October 25, 1989

Docket No. 50-277

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REMartin	Wanda Jones
RClark	JTsao
MO'Brien	JCalvo
JDyer	BGrimes
EWenzinger	

Mr. George A. Hunger, Jr.
Director-Licensing
Philadelphia Electric Company
Correspondence Control Desk
955 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

Dear Mr. Hunger:

SUBJECT: PRESSURE-TEMPERATURE LIMITS FOR THE REACTOR VESSEL (TAC NO. 73600)

RE: PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

The Commission has issued the enclosed Amendment No. ~~153~~ 150 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated May 15, 1989.

This amendment modifies the pressure temperature limits for the reactor vessel.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Bi-Weekly Federal Register Notice.

Sincerely,

/S/

Gene Y. Suh for

Robert E. Martin, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. ~~153~~ 150 to License No. DPR-44
2. Safety Evaluation

OFo1

cc w/enclosures:
See next page

Previously concurred* GYS for

[HUNGER AMEND]

PDI-2/LA*	PDI-2/PM	OGC*	PDI-2/D
MO'Brien	REMartin:tr	PAJ	WButler
09/25/89	10/23/89	10/04/89	10/23/89

WB

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Robert E. Martin, Project Manager
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Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

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1. Amendment No. to License No. DPR-44
2. Safety Evaluation

cc w/enclosures:
See next page

[HUNGER AMEND]

PDI-2/LA	PDI-2/PM	OGC PAJ	PDI-2/D WB
MO'Brien	REMartin:tr		WButler
9/25/89	9/17/89	10/4/89	/ /89

(2) Technical Specifications

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

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 25, 1989

PDI-2/LA MO'Brien 9/25/89 | PDI-2/PM REMartin:tr 9/17/89 | OGC PAJehle 10/4/89 | PDI-2/D WButler 10/23/89 WB

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

October 25, 1989

Docket No. 50-277

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Correspondence Control Desk
955 Chesterbrook Boulevard
Wayne, Pennsylvania 19087-5691

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This amendment modifies the pressure temperature limits for the reactor vessel.

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Sincerely,

Gene Y. Suh
for Robert E. Martin, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. ~~153~~ 150 to License No. DPR-44
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. George A. Hunger, Jr.
Philadelphia Electric Company

Peach Bottom Atomic Power Station,
Units 2 and 3

cc:

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

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 153
License No. DPR-44

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

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated May 15, 1989, 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 or safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

(2) Technical Specifications

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

3. This license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter R. Butler

Walter R. Butler, Director
Project Directorate I-2
Division of Reactor Projects I/II

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 25, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 153

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

iva

143

144

152

152a

164

164a

164b

164c

LIST OF FIGURES

Figure	Title	Page
3.5.1.I	MAPLHGR vs. Planar Average Exposure Unit 2, P8X8R Fuel, Type P8DRB284H, 80 mil & 100 mil channel & 120 mil channels	142h
3.5.1.J	MAPLHGR vs. Planar Average Exposure Unit 2, P8X8R and BP8X8R Fuel, Type P8DRB299 and BP8DRB299, 100 mil channels	142i
3.5.1.K	MAPLHGR vs. Planar Average Exposure Unit 2, P8X8R Fuel (Generic)	142j
3.5.1.L	MAPLHGR vs. Planar Average Exposure Unit 2, BP8X8R Fuel, Type BP8DRB299H	142k
3.5.1.M	MAPLHGR vs. Planar Average Exposure Unit 2, GE8X8EB Fuel, Type BD319A	142l
3.5.1.N	MAPLHGR vs. Planar Average Exposure Unit 2, GE8X8EB, Type BD321A	142m
3.5.1.O	MAPLHGR vs. Planar Average Exposure Unit 2, GE8X8EB, Type LTA310	142n
3.6.1	Minimum Temperature for Pressure Tests such as required by Section XI	164
3.6.2	Minimum Temperature for Mechanical Heatup or Cooldown following Nuclear Shutdown	164a
3.6.3	Minimum Temperature for Core Operation (Criticality)	164b
3.6.4	Deleted	164c
3.6.5	Thermal Power and Core Flow Limits	164d
3.8.1	Site Boundary and Effluent Release Points	216e
6.2-1	Management Organization Chart	244
6.2-2	Organization for Conduct of Plant Operations	245

LIMITING CONDITIONS FOR OPERATION

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal and Pressurization Limitations

1. The average rate of reactor coolant temperature change during normal heatup or cool-down shall not exceed 100° F increase (or decrease) in any one-hour period.

2. The reactor vessel shall not be pressurized for inservice hydrostatic testing above the pressure allowable for a given temperature by Figure 3.6.1.

The reactor vessel shall not be pressurized during heatup by non-nuclear means, during cooldown following nuclear shut down or during low level physics tests above the pressure allowable by Figure 3.6.2, based on the temperatures recorded under 4.6.A.

The reactor vessel shall not be pressurized during operation with a critical core above the pressure allowable by Figure 3.6.3, based on the temperatures recorded under 4.6.A.

SURVEILLANCE REQUIREMENTS

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal and Pressurization Limitations

1. During heatups and cool-downs, the following temperatures shall be permanently logged at least every 15 minutes until the difference between any 2 readings taken over a 45 minute period is less than 5° F.

(a) Bottom head drain
(b) Recirculation loop A and B.

2. Reactor vessel temperature and reactor coolant pressure shall shall be permanently logged at least every 15 minutes whenever the shell temperature is below 220° F and the reactor vessel is not vented.

Test specimens of the reactor vessel base, weld and heat effected zone metal subjected to the highest fluence of greater than 1 Mev neutrons shall be installed in the reac-vessel adjacent to the vessel wall at the core midplane level. The specimens and sample program shall conform to ASTM E 185-66 to the degree discussed in the FSAR.

LIMITING CONDITIONS FOR OPERATION

3.6.A Thermal and Pressurization Limitations (Cont'd)

3. The reactor vessel head bolting studs shall not be under tension unless the temperatures of the closure flanges and adjacent vessel and head materials are greater than 70° F.

4. The pump in an idle recirculation loop shall not be started unless the temperatures of the coolant within the idle and operating recirculation loops are within 50° F of each other.

5. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145° F.

SURVEILLANCE REQUIREMENTS

4.6.A Thermal and Pressurization Limitations (Cont'd)

Selected surveillance specimens shall be removed* and tested to experimentally verify or adjust the calculated values of integrated neutron flux and irradiation embrittlement that are used to determine the RT_{NDT} for Figures 3.6.1, 3.6.2 and 3.6.3, and the figures shall be updated based on the results.

3. When the reactor vessel head bolting studs are tensioned and the reactor is in a Cold Condition, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

4. Prior to and during startup of an idle recirculation loop, the temperature of the reactor coolant in the operating and idle loops shall be permanently logged.

5. Prior to starting a recirculation pump, the reactor coolant temperatures in the dome and in the bottom head drain shall be compared and permanently logged.

*Specimen Removal Schedule

1	Removed at 7.53 EFPY actual
2	15-18 EFPY
3	Standby

3.6.A & 4.6.A BASES (Cont'd)

Operating limits on the reactor pressure and temperature were developed after consideration of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G to 10 CFR Part 50. These considerations involved the reactor vessel beltline and certain areas of discontinuity (e.g. feedwater nozzles and vessel head flange). These operating limits (Figures 3.6.1, 3.6.2 and 3.6.3) assure that a postulated surface flaw can be safely accommodated. Figure 3.6.3 includes an additional 40° F margin required by 10 CFR 50 Appendix G.

The fracture toughness of the vessel low alloy steel in the core region, referred to as beltline, gradually decreases with exposure to neutrons, and it is necessary to account for this change. Regulatory Guide 1.99, Revision 2 provides methods for predicting decreased fracture toughness, in terms of shift in reference temperature of nilductility (RT_{NDT}). Generic methods are used until two surveillance capsules are removed and tested, at which time the surveillance test results may be used to develop plant-specific relationships of RT_{NDT} shift versus fluence.

Three capsules of neutron flux wires and samples of vessel material were installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The first capsule of wires and samples was removed at the end of Cycle 7 and tested in 1988 to experimentally verify the irradiation shift in RT_{NDT} predicted by Regulatory Guide 1.99, Revision 2 methods. The results of the testing are documented in GE Report SASR 88-24 of DRF B13-01445. The results of vessel material testing will not be factored into Figures 3.6.1, 3.6.2 and 3.6.3 until the second capsule is tested. However, the flux wire results were used to predict the design fluence (valid to 32 effective full power years (EFPY)).

The flux wire test results provide the flux at one location in the vessel. The flux distribution can be determined analytically from the core physics data. The ratio of the flux at the peak vessel location to that at the flux wire location, known as the lead factor, was calculated to relate the flux wire test results to the maximum value for the vessel. In developing Figures 3.6.1, 3.6.2 and 3.6.3, the shift predicted by Regulatory Guide 1.99, Revision 2 methods for 32 EFPY of fluence was taken into account. However, in comparing the beltline operating limits (with 32 EFPY shift) to the feedwater nozzle limits, it was determined that the feedwater nozzle was more limiting. Since the feedwater nozzles do not experience significant changes in fracture toughness due to irradiation, the pressure-temperature limits in Figures 3.6.1, 3.6.2 and 3.6.3 apply, without any RT_{NDT} shifting, through 32 EFPY of operation.

As described in paragraph 4.2.5 of the Final Safety Analysis Report, detailed stress analyses have been made on the reactor vessel for both steady state and transient conditions with respect to material fatigue. The results of these transients are compared to allowable stress limits. Requiring the coolant temperature in an idle recirculation loop to be within 50° F of the operating loop temperature before a recirculation pump is started assures that the changes in coolant temperature at the reactor vessel nozzles and bottom head region are acceptable.

3.6.A & 4.6.A BASES (Cont'd)

The design basis event for protection from pressure in excess of vessel design pressure, as required by the ASME Boiler and Pressure Vessel Code, is the closure of all MSIVs resulting in a high flux scram (the slowest indirect scram due to high pressure). The reactor vessel pressure Code limit of 1375 psig is well above the peak pressure produced by this most limiting overpressure event. This is discussed in more detail in Section 4.4.6 of the FSAR and GE safety analyses NEDE-24011-P-A.

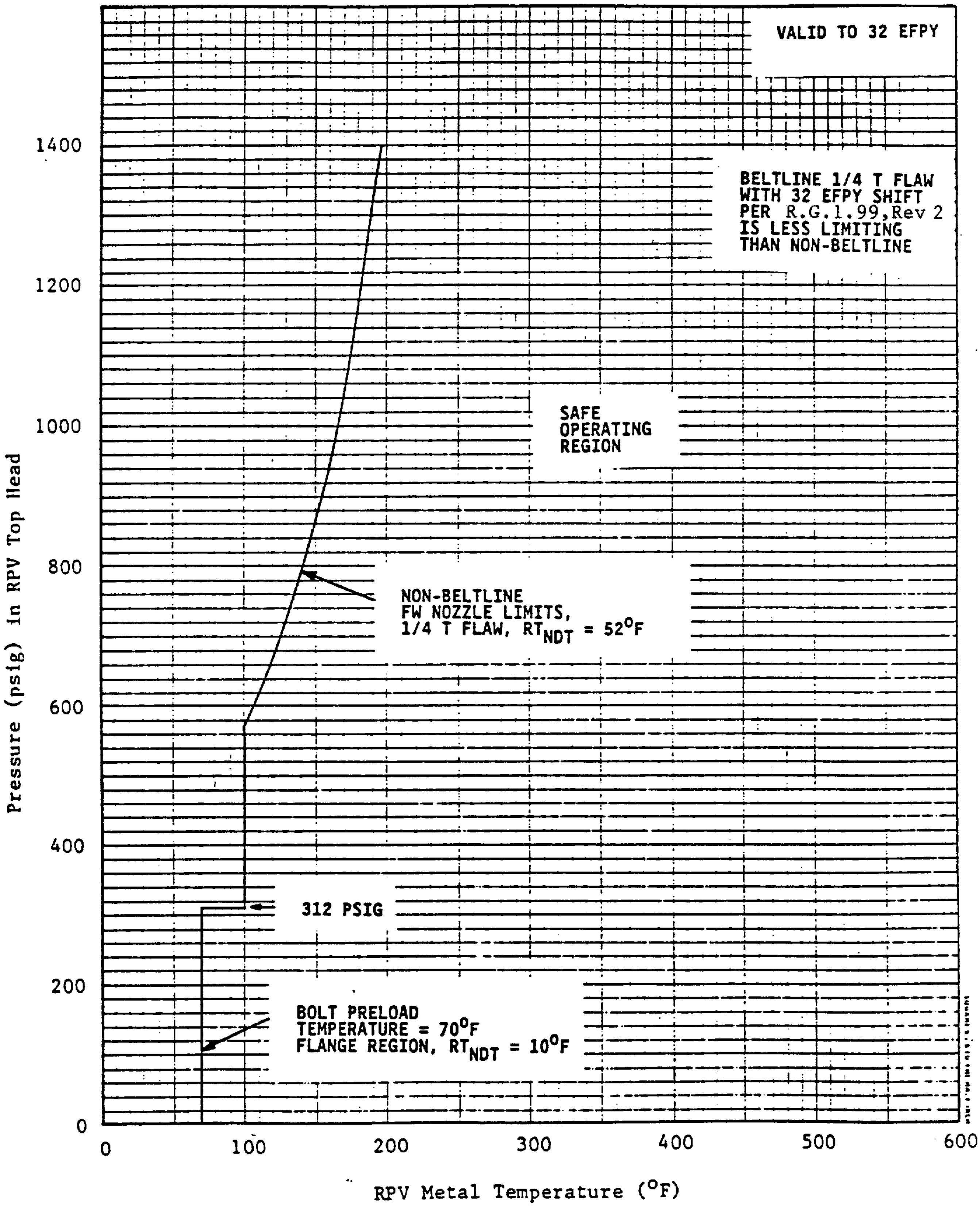

Figure 3.6.1 Peach Bottom 2 Minimum Temperature for Pressure Tests Such as Required by Section XI

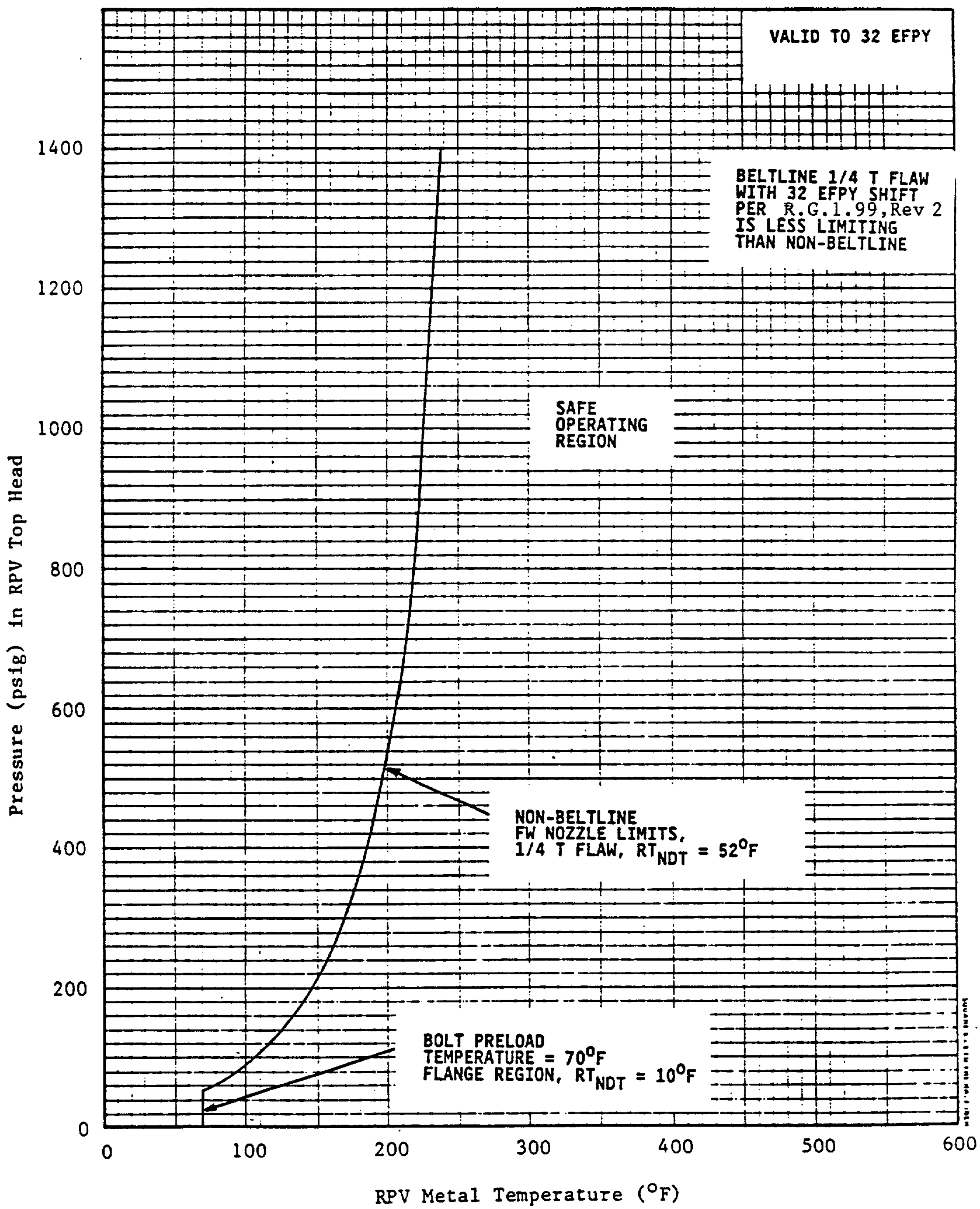

Figure 3.6.2 Peach Bottom 2 Minimum Temperature for Mechanical Heatup or Cooldown Following Nuclear Shutdown

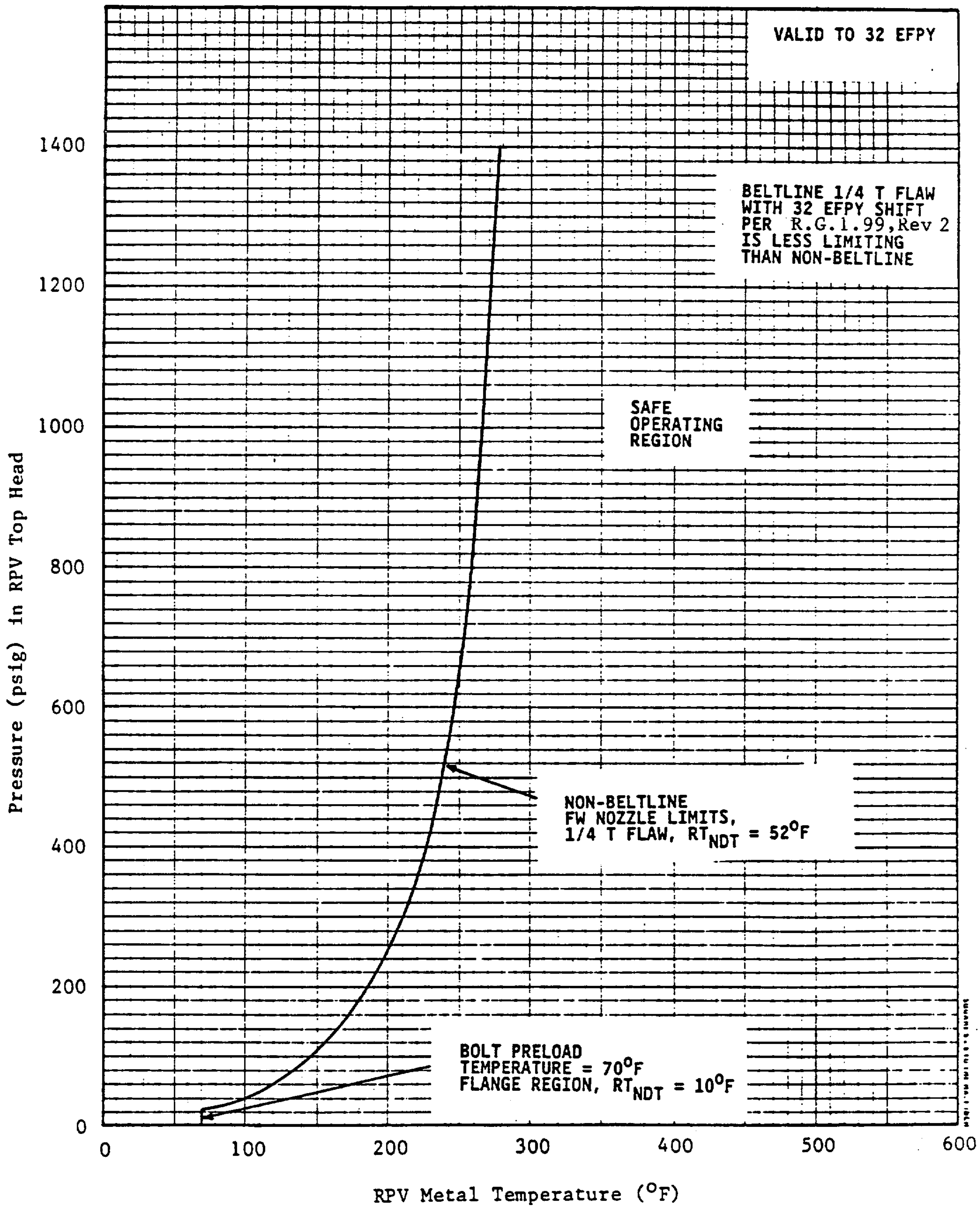

Figure 3.6.3 Peach Bottom 2 Minimum Temperature for Core Operation (Criticality)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION SUPPORTING

AMENDMENT NO. 153 TO FACILITY OPERATING LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

DOCKET NO. 50-277

1.0 INTRODUCTION

By letter dated May 15, 1989, Philadelphia Electric Company requested an amendment to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The proposed amendment modifies the pressure-temperature limits for the reactor vessel.

In response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Philadelphia Electric Company (the licensee) requested permission to revise the pressure/temperature (P/T) limits in the Peach Bottom Atomic Power Station, Unit 2 (hereinafter, Peach Bottom 2) Technical Specifications, Section 3/4.6. The purpose of the revision is to change the effectiveness of the P/T limits for 32 effective full power years (EFPY). The proposed P/T limits were based on Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cool-down, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses 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); 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 of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. 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). Generic Letter 88-11 requested that licensees and permittees 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

The staff evaluated the effect of neutron irradiation embrittlement on each belt-line material in the Peach Bottom 2 reactor vessel. The amount of neutron irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The material with the highest ART at 32 EFPY was the lower-intermediate shell plate C2873-1 with 0.12% copper (Cu) and 0.57% nickel (Ni), and an initial RT of -6°F.

The licensee has removed one surveillance capsule from Peach Bottom 2. The results from that surveillance capsule were published in General Electric Report SASR 88-24, DRF B13-01445, which is an attachment to a letter from J. W. Gallagher to T. E. Murley dated May 13, 1988. The surveillance capsule contained Charpy impact specimens and tensile specimens which were made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, C2873-1, the staff calculated the ART to be 51°F at 1/4T (T = reactor vessel beltline thickness) for 32 EFPY. The staff calculated the ART by the method described in Section 1 of RG 1.99, Rev. 2 because only one surveillance capsule had been withdrawn from the Peach Bottom 2 reactor vessel. The licensee calculated the same 51°F for the ART. Substituting the ART of 51°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest 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. Figure 3.6.2 of the proposed Technical Specifications for Peach Bottom 2 shows that the temperature for heatup or cooldown following nuclear shutdown is approximately 165°F at 300 psig. Figure 3.6.1 of the proposed Technical Specifications for Peach Bottom 2 shows that the minimum temperature for pressure tests required by Section XI of the ASME Code is 100°F at 312 psig. Based on the flange reference temperature of 10°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G for normal operation, hydrostatic pressure and leak tests.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life (EOL) be above 50 ft-lb. The initial USE for the limiting beltline material, the lower-intermediate shell plate metal (C2873-1), was not supplied. However, the calculated USE for the surveillance base metal (C2761-2) at EOL is 84.5 ft-lb, which is higher than the Appendix G EOL USE requirement. The surveillance base metal (C2761-2) was produced by the same manufacturer to the same ASTM specification as the limiting beltline material, and has copper and nickel contents that are very close to those of the limiting beltline material (0.11% Cu and 0.54% Ni for C2761-2 vs 0.12% Cu and 0.57% Ni for C2873-1). Based on this comparison, the staff believes that the EOL USE of the limiting beltline material (C2873-1) meets the Appendix G 50 ft-lb requirement.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Rev. 2, to calculate the ART. Hence, the proposed P/T limits may be incorporated into the Peach Bottom 2 Technical Specifications.

The licensee also proposed certain administrative changes to the Technical Specification pages involved with the changes discussed above. These changes include the deletion of Figure 3.6.4 which provides information on estimated transition temperature shift relative to fluence; rewording T.S. 3.6.A.3 to more accurately describe the vessel materials and appurtenances involved; revision of the "neutron flux specimen" terminology in T.S. 4.6.A.2 to "surveillance specimen"; revisions to T.S. page 144 to reflect removal and testing of a surveillance capsule and deletion of Figure 3.6.4; related changes in the List of Figures; and minor format and typographical changes on page 143 and 144. Related changes to the T.S. Bases are also proposed.

The staff finds that these proposed changes reflect the results of material analyses conducted as part of the reactor coolant pressure boundary material surveillance program. These changes are consistent with the proposed changes to the reactor vessel pressure-temperature limits and are thus acceptable. The staff also finds the proposed addition of Figure 3.6.5 to the List of Figures properly reflects its addition in a previously approved license amendment, and is thus acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

This amendment involves a change to 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 changes to the surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 31116) on July 26, 1989 and consulted with the Commonwealth of Pennsylvania. No public comments were received and the Commonwealth of Pennsylvania did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Tsao

Dated: October 25, 1989

TABLE 1

The NRC Staff Calculated Adjusted Reference Temperature for the Limiting Reactor Beltline Material at Peach Bottom Atomic Power Station, Unit 2.

Limiting Beltline Material:		Lower-intermediate shell plate
Code No.:		C2873-1
Copper Content:		0.12%
Nickel Content:		0.57%
Initial Reference Temperature:		-6°F
Reactor Vessel Beltline Thickness (in.)		6.31
Reactor Vessel Beltline Inside Radius (in.)		125.5
Chemistry Factor (CF) Used in Calculation		82.4
Neutron Fluence n/cm^2 at 32 EFPY:		
	At I.D.	1.0E18
	At 1/4T	0.69E18
	AT 3/4T	0.33E18
Fluence Factor		
	At I.D.	0.417
	At 1/4T	0.347
	At 3/4T	0.231
Margin		28.5°F
ART at 1/4T at 32 EFPY:		51°F (Licensee calculated 51°F)
ART at 3/4T at 32 EFPY		32°F (Licensee did not provide an ART for 3/4T in the GE report, SA 88-24)