AUGUST 4 1978

Docket Nos. ~~50-277 and 50-278~~

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Philadelphia Electric Company
ATTN: Mr. Edward G. Bauer, Jr., Esquire
Vice President and General
Counsel
2301 Market Street
Philadelphia, Pennsylvania 19101

Gentlemen:

The Commission has issued the enclosed Amendments Nos. 45 and 45 to Facility Operating Licenses Nos. DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Units Nos. 2 and 3. The amendments revise the Technical Specifications in response to your request of July 16, 1976, as supplemented April 18 and August 29, 1977, March 27 and April 26, 1978.

These amendments revise the Technical Specifications to modify the reactor coolant system thermal and pressurization limitations to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature.

Copies of the related Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 45 to DPR-44
2. Amendment No. 45 to DPR-56
3. Safety Evaluation
4. Notice

cc w/enclosures:
see next page

OFFICE➤	ORB#3	ORB#3	OELD	ORB#3		
SURNAME➤	SSheppard	DVerrelli:acr	L. Bonner CUTCHIN	TIppolito		
DATE➤	7/ 24 /78	7/ 26 /78	7/ 30 /78	8/ 4 /78		

NRC FORM 318 (9-76) NRCM 0240 ★ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624 cc:

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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. 45
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 July 16, 1976, as supplemented by letters dated April 18 and August 29, 1977, March 27 and April 26, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-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. 45, 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

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 45

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iv	iv
143	143
144	144
151	151
152	152
-	152a
163	163*
164	164
-	164a
-	164b
-	164c

*No changes on this page

LIST OF FIGURES

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 cooldown 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 cooling 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 cooldowns, 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 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 reactor 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	SURVEILLANCE REQUIREMENTS
3.6.A <u>Thermal and Pressurization Limitations</u> (Cont'd)	4.6.A <u>Thermal and Pressurization Limitations</u> (Cont'd)
Figures 3.6.1, 3.6.2, and 3.6.3 will be updated to account for radiation damage prior to 9 effective full power years of operation.	Selected neutron flux specimens shall be removed during the third refueling outage and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the RT_{NDT} for Figure 3.6.4.
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.	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. 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.	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. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	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.
6. Reactor vessel pressure shall not exceed 1020 psig at any time during normal steady state reactor power operation. In the event that this LCO is exceeded, steps shall be immediately initiated to reduce the pressure below 1020 psig. If this cannot be done, shutdown to cold conditions shall be accomplished within 24 hours.	6. The reactor pressure shall be logged once per day.

3.6.A & 4.6.A BASES

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel are discussed in Section 4.2 of the FSAR.

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III, 1965 Edition including Summer 1966 Addenda, stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The manufacturer performed detailed stress analysis as shown in FSAR Appendix K, "Reactor Vessel Report". This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling rate applied continuously over a 100°F to 550°F range. The differential is due to the sluggish temperature response of the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected.

3.6.1 & 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 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 flow can be safely accommodated.

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this shift in operation of the reactor pressure vessel. This decrease is also dependent on the copper and phosphorous content of the steel.

A relationship between neutron fluence and change in operating temperature limit was developed based on the upper bound limit of data generated from General Electric Company research programs and BWR surveillance program. Since data was unavailable to support development of the relationship below 20 degree F, General Electric recommended that a 20 degree F temperature shift be applied to the operating limits subsequent to the first refueling outage. This recommendation is included in Figures 3.6.1, 3.6.2, and 3.6.3.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than 5 years because of the limitation of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

Amendment No. ~~36~~, ~~37~~, 45

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

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6.1, 3.6.2, or 3.6.3 in conjunction with Figure 3.6.4. Note: Figure 3.6.3 includes an additional 40°F margin required by 10 CFR 50 Appendix G.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested to experimentally verify the values used for Figure 3.6.4.

As described in paragraph 4.2.5 of the 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.

The plant safety analyses (Ref: NEDO-21578) states that all MSIV valve closure - Flux scram is the event which satisfies the ASME Boiler and Pressure Code requirements for protection from the consequences of pressure in excess of the vessel design pressure. The reactor vessel pressure code limit of 1375 psig, given in Subsection 4.2 of the FSAR, is well above the peak pressure produced by the above overpressure event. Pressure transients and overpressurization events are analyzed assuming a maximum initial dome pressure of 1020 psig. A safety limit of 1020 psig will assure that the reactor operating pressure will not exceed the initial pressure assumed in the ASME vessel code compliance analysis.

3.6.G & 4.6.G <u>BASES</u> (Cont'd.)

Category E-2

At the present time there is no practical way to volumetrically or visually inspect the bottom head penetrations or drain nozzle weld because of the combination of insulation and control rod and in-core monitor housings configuration. Also, the design of core differential pressure and shell instrumentation nozzles is such that present day volumetric inspection techniques are not practical to utilize. The combination of hydrostatic test and visual checks to be performed do provide reasonable assurance these examination areas are free of gross defects.

Category L-2

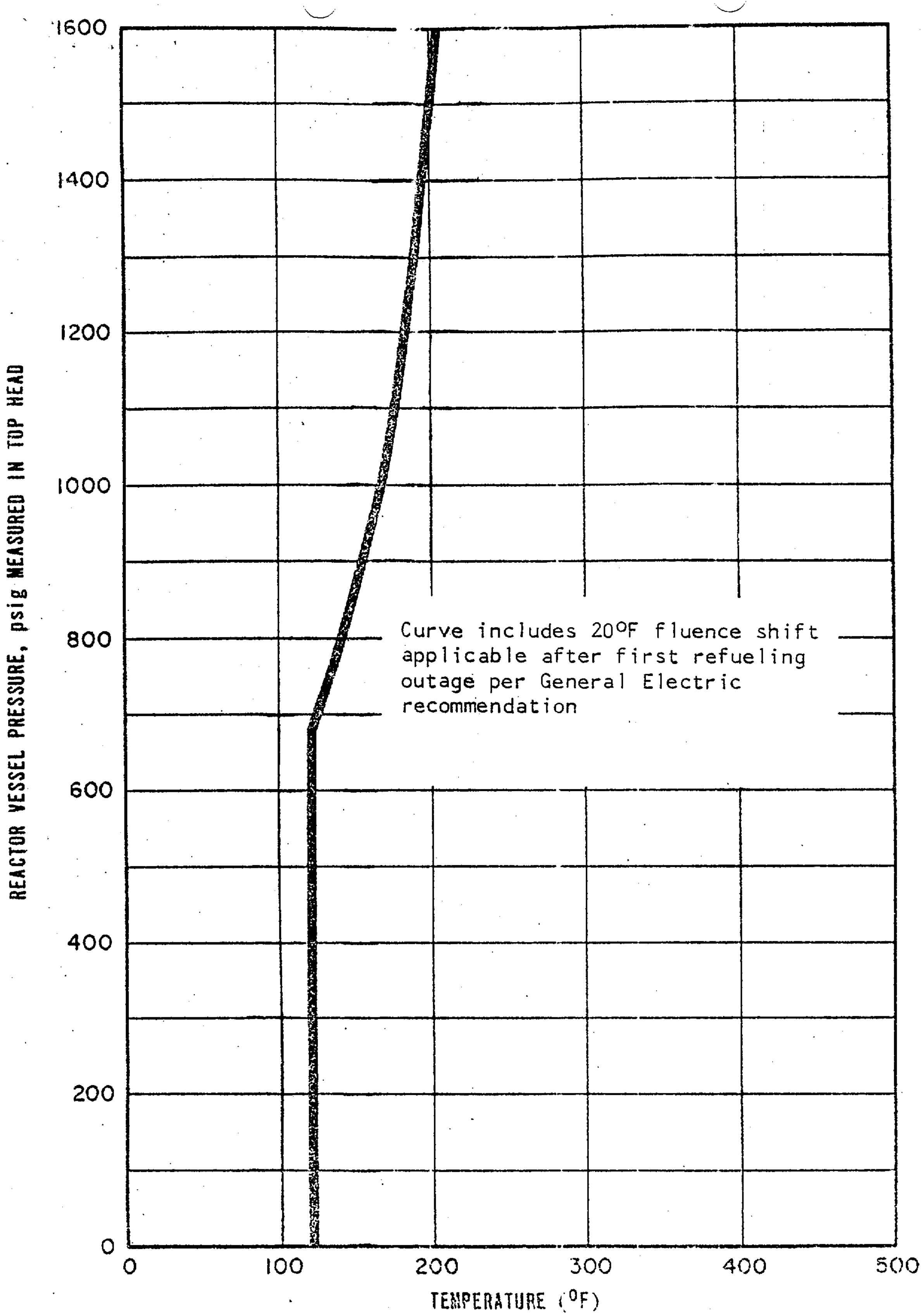
It is the intent that no internal examination be performed on the recirculation pumps unless they are disassembled for maintenance because of the high personnel radiation exposures which would be involved.

Category M-2

There are several valves in the primary pressure boundary which cannot be inspected unless the reactor fuel is removed and reactor water level lowered to the level of the entrance to the jet pump mixer assembly resulting in high personnel radiation exposures from the loss of shielding from the water. Therefore, those valves which would require the reactor water level to be lowered below the low-low water level protection system trip point are excluded from the requirement of visual inspection of internals.

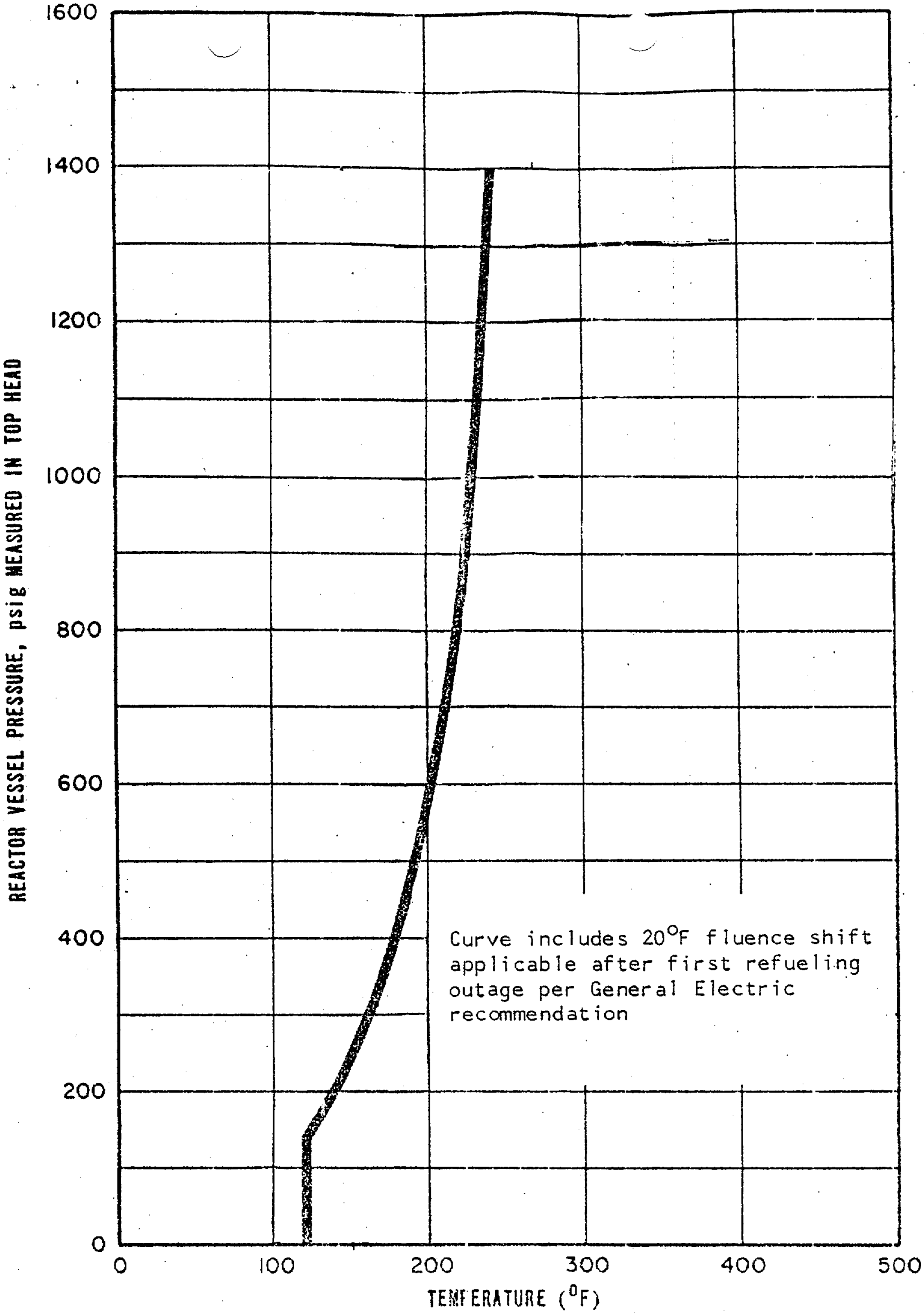
The more frequent inspections delineated for the Category J, Group I pipe welds are to provide additional conservatism in the overall approach of protection against pipe whip which has the potential to breach the containment. A pipe whip protection system is being installed consisting of steel members attached to a reinforcing plate and located such that the postulated pipe weld failure will not breach the containment. Additional inspection of critical welds is also included in the inservice inspection program. The Group I welds listed are those pipe welds of interest.

After five years of operation, a program for in-service inspection of piping and components within the associated auxiliary systems and engineered safety features boundaries, as defined in Section XI of the 1970 ASME Boiler and Pressure Vessel Code, shall be submitted to the AEC.



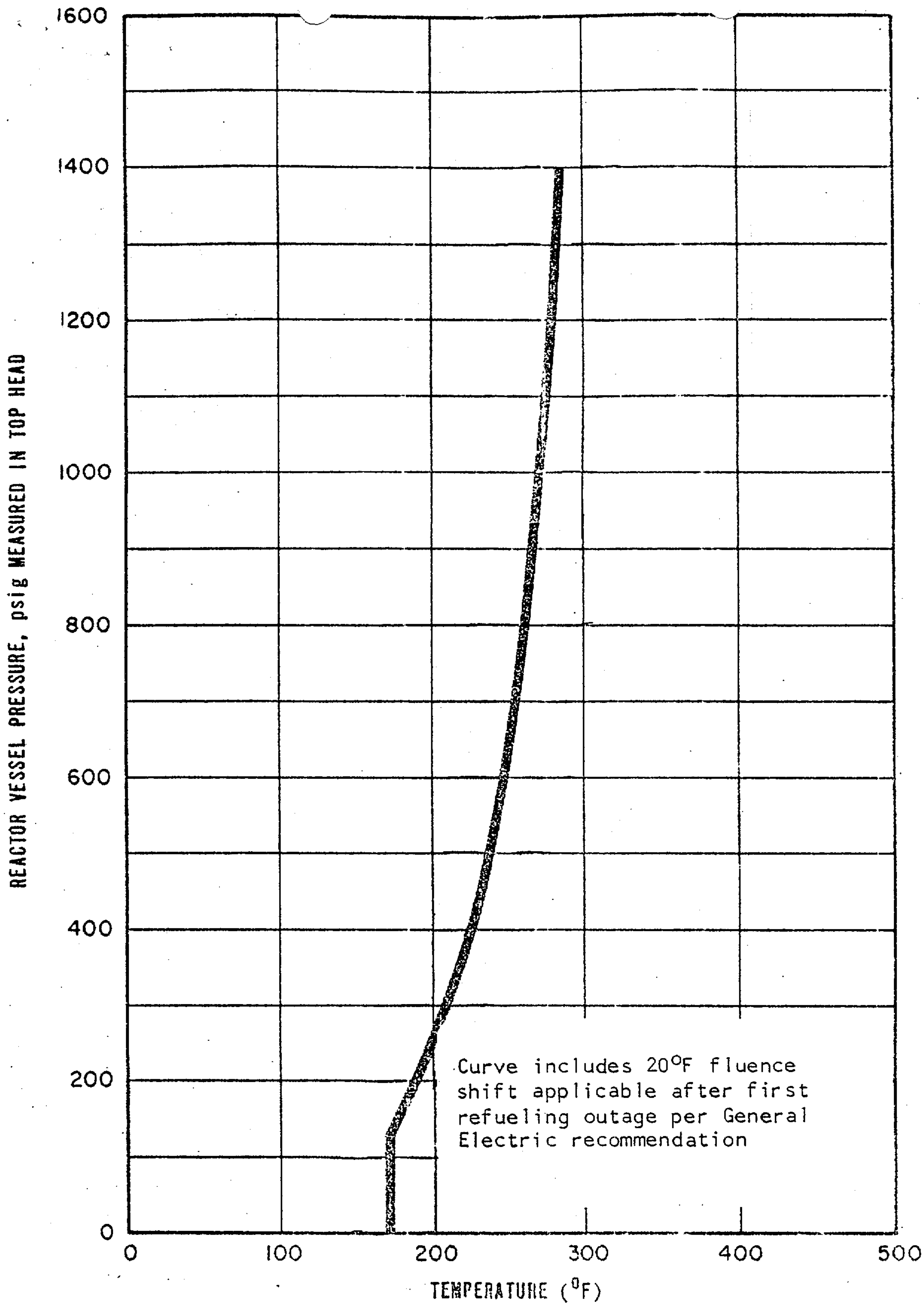
PEACH BOTTOM UNITS 2 & 3 - MINIMUM TEMPERATURE FOR PRESSURE TESTS SUCH AS REQUIRED BY SECTION XI.

Figure 3.6.1



PEACH BOTTOM UNITS 2 & 3 - MINIMUM TEMPERATURE FOR MECHANICAL HEATUP OR COOLDOWN FOLLOWING NUCLEAR SHUTDOWN.

Figure 3.6.2



PEACH BOTTOM UNITS 2 & 3 - MINIMUM TEMPERATURE FOR CORE OPERATION (CRITICALITY). (INCLUDES ADDITIONAL 40°F MARGIN REQUIRED BY 10CFR50 APPENDIX G).

Figure 3.6.3

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. 50-278

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 45
License No. DPR-56

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

   A. The application for amendment by Philadelphia Electric Company, et al, (the licensee) dated July 16, 1976, as supplemented by letters dated April 18 and August 29, 1977, March 27 and April 26, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C(2) of Facility Operating License No. DPR-56 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, 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

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 4, 1978

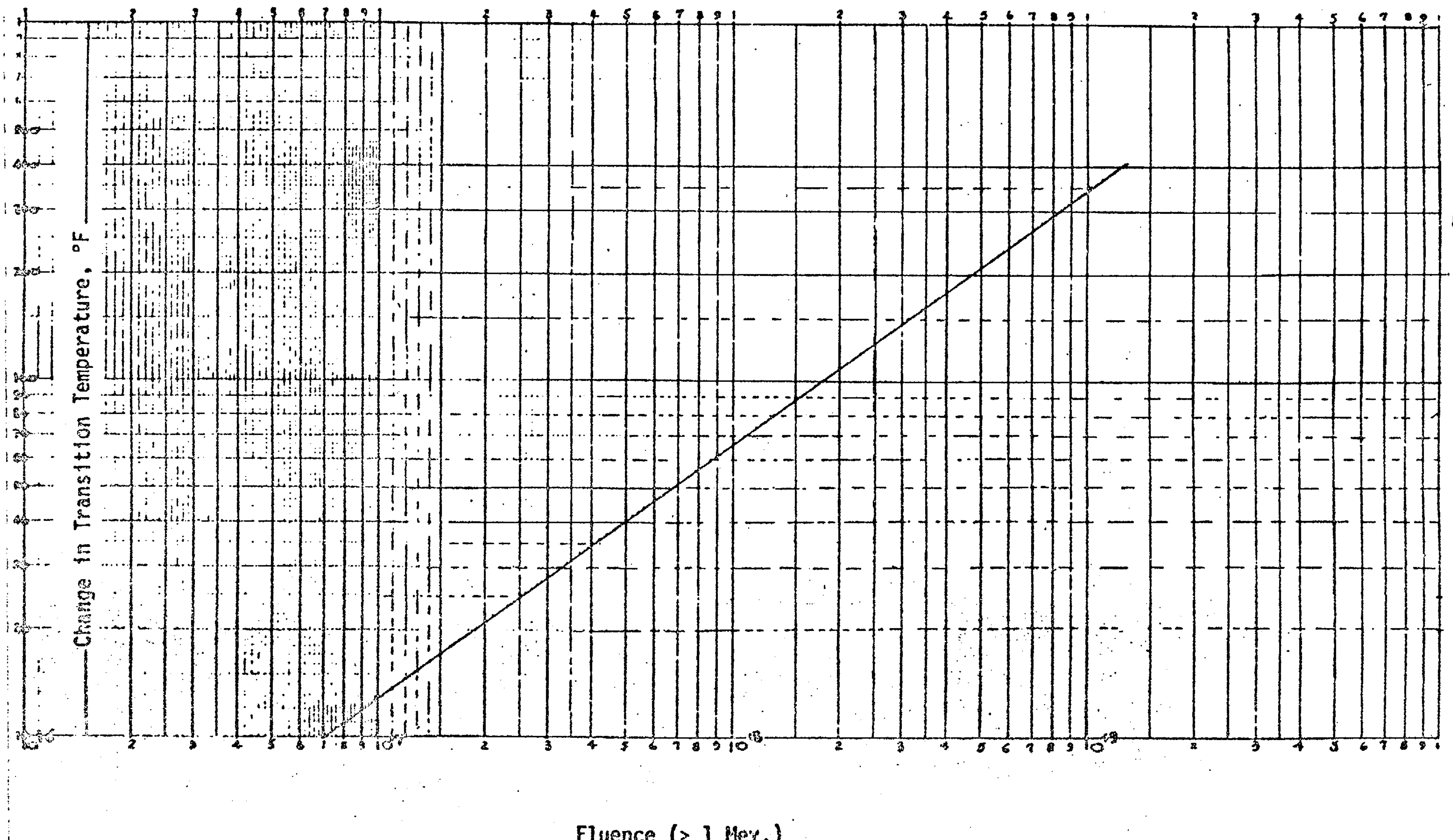


Figure 3.6.4

TRANSITION TEMPERATURE SHIFT VS. FLUENCE
(UPPER BOUND CURVE TO BE USED UNTIL FINALIZED Cu, P CURVES ARE AVAILABLE)

ATTACHMENT TO LICENSE AMENDMENT NO. 45

FACILITY OPERATING LICENSE NO. DPR-56

DOCKET NO. 50-278

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
iv	iv
143	143
144	144
151	151
152	152
-	152a
163	163*
164	164
-	164a
-	164b
-	164c

*No changes on this page

LIST OF FIGURES

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS

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 cooldown 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.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor cooling 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 cooldowns, the following temperatures shall be permamently 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 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 reactor 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	SURVEILLANCE REQUIREMENTS
3.6.A Thermal and Pressurization Limitations (Cont'd) Figures 3.6.1, 3.6.2, and 3.6.3 will be updated to account for radiation damage prior to 9 effective full power years of operation.	4.6.A Thermal and Pressurization Limitations (Cont'd) Selected neutron flux specimens shall be removed during the third refueling outage and tested to experimentally verify or adjust the calculated values of integrated neutron flux that are used to determine the RT_{NDT} for Figure 3.6.4.
3. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel head flange and the head is greater than 100°F.	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. 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.	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. The reactor recirculation pumps shall not be started unless the coolant temperatures between the dome and the bottom head drain are within 145°F.	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.
6. Reactor vessel pressure shall not exceed 1020 psig at any time during normal steady state reactor power operation. In the event that this LCO is exceeded, steps shall be immediately initiated to reduce the pressure below 1020 psig. If this cannot be done, shutdown to cold conditions shall be accomplished within 24 hours.	6. The reactor pressure shall be logged once per day.

3.6.A & 4.6.A BASES

Thermal and Pressurization Limitations

The thermal limitations for the reactor vessel are discussed in Section 4.2 of the FSAR.

The allowable rate of heatup and cooldown for the reactor vessel contained fluid is 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 546°F. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III, 1965 Edition including Summer 1966 Addenda, stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The manufacturer performed detailed stress analysis as shown in FSAR Appendix K, "Reactor Vessel Report". This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

The permissible flange to adjacent shell temperature differential of 145°F is the maximum calculated for 100°F hour heating and cooling rate applied continuously over a 100°F to 550°F range. The differential is due to the sluggish temperature response of the flange metal and its value decreases for any lower heating rate or the same rate applied over a narrower range.

The coolant in the bottom of the vessel is at a lower temperature than that in the upper regions of the vessel when there is no recirculation flow. This colder water is forced up when recirculation pumps are started. This will not result in stresses which exceed ASME Boiler and Pressure Vessel Code, Section III limits when the temperature differential is not greater than 145°F.

The reactor coolant system 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, restrictions have been placed on the operating conditions to which it can be subjected.

3.6.1 & 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 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 flow can be safely accommodated.

The fracture toughness of all ferritic steels gradually and uniformly decreases with exposure to fast neutrons above a threshold value, and it is prudent and conservative to account for this shift in operation of the reactor pressure vessel. This decrease is also dependent on the copper and phosphorous content of the steel.

A relationship between neutron fluence and change in operating temperature limit was developed based on the upper bound limit of data generated from General Electric Company research programs and BWR surveillance program. Since data was unavailable to support development of the relationship below 20 degree F, General Electric recommended that a 20 degree F temperature shift be applied to the operating limits subsequent to the first refueling outage. This recommendation is included in Figures 3.6.1, 3.6.2, and 3.6.3.

The neutron fluence at any point in the pressure vessel wall can be computed from core physics data. The neutron fluence can also be measured experimentally on the ID of the vessel wall. At present valid experimental measurements can be made only over time periods of less than 5 years because of the limitation of the dosimeter materials. This causes no problem because of the exact relationship between thermal power produced and the number of neutrons produced from a given core geometry. A single experimental measurement in a time period of one year can be used to predict the fluence for the life of the plant in terms of thermal power output if no great changes in core geometry are made.

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

The vessel pressurization temperatures at any time period can be determined from the thermal power output of the plant and its relation to the neutron fluence and from Figure 3.6.1, 3.6.2, or 3.6.3 in conjunction with Figure 3.6.4. Note: Figure 3.6.3 includes an additional 40°F margin required by 10 CFR 50 Appendix G.

Neutron flux wires and samples of vessel material are installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The wires and samples will be removed and tested to experimentally verify the values used for Figure 3.6.4.

As described in paragraph 4.2.5 of the 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.

The plant safety analyses (Ref: NEDO-24039-1) states that all MISV valve closure - Flux scram is the event which satisfies the ASME Boiler and Pressure Code requirements for protection from the consequences of pressure in excess of the vessel design pressure. The reactor vessel pressure code limit of 1375 psig, given in Subsection 4.2 of the FSAR, is well above the peak pressure produced by the above overpressure event. Pressure transients and overpressurization events are analyzed assuming a maximum initial dome pressure of 1020 psig. A safety limit of 1020 psig will assure that the reactor operating pressure will not exceed the initial pressure assumed in the ASME vessel code compliance analysis.

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

Category E-2

At the present time there is no practical way to volumetrically or visually inspect the bottom head penetrations or drain nozzle weld because of the combination of insulation and control rod and in-core monitor housings configuration. Also, the design of core differential pressure and shell instrumentation nozzles is such that present day volumetric inspection techniques are not practical to utilize. The combination of hydrostatic test and visual checks to be performed do provide reasonable assurance these examination areas are free of gross defects.

Category L-2

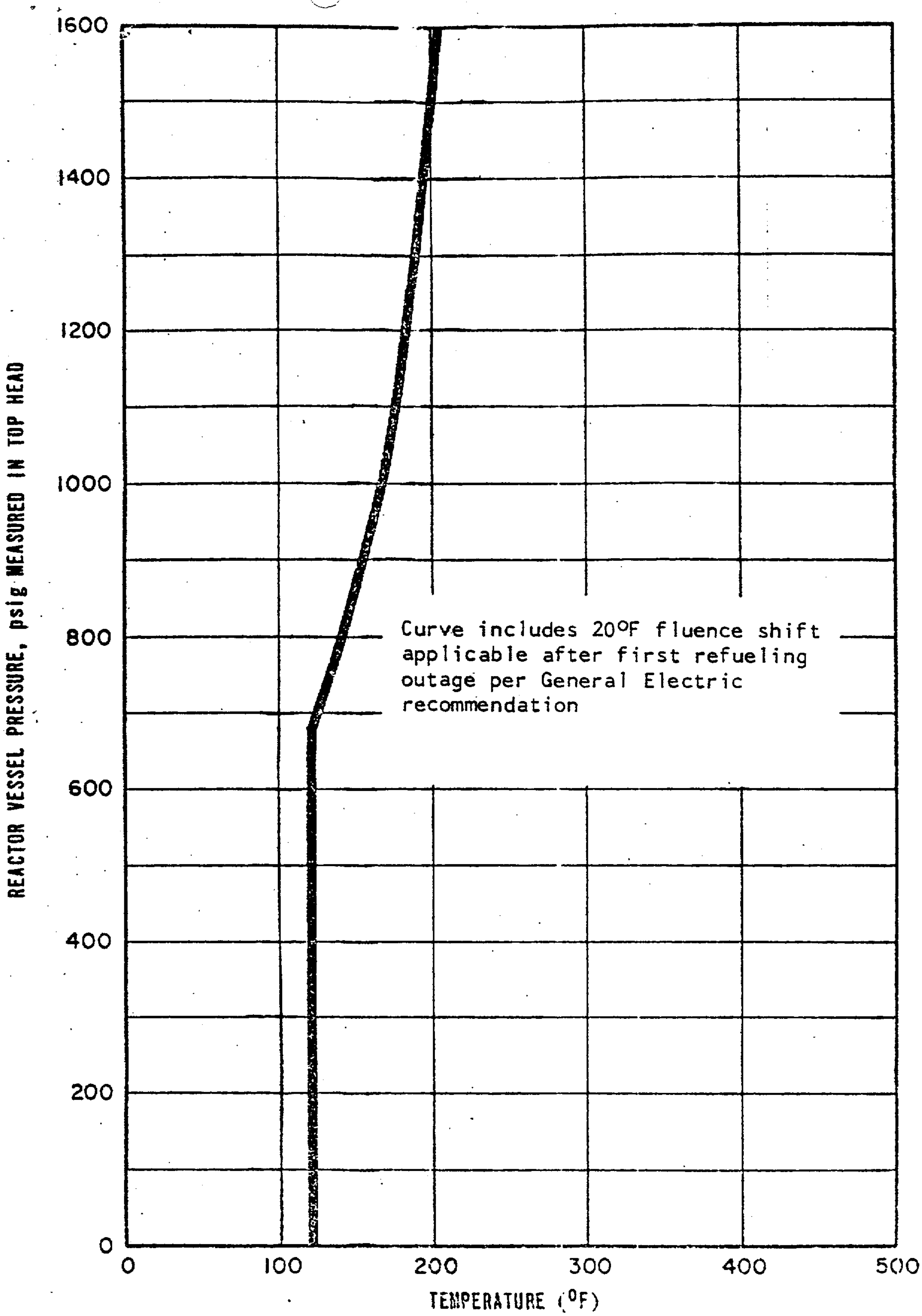
It is the intent that no internal examination be performed on the recirculation pumps unless they are disassembled for maintenance because of the high personnel radiation exposures which would be involved.

Category M-2

There are several valves in the primary pressure boundary which cannot be inspected unless the reactor fuel is removed and reactor water level lowered to the level of the entrance to the jet pump mixer assembly resulting in high personnel radiation exposures from the loss of shielding from the water. Therefore, those valves which would require the reactor water level to be lowered below the low-low water level protection system trip point are excluded from the requirement of visual inspection of internals.

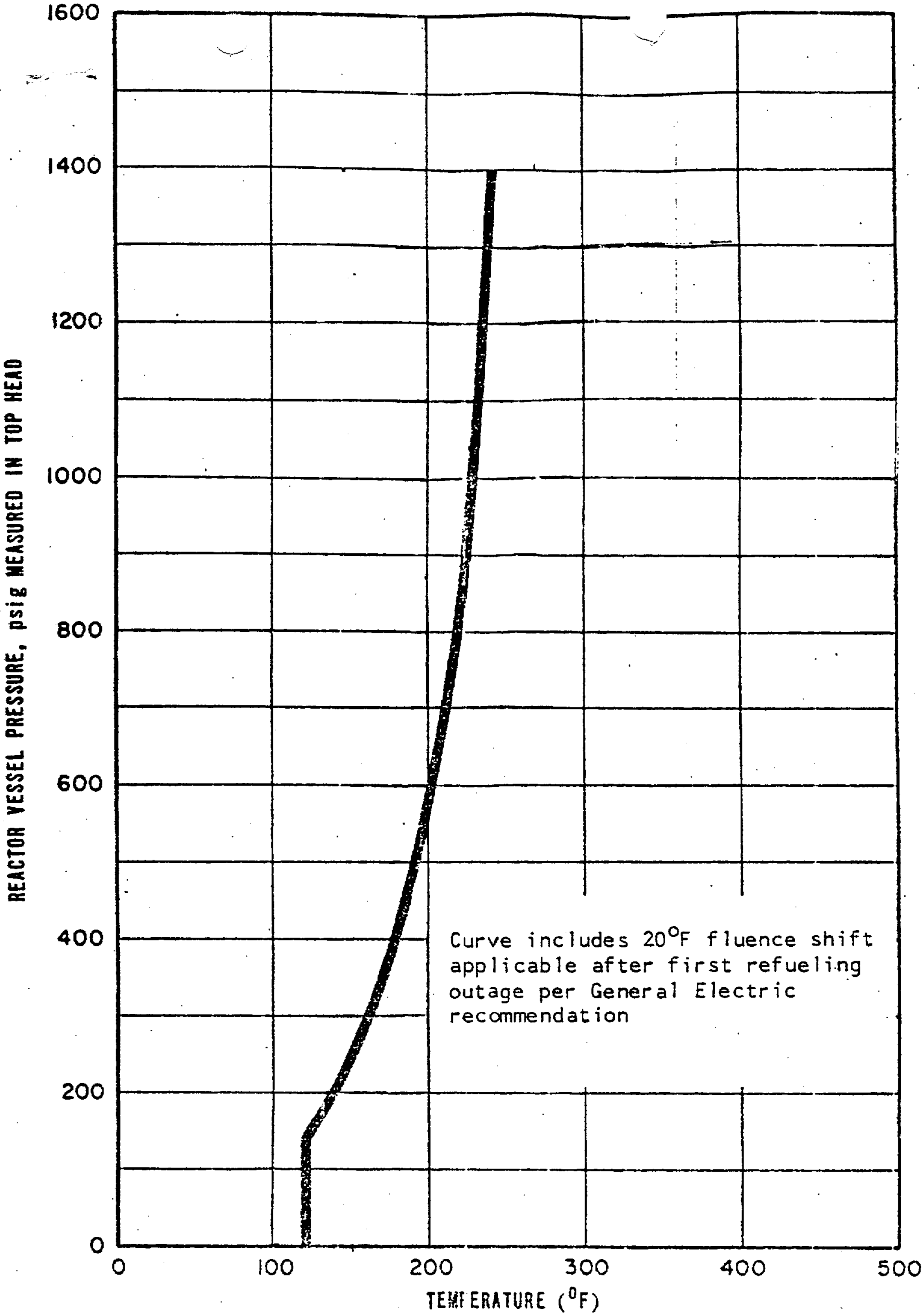
The more frequent inspections delineated for the Category J, Group I pipe welds are to provide additional conservatism in the overall approach of protection against pipe whip which has the potential to breach the containment. A pipe whip protection system is being installed consisting of steel members attached to a reinforcing plate and located such that the postulated pipe weld failure will not breach the containment, Additional inspection of critical welds is also included in the inservice inspection program. The Group I welds listed are those pipe welds of interest.

After five years of operation, a program for in-service inspection of piping and components within the associated auxiliary systems and engineered safety features boundaries, as defined in Section XI of the 1970 ASME Boiler and Pressure Vessel Code, shall be submitted to the AEC.



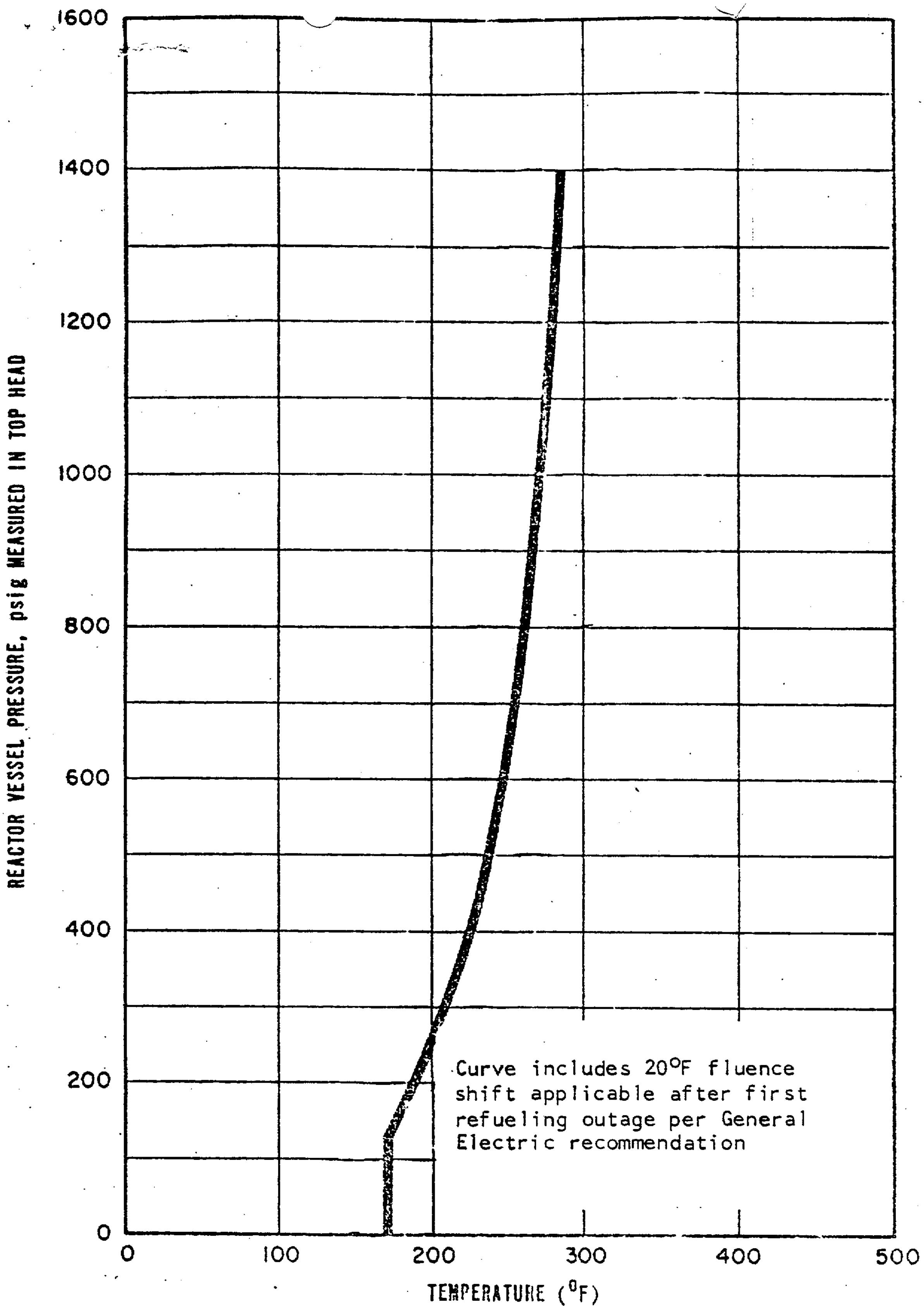
PEACH BOTTOM UNITS 2 & 3 - MINIMUM TEMPERATURE FOR PRESSURE TESTS SUCH AS REQUIRED BY SECTION XI.

Figure 3.6.1



PEACH BOTTOM - MINIMUM TEMPERATURE FOR MECHANICAL HEATUP OR
UNITS 2 & 3 COOLDOWN FOLLOWING NUCLEAR SHUTDOWN.

Figure 3.6.2



PEACH BOTTOM UNITS 2 & 3 - MINIMUM TEMPERATURE FOR CORE OPERATION (CRITICALITY). (INCLUDES ADDITIONAL 40°F MARGIN REQUIRED BY 10CFR50 APPENDIX G).

Figure 3.6.3

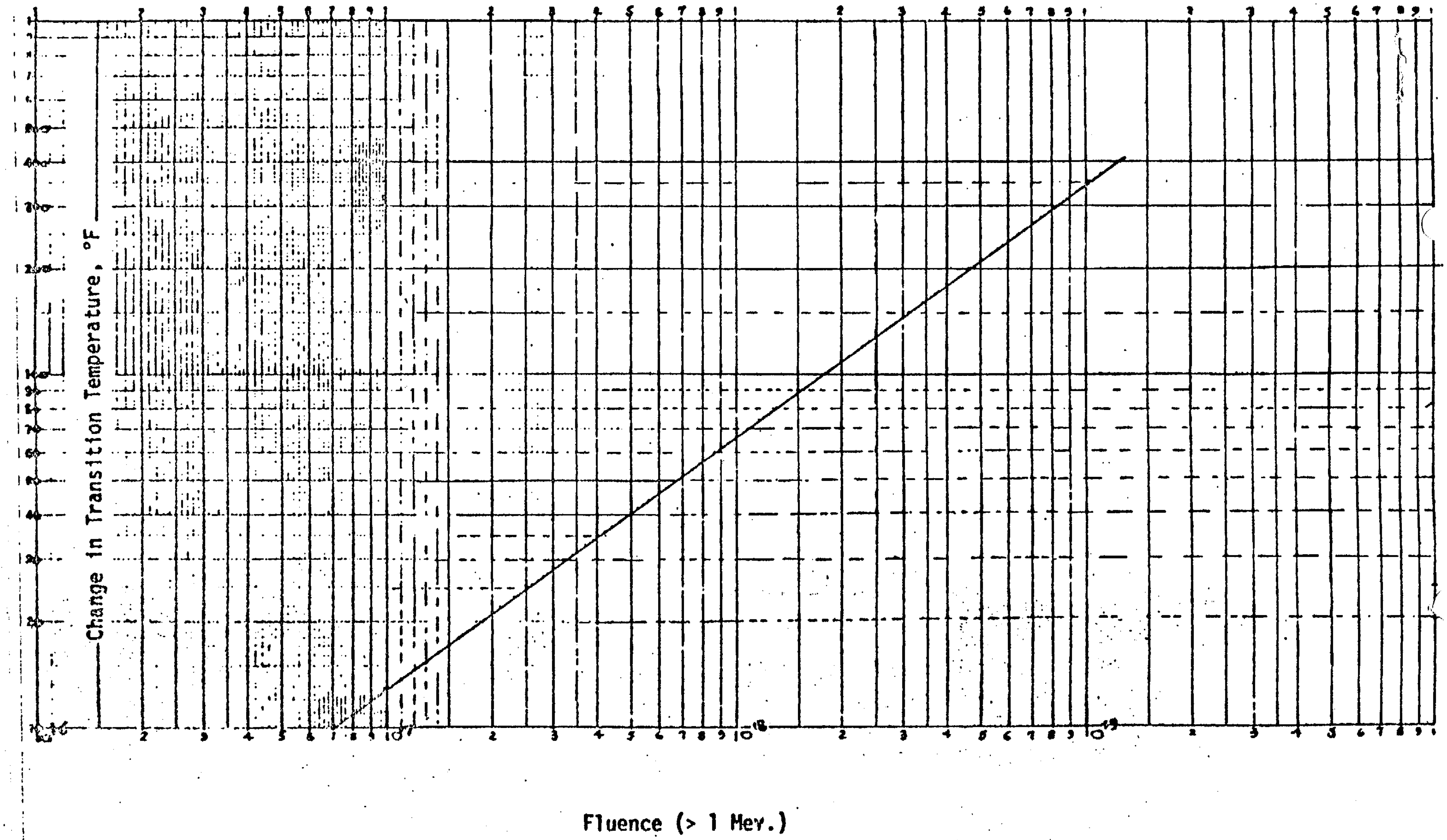


Figure 3.6.4

TRANSITION TEMPERATURE SHIFT VS. FLUENCE
(UPPER BOUND CURVE TO BE USED UNTIL FINALIZED Cu, P CURVES ARE AVAILABLE)

Amendment No. 45

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 45 AND 45 TO FACILITY LICENSE NOS. DPR-44 AND DPR-56

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION UNITS NOS. 2 AND 3

DOCKETS NOS. 50-277 AND 50-278

I. INTRODUCTION

By letter dated July 16, 1976 as supplemented by letters dated April 18 and August 29, 1977, March 27 and April 26, 1978, Philadelphia Electric Company (PECo) proposed changes to the Technical Specifications appended to Operating Licenses DPR-44 and DPR-56 for Peach Bottom Atomic Power Station Units Nos. 2 and 3. The changes would modify the reactor coolant system thermal and pressurization limitations to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature (RT_{NDT})[1]. The PECo submittal was based on their determination that certain changes were necessary to bring the reactor coolant system pressure-temperature limits into conformity with the requirements of Appendix G to 10 CFR 50.

DISCUSSION

Title 10 CFR Part 50, Appendix G "Fracture Toughness Requirements", requires that pressure-temperature limits be established for reactor coolant system heatup and cooldown operations, inservice leak and hydrostatic tests, and reactor core operation. These limits are required to ensure that the stresses in the reactor vessel remain within acceptable limits. They are intended to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences.

The specific pressure-temperature limits which are initially established depend upon the metallurgical properties of the reactor vessel material

[1] RT_{NDT} is the temperature associated with the transition from ductile to brittle fracture mode of failure.

and the design service conditions. However, the metallurigical properties vary over the lifetime of the reactor vessel because of the effects of neutron irradiation. One principal effect of the neutron irradiation is that it causes the reactor vessel nil ductility temperature (RT_{NDT}) to increase or shift with time. The practical results of the RT_{NDT} shift is that, for any given value of reactor pressure, the reactor vessel metal temperature must be maintained at higher values during the heatup and cooldown process. By periodically revising the pressure-temperature limits to account for neutron irradiation induced increases in RT_{NDT}, the stresses in the reactor vessel are maintained within acceptable limits.

EVALUATION

The PECo submittal dated July 16, 1976 included a radiation damage estimate curve and pressure-temperature limits for hydrostatic testing, mechanical heatup or cooldown and minimum temperature for Core Operation (Criticality). During our review of this submittal we determined that the radiation damage estimate curve, i.e. the effect of neutron fluence on RT_{NDT}, did not appear to be as conservative as that presented in Regulatory Guide 1.99. In response to our request for additional information, PECo provided by letters dated August 29, 1977 and April 26, 1978 revised radiation damage curves. The curve provided by the August 29 submittal is the upper bound limit of data generated from General Electric Company research programs and BWR surveillance programs. The revised damage estimate curve in the April 26 submittal is based on new data contained in General Electric Licensing Topical Report "Radiation Effects in Boiling Water Reactor Pressure Vessel Steel" (NEDO-21708). Since we have not completed our review of NEDO-21708, our evaluation of the pressure-temperature limits is based on the upper bound curve. This curve is comparable to that contained in Regulatory Guide 1.99 which is based on a large amount of data from both test reactors and material surveillance programs. Thus, the estimate of radiation damage is conservative.

Branch Technical Position, MTEB 5-2 "Fracture Toughness Requirements" requires that calculations be performed in regions of high stress unless the assumed RT_{NDT} of the beltline is at least 50°F above the RT_{NDT} of all higher stressed regions. To satisfy this requirement, PECo obtained stress intensities in regions of discontinuities by adjusting the results of a generic analysis made on the BWR/6 251 reactor vessel to account for differences between the design and materials of the Peach Bottom vessels and those of the reference plant. We have reviewed the licensee's submittal and determined that this is an acceptable procedure for calculating pressure-temperature operating limits since both type vessels were designed to the same rule, i.e., section III of the ASME Code.

The operating limits for hydrostatic testing, mechanical heatup or cool-down, and minimum temperature for core operation (criticality) were calculated by PECo and are presented in Figures 4, 5 and 6 of the August 29, submittal. These limits were revised in the April 26 submittal to include a 20°F shift in RT_{NDT} to account for irradiation damage. Based on the use of the previously discussed upper bound damage estimate curve, a neutron fluence of 2.5 x 10" n/cm^2 per one Effective Full Power Year (EFPY) at the one-quarter thickness (1/4T) location and the 20°F shift in RT_{NDT} included in the PECo submittal of April 26, we conclude that these limits are acceptable for operation through approximately 10 EFPY. Accordingly, the staff added a Specification to the thermalization and pressurization limits to require that the figures for hydrostatic testing, mechanical heatup or cooldown and minimum temperature for core operation (criticality) will be updated to account for radiation damage prior to 9 EFPY. This additional requirement was discussed with the licensee and he agreed.

We conclude that the pressure-temperature operating limits as amended by the staff are acceptable through 9 EFPY. For this operating period the proposed pressure-temperature operating limits are in accordance with Appendix G, 10 CFR Part 50. Compliance with Appendix G in establishing safe operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: August 4, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-277 AND 50-278

PHILADELPHIA ELECTRIC COMPANY, ET AL

PEACH BOTTOM UNITS NOS. 2 AND 3

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 45 and 45 to Facility Operating License Nos. DPR-44 and DPR-56, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company and Atlantic City Electric Company, which revised the Technical Specifications for operation of the Peach Bottom Atomic Power Station Units Nos. 2 and 3, located in York County, Pennsylvania. The amendments are effective as of the date of issuance.

These amendments revise the Technical Specifications to modify the reactor coolant system thermal and pressurization limitations to account for neutron irradiation induced increases in reactor vessel metal nil ductility temperature.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) application for amendment dated July 16, 1978, as supplemented April 18 and August 29, 1977, March 27 and April 26, 1978, (2) Amendments Nos. 45 and 45 to License Nos. DPR-44 and DPR-56, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 4th day of August 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

TAIppolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors