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

April 20, 1990

Docket No. 50-352

Mr. George A. Hunger, Jr.
Director-Licensing, MC 5-2A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: PRESSURE-TEMPERATURE OPERATING LIMIT CURVES, REQUEST NO. 89-08 (TAC NO. 75585) AND GENERIC LETTER 88-11 (TAC NO. 71509)

RE: LIMERICK GENERATING STATION, UNIT 1

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 29, 1989.

This amendment changes the TSs to specify the revised time period for which the reactor pressure vessel pressure-temperature operating limit curves are valid. The application for this amendment was a commitment you made in your response of November 23, 1988 to Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation." Our assessment of this proposed TS change also included an evaluation of your response to GL 88-11 with the assistance of our contractor, EG&G Idaho. As discussed in the enclosed safety evaluation, we have concluded that your response to GL 88-11 is acceptable for Limerick, Unit 1.

A partial evaluation of your response to GL 88-11 for Limerick, Unit 2 was included in SSER-9 to NUREG-0991. Our final evaluation will be the subject of separate correspondence.

With issuance of the enclosed TSs and safety evaluation, we have determined that your response to GL 88-11 and MPA A-023 is fully implemented as of this date for Limerick, Unit 1.

DF01
1/1

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

Sincerely,

Original signed by
Richard J. Clark

Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 36 to
   License No. NPF-39
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION w/enclosures:

Docket File	ACRS (10)	JCalvo
NRC PDR	GPA/PA	JDyer
Local PDR	OGC	RBlough
PDI-2 Rdg File	Rita Jaques,ARM/LFMB	LDoerflein
SVarga	CSchulten,OTSB	
BBoger	GHill (4)	
WButler	EJordan	
RClark	DHagan	
GSuh	Wanda Jones	
MO'Brien	JTsao	

[71509]

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:		WButler
4/11/90	04/02/90	4/17/90	4/20/90

DF01
1/1

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Sincerely,

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See next page

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

Limerick Generating Station
Units 1 & 2

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

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
License No. NPF-39

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

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated December 29, 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 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. NPF-39 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 36 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1990

PDI-2/LA	PDI-2/PM	OGC	PDI-2/D
MO'Brien	RClark:mj		WButler
4/ /90	04/02/90	4/11/90	4/20/90

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

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 20, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached page. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 4-17	3/4 4-17*
3/4 4-18	3/4 4-18
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6*
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8*

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS IS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for $\bar{E}$ Determination	At least once per 6 months*	1
4. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1**, 2**, 3**, 4**
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.	1, 2
5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135, and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**Until the specific activity of the primary coolant system is restored to within its limits.

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 curve A, B, or C as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor flux wire specimens shall be removed at the first refueling outage and examined to determine reactor pressure vessel fluence as a function of time and power level and used to modify Figure B 3/4 4.6-1. The results of these fluence determinations shall be used to adjust the curves of Figure 3.4.6.1-1, as required.

4.4.6.1.5 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

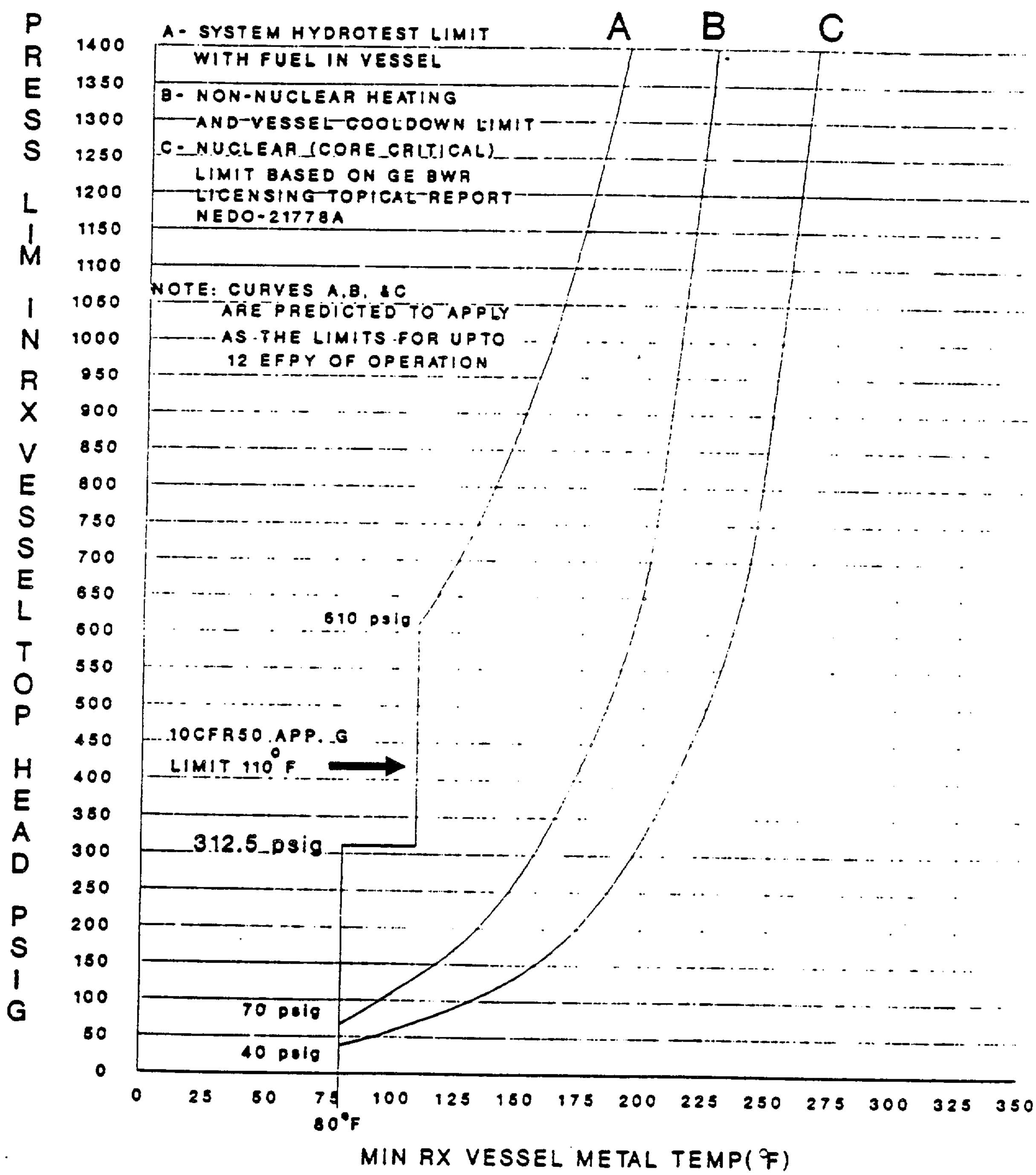
a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

   2. $\leq$ 90°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

MIN RPV METAL TEMP VS. RX VESSEL PRESS
FIGURE 3.4.6.1-1



PRESSURE/TEMPERATURE LIMITS (Continued)

The operating limit curves of Figure 3.4.6.1-1 are derived from the fracture toughness requirements of 10 CFR 50 Appendix G and ASME Code Section III, Appendix G. The curves are based on the RT_{NDT} and stress intensity factor information for the reactor vessel components. Fracture toughness limits and the basis for compliance are more fully discussed in FSAR Chapter 5, Paragraph 5.3.1.5, "Fracture Toughness."

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel content and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, Figure 3.4.6.1-1, curves A, B and C, are based on the non-beltline, discontinuity areas of the RPV which do not receive significant neutron fluence and the RT_{NDT}s will, therefore, not shift. These limit curves are predicted to be bounding for all areas of the RPV until 12 EFPY when the beltline material's RT_{NDT} will shift due to neutron fluence and the beltline curves will intersect the non-beltline discontinuity curves. The non-limiting beltline curves are not shown on Figure 3.4.6.1-1, but are included on FSAR Figure 5.3-4.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR Part 50, Appendix H, irradiated reactor vessel flux wire and Charpy specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the flux wires, Charpy specimens and vessel inside radius are essentially identical, the irradiated Charpy specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the flux wire and Charpy specimen data and recommendations of Regulatory Guide 1.99, Revision 2. This would include showing the beltline (versus non-beltline discontinuity) limits when they become bounding.

The pressure-temperature limit lines shown in Figures 3.4.6.1-1, curves C, and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Winter 1972.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

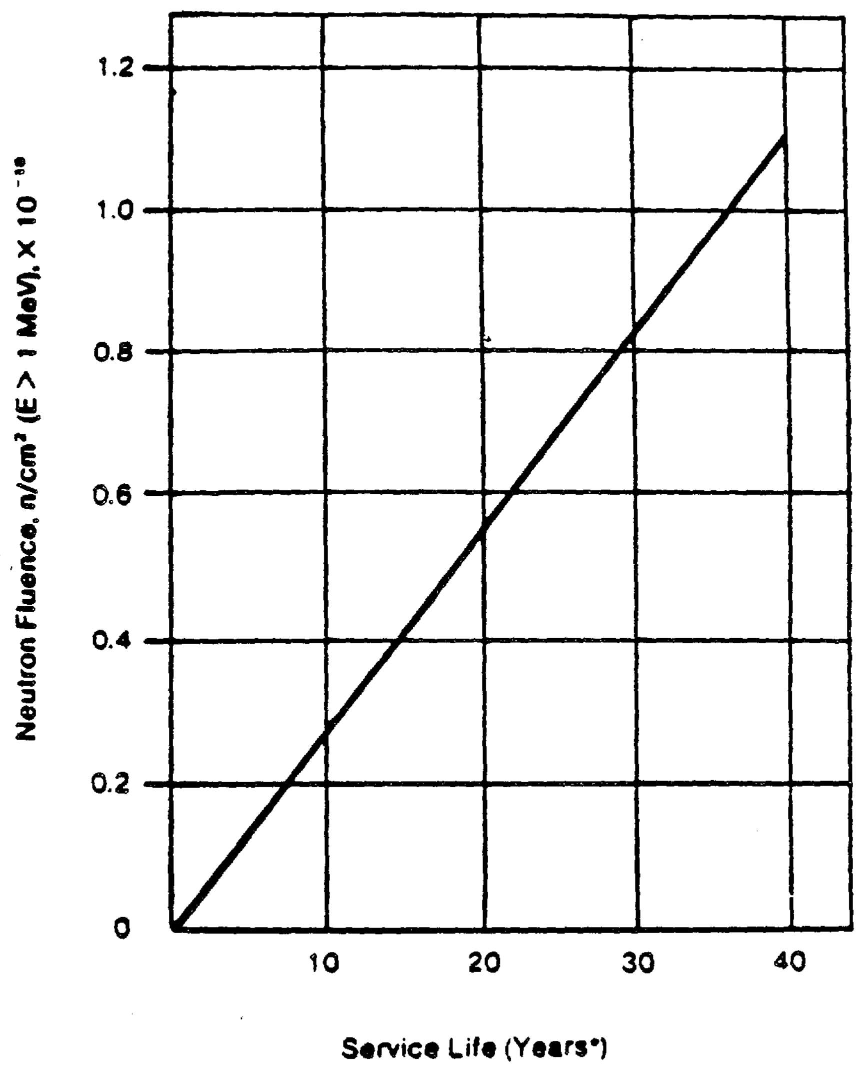
REACTOR VESSEL TOUGHNESS

BELTLINE COMPONENT	WELD SEAM I.D. OR MAT'L TYPE	HEAT/SLAB OR HEAT/LOT	CU (%)	Ni (%)	STARTING RT_{NDT} (°F)	ΔRT_{NDT} MAX. * (°F)	MIN. UPPER SHELF (LFT-LBS)	MAX. RT_{NDT} (°F)
Plate	SA-533 Gr B CL.1	C 7677-1	.11	.5	+20	+66	NA	+86
Weld	SFA 5.5, (E 80T8-G)	662A746/ H013A27A	.03	.88	-20	+25	NA	+5

NOTE:* These values are given only for the benefit of calculating the end-of-life (EOL) RT_{NDT}

NON-BELTLINE COMPONENT	MT'L TYPE OR WELD SEAM I.D.	HEAT/SLAB OR HEAT/LOT	HIGHEST STARTING RT_{NDT} (°F)
Shell Ring	SA 533, Gr. B, CL. 1	C7711-1	+20
Bottom Head Dome	"	C7973-1	+12
Bottom Head Torus	"	C7973-1	+12
Top Head Dome	"	A6834-1	+10
Top Head Torus	"	B1993-1	+10
Top Head Flange	SA-508, CL. 2	123B195-289	0
Vessel Flange	"	2V1924-302	-30
Feedwater Nozzle	"	Q2Q22W-412	-10
Weld	Non-Beltline	All	0
LPCI Nozzle **	SA-508, CL. 2	Q2Q25W	-6
Closure Studs	SA-540, Gr. B-24	All	Meet requirements of 45 ft-lbs and 25 mils Lat. Exp. at +10°F

Note:** The design of the LPCI nozzles results in their experiencing an EOL fluence in excess of 10^{17} N/Cm^2 which predicts an EOL RT_{NDT} of +39°F.



FAST NEUTRON FLUENCE (E>1 MeV) AT ¼ T AS A FUNCTION
OF SERVICE LIFE*

BASES FIGURE B 3/4.4.6-1

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* At 90% of RATED THERMAL POWER and 90% availability.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-39

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNIT 1

DOCKET NO. 50-352

1.0 INTRODUCTION

By letter dated December 29, 1989, Philadelphia Electric Company (the licensee) requested an amendment to Facility Operating License No. NPF-39 for the Limerick Generating Station, Unit 1. The proposed amendment would change the Technical Specifications (TSs) to specify the revised time period for which the reactor pressure vessel pressure-temperature operating limit curves are valid.

2.0 BACKGROUND

Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operation" was issued July 12, 1988. Licensees were advised that they should use the methods described in Revision 2 to Regulatory Guide 1.99, which became effective May 1988, to predict the effect of neutron radiation on reactor vessel materials as required by paragraph V.A. of 10 CFR Part 50 Appendix G, unless they can justify the use of different methods. Licensees were advised that "the use of the Revision 2 methodology may result in a modification of the pressure-temperature limits contained in Technical Specifications in order to continue to satisfy the requirements of Sec. V of 10 CFR Part 50, Appendix G." Licensees were requested to submit the results of their technical analysis (of RG 1.99, Rev. 2) and a proposed schedule for whatever actions they proposed to take.

Philadelphia Electric Company submitted a response to GL 88-11 by letter dated November 23, 1988. The current pressure-temperature curves in the Limerick, Unit 1 TSs are identified as being valid for 32 equivalent full power years (EFPY). The licensee's reanalysis using RG 1.99, Rev. 2 indicated that the time period for which the present curves were valid needed to be reduced. The licensee stated that an amendment application would be submitted to revise the operating limits curves. The application which is the subject of this safety evaluation is the result of the licensee's reanalysis in accordance with GL 88-11.

Specifically, the amendment would change the note on the Pressure-Temperature Operating Limit (PTOL) curve, figure 3.4.6.1-1, to reflect new, more conservative adjusted reference temperatures calculated in accordance with RG 1.99, Rev. 2. (These limits are subsequently referred to as P/T limits.) The present note states that "curves A, B, and C are predicted to apply as the limits for 40 years (32 equivalent full power years, EFPY) of operation." The revised note for the same curves, (which are not being changed), states that "curves A, B, and C are predicted to apply as the limits for up to 12 EFPY of operation." Three curves in the same figure are also being deleted. These three curves are explicitly identified in the present TSs as not being limiting curves and have been "shown for information only." Sections 3.4.6.1 and 4.4.6.1 are being revised to remove the references to the A', B', and C' curves. The Bases section 3/4 4.6 and Bases Table B 3/4 4.6-1 are being revised to reflect the above changes. The proposed pressure-temperature (P/T) limits were developed based on RG 1.99, Rev. 2. The proposed revisions provide up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

3. DISCUSSION

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, Rev. 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, Rev. 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.

4.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Limerick 1 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 12 EFPY was plate C7677-1 with 0.11% copper (Cu), 0.50% nickel (Ni), and an initial RT_{ndt} of 20°F.

The licensee has not removed any surveillance capsules from Limerick 1. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate C7677-1, the staff calculated the ART to be 60.4°F at 1/4T (T = reactor vessel beltline thickness) and 46.2°F for 3/4T at 12 EFPY. The staff used a neutron fluence of 4.5E17 n/cm² at 1/4T and 2.1E17 n/cm² at 3/4T. The ART was determined using Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Limerick 1 reactor vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 59.8°F at 12 EFPY at 1/4T for the same limiting plate material. The staff judges that a difference of 0.6°F between the licensee's ART of 59.8°F and the staff's ART of 60.4°F is acceptable. Substituting the ART of 60.4°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. 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 reload." Based on the flange reference temperature of 0°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

5.0 SUMMARY

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 12 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 Limerick 1 Technical Specifications.

6.0 ENVIRONMENTAL CONSIDERATION

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 a change 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.

7.0 CONCLUSION

The Commission made a proposed determination that the amendment involves no significant hazards consideration which was published in the Federal Register (55 FR 4274) on February 7, 1990 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 this amendment will not be inimical to the common defense and the security nor to the health and safety of the public.

Dated: April 20, 1990

Principal Contributors:

Richard Clark
John Tsao

REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2, Pressure-Temperature Limits

3. Limerick Generating Station Unit 1, FSAR

4. November 23, 1988, Letter from J. W. Gallaugher (PECo) to T. Murley (USNRC), Subject: Response to Generic Letter 88-11

5. December 29, 1989, Letter from G. A. Hunger, Jr. (PECo) to USNRC, Document Control Desk, Subject: Application for Amendment to License NPF-39, Consisting of Technical Specification Change Request 89-08 to Section 3/4.4.6