable boltup temperature is established as 80°F (RT_{NDT} + 60°F)

which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in Curve A of Figure 3.6-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 80°F for reactor pressures less than 312 psig. This 80°F minimum boltup temperature is based on an RT_{NDT} of 20°F for the top head plate (most limiting material) and a 60°F conservatism required by the original ASME Code of construction.

At reactor pressures greater than 312 psig the minimum vessel metal temperature is established as 110°F. The 110°F minimum temperature is based on a closure flange region RT_{NDT} of 20°F and a 90°F conservatism required by 10 CFR 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig).

At approximately 620 psig reactor pressure the effects of pressurization become more limiting than the boltup stresses at the closure flange region, as shown by the non-linear portion of Curve A intersecting the vertical 110°F line. The non-linear portion of the curve is dependent on the non-beltline region (which is actually more limiting than the beltline region through a vessel exposure of 22 effective full power years), and based on an RT_{NDT} of 40°F.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.1 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 (non-beltine stresses).

As indicated by the vertical 80°F line, the boltup stresses at the closure flange region are most limiting below approximately 80 psig. Above appproximately 80 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the non-linear portion of Curve B. The non-linear portion of the curve is dependent on non-beltline region (which is actually more limiting than the beltline region through a vessel exposure of 22 effective full power years), and based on an RT_{NDT} of 40°F.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.1, is generated in accordance with 10 CFR 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Curve A or B limits. Since Curve B is more limiting, Curve C is Curve B plus 40°F.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 111
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 November 28, 1988, June 26, 1989, October 23, 1989, March 23, 1990, and July 26, 1991, 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. 111 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 5, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 111

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 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
3/4.6-2	3/4.6-2
3/4.6-23	3/4.6-23
B 3/4.6-26	B 3/4.6-26
	B 3/4.6-26a

List of Tables (continued)

List of Figures

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when reactor vessel metal temperature is equal to or above that shown in the appropriate curve of Figure 3.6.1. Figure 3.6.1 is effective through 16 effective full power years. At least six months prior to 16 effective full power years, new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 100°F.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

B. Pressurization Temperature

1. Reactor Vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

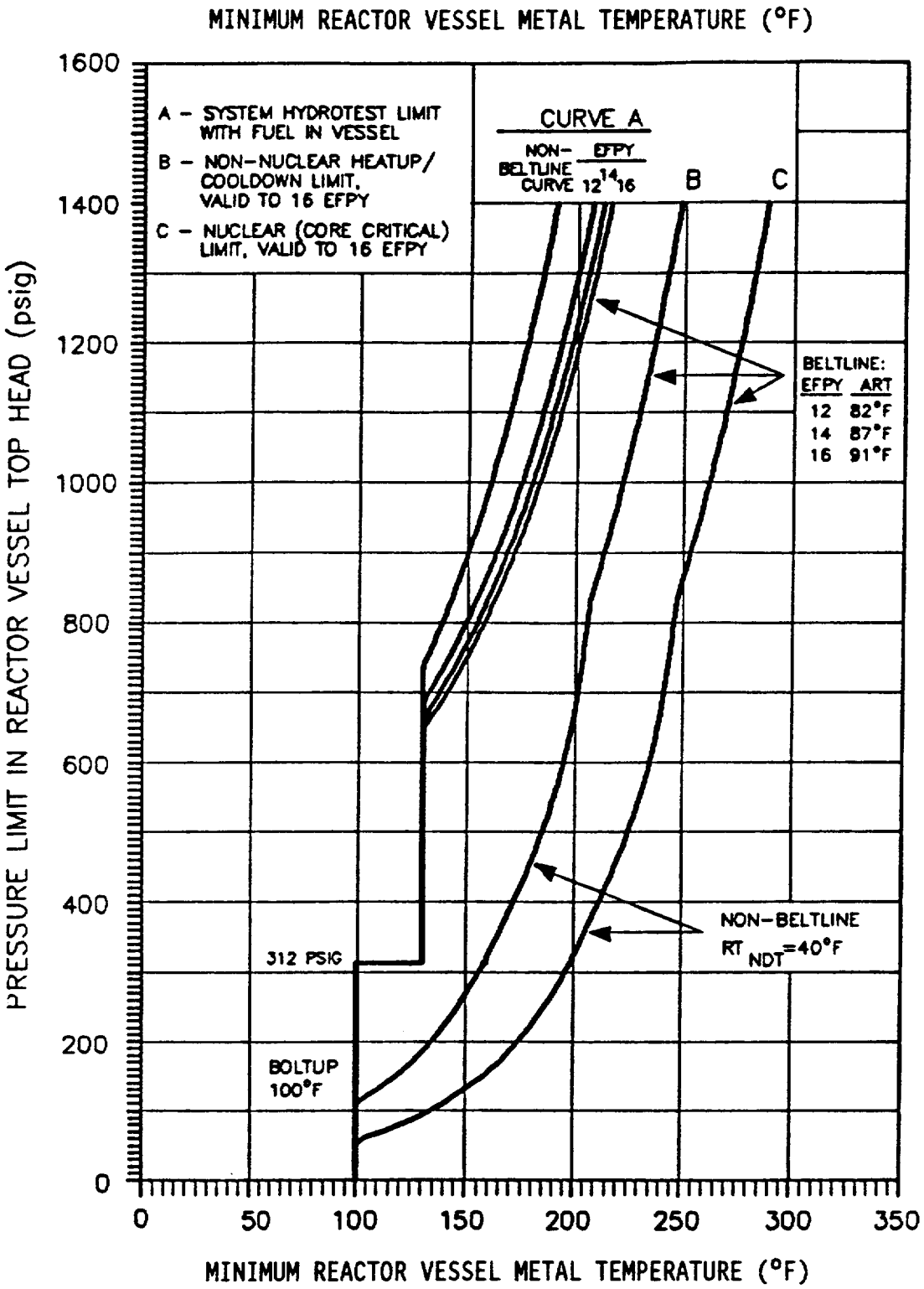
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)
A - SYSTEM HYDROTEST LIMIT WITH FUEL IN VESSEL
B - NON-NUCLEAR HEATUP/ COOLDOWN LIMIT, VALID TO 16 EFPY
C - NUCLEAR (CORE CRITICAL) LIMIT, VALID TO 16 EFPY
CURVE A
NON-BELTLINE CURVE
EFPY
12 14 16
B
C
BELTLINE:
EFPY ART
12 82°F
14 87°F
16 91°F
NON-BELTLINE
RT NDT=40°F
312 PSIG
BOLTUP 100°F
PRESSURE LIMIT IN REACTOR VESSEL TOP HEAD (psig)
0
200
400
600
800
1000
1200
1400
1600
0
50
100
150
200
250
300
350
MINIMUM REACTOR VESSEL METAL TEMPERATURE (°F)

FIGURE 3.6.1.

3.6 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

B. Pressurization Temperature - The reactor vessel is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, pressure-temperature limits have been established for the operating conditions to which the reactor vessel can be subjected. Figure 3.6.1 presents the pressure-temperature curves for those operating conditions; Inservice Hydrostatic Testing (Curve A), Non-Nuclear Heatup/Cooldown (Curve B), and Core Critical Operation (Curve C). These curves have been established to be in conformance with Appendix G to 10 CFR 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.

Three 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 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core (between the bottom and top of active fuel), and is subject to an RT_{NDT} adjustment to account for irradiation embrittlement. The non-beltline and closure flange 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 region is a non-beltline region, it (the closure flange region) is treated separately for the development of the pressure-temperature curves to address 10 CFR 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F)

which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in Curve A of Figure 3.6.1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on an RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction.

At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10 CFR 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig).

At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.1 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.

As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the non-linear portion of Curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltine region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.1, is generated in accordance with 10 CFR 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any Curve A or B limits. Since Curve B is more limiting, Curve C is Curve B plus 40°F.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 111 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

In response to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," Commonwealth Edison Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Dresden Station, Units 2 and 3 Technical Specifications. The licensee's initial response to GL 88-11 was dated November 28, 1988 and supplemented on June 26, 1989. Revisions to the Technical Specifications (TS) were proposed in a letter dated October 23, 1989 and supplemental information was provided in letters dated March 23, 1990 and July 26, 1991. The March 23, 1990, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The July 26, 1991, letter revised the Bases sections for both Dresden and Quad Cities to be consistent, but did not change the technical content. The proposed TS changes the P/T limits for Dresden, Unit 2 and Unit 3 from 6 to 16 effective full power years (EFPY).

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide (RG) 1.99, Revision 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions for operation be included in the Technical Specifications. The P/T limits are among the limiting conditions for operation in the Technical Specifications for all commercial nuclear plants in the United States. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be

tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE).

GL 88-11 requested that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

2.1 Dresden, Unit 2

The staff evaluated the effect of neutron irradiation embrittlement on each of the beltline materials used in the Dresden 2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 16 EFPY was the lower intermediate shell plate B4065-1 with 0.23% copper (Cu), 0.52% nickel (Ni), and an initial RT_{NDT} of 20°F.

The licensee has reported four withdrawn surveillance capsules from Dresden 2. The results from capsules 3 and 4 were published in General Electric Report NEDC-12585. The results from capsules 8 and 5 were published in Southwest Research Report SWRI-06-6901-002 and Battelle Columbus Report BCL-585-10, respectively. The staff has determined that the licensee does not comply with ASTM E 185 and 10 CFR 50, Appendix H because the specimens were not adequately identified to determine which beltline material they correspond to. However, since the licensee based the P/T limits on the limiting beltline material, the staff finds these limits acceptable.

For the limiting beltline material, plate B4065-1, the staff calculated the ART to be 59.5°F at 1/4T (T = reactor vessel beltline thickness) and 43.7°F for 3/4T at 16 EFPY. The staff used a neutron fluence of 1.2E17 n/cm² at 1/4T and 5E16 n/cm² at 3/4T. The licensee calculated an ART of 60°F for the plate B4065-1 in its October 23, 1989 submittal. Substituting the ART of 59.5°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest for Dresden Unit 2 meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% 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°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states, "An exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 20°F, the staff has determined that the proposed P/T limits for Dresden Unit 2 satisfy Section IV.2 of Appendix G.

2.2 Dresden Unit 3

For the Dresden 3 reactor vessel, the staff has determined that the material with the highest ART at 16 EFPY was lower intermediate to lower shell longitudinal weld PQ1300 with 0.3% Cu, 0.33% Ni, and an initial RT_{NDT} of 40°F.

The licensee has reported three withdrawn surveillance capsules from Dresden 3. The results from capsules 6, 16, and 18 were published in Battele Columbus Report BCL-585-14, Westinghouse Report WCAP-10030, and Southwest Research Report SWRI-7484-003/1, respectively. The staff has determined that the licensee does not comply with ASTM E 185 and 10 CFR 50, Appendix H because the specimens were not adequately identified to determine the beltline material they correspond to. However, since the licensee based the P/T limits on the limiting beltline material, the staff finds these limits acceptable.

For the limiting beltline material, weld PQ1300, the staff calculated the ART to be 90.8°F at 1/4T and 71.2°F for 3/4T at 16 EFPY. The staff used a neutron fluence of 1.7E17 n/cm² at 1/4T and 8E16 n/cm² at 3/4T. The ART was also determined by Section 1 of RG 1.99, Revision 2. The licensee calculated an acceptable ART of 90°F for weld PQ1300 in its October 23, 1989 submittal. Substituting the ART of 90.8°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest, for Dresden Unit 3 meet the beltline material requirements in Appendix G of 10 CFR Part 50.

The reference temperature for reactor vessel closure flange materials is discussed under the Dresden Unit 2 evaluation. Based on the flange reference temperature of 40°F, the staff has determined that the proposed P/T limits for Dresden Unit 3 satisfy Section IV.2 of Appendix G.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendment. 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. 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 (55 FR 10530). 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 determined that the proposed Dresden Units 2 and 3 P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 16 EFPY because the limits conform to the requirements of Appendix G of 10 CFR Part 50. The limits satisfy GL 88-11 because the licensee used the method in RG 1.99, Revision 2 to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Dresden Unit 2 and Unit 3 Technical Specifications.

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 amendment 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 5, 1991