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10 CFR 50.90

NUCLEAR ENGINEERING & SERVICES DEPARTMENT

March 28, 1991

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Peach Bottom Atomic Power Station, Units 2 and 3
Technical Specifications Change Request

Dear Sir:

Philadelphia Electric Company hereby submits Technical Specifications Change Request No. 90-17, in accordance with 10 CFR 50.90, requesting an amendment to the Peach Bottom Units 2 and 3 Technical Specifications (Appendix A) of Facility Operating License Nos. DPR-44 and DPR-56. Information supporting this Change Request is contained in Attachment 1 to this letter, and the proposed replacement Technical Specifications pages are contained in Attachment 2.

The Company requests Technical Specifications changes to modify the pressure-temperature limits for the reactor vessels.

If you have any questions regarding this matter, please feel free to contact us.

Very truly yours,

G. J. Beck, Manager
Licensing Section
Nuclear Engineering & Services

Enclosure: Affidavit
Attachments 1 and 2

A001
1/1

cc: T. T. Martin, Administrator, Region I, USNRC
J. J. Lyash, USNRC Senior Resident Inspector, PB

COMMONWEALTH OF PENNSYLVANIA:

: ss.

COUNTY OF CHESTER :

D. M. Smith, being first duly sworn, deposes and says:

That he is Senior Vice President-Nuclear of Philadelphia Electric Company, the Applicant herein; that he has read the enclosed request for amendment of Peach Bottom Units 2 and 3 Facility Operating License Nos. DPR-44 and DPR-56 (Change Request 90-17) and knows the contents thereof; and that the statements and matters set forth therein are true and correct to the best of his knowledge, information and belief.

D M Smith

Senior Vice President-Nuclear

Subscribed and sworn to before me this 28th day of March 1991.

Catherine A. Mendez

Notary Public

NOTARIAL SEAL
CATHERINE A. MENDEZ, Notary Public
Tredyffrin Twp., Chester County
My Commission Expires Sept. 4, 1993

ATTACHMENT 1

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

TECHNICAL SPECIFICATIONS CHANGE REQUEST
NUMBER 90-17

"Revision of Pressure-Temperature Limits for the Reactor Vessels"

11 Pages

Philadelphia Electric Company, Licensee under Facility Operating Licenses DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station (PBAPS) Units 2 and 3, requests that the Technical Specifications contained in Appendix A of the Operating License be amended by revising the following pages of Unit 3: iv, iva, 143, 144, 152, 152a, 164, 164a, 164b and 164c; and, the following pages of Unit 2: 143, 144 and 164a. Revisions are indicated with a vertical bar in the page margins.

This amendment reflects the results of material analyses conducted as part of the reactor coolant pressure boundary material surveillance program pursuant to 10 CFR 50, Appendix G and Appendix H. The requested changes will alter the reactor vessel pressure-temperature operating limits for Unit 3. Additionally, a curve for the bottom head limits is being added to the PBAPS Units 2 and 3 Technical Specifications.

Also included in this amendment is the proposed removal of the withdrawal schedule for the reactor vessel material specimens in accordance with guidance provided in Generic Letter 91-01 ("Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications"). Miscellaneous administrative changes are also proposed.

Introductory Technical Discussion

Revision to the Pressure-Temperature Curves

A surveillance capsule was removed from the Peach Bottom Atomic Power Station Unit 3 reactor vessel at the end of Fuel Cycle 7 (removed in June 1989). The capsule contained flux wires for neutron fluence measurement, and Charpy and tensile test specimens for material property evaluation. A combination of flux wire testing and computer analysis was used to establish the vessel peak flux location and magnitude. Charpy V-Notch impact testing and uniaxial tensile testing were performed to establish the material properties of the irradiated vessel beltline (core region).

The irradiation effects were projected in accordance with the guidance in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", to conditions for 32 effective full power years (EFPY) of operation. The 32 EFPY conditions are predicted to be less severe than the limits that would require vessel thermal annealing. Pressure-temperature operating limits valid to 32 EFPY were

developed in accordance with the July 1983 requirements of 10 CFR 50, Appendix G. The irradiation shift in nil-ductility transition temperature was accounted for in accordance with the guidance in Regulatory Guide 1.99, Revision 2. As recommended by the Regulatory Guide, the material property test results were not used to develop the operating limits; they will be used after the second set of specimens are tested. The results of the analyses show that the non-beltline limits are more severe than the beltline limits, even including predicted 32 EFPY shift.

The surveillance capsule withdrawal and test results discussed above were the subject of a technical report submitted to the NRC on June 27, 1990 (GE Nuclear Energy, SASR 90-50). Based on the results of the test specimen analyses, Licensee requests several changes to the Technical Specifications which are discussed separately in the "Description of Proposed Changes".

The proposed changes to the thermal and pressurization limitations displayed in the new Unit 3 Figures 3.6.1, 3.6.2, and 3.6.3 were developed considering the most limiting conditions of the discontinuity regions and the irradiated beltline region in order to bound all operating conditions. The limiting regions of the vessel affecting the curve's shapes are the feedwater nozzles, bottom head and closure flange regions. Since the temperature of the bottom head can lag behind the rest of the vessel under certain non-nuclear heatup/cooldown situations, a second curve (B_{BH}) for the bottom head limits, has been added to Figure 3.6.2 of Units 2 and 3. Curve B on Figure 3.6.2 will be used for the feedwater nozzle and vessel flange limits whereas Curve B_{BH} will be used for the bottom head Control Rod Drive (CRD) penetration limits. The predicted irradiation shifts for the beltline materials are low enough that the beltline is not predicted to be limiting through 32 EFPY of operation.

The Unit 2 bottom head limits contained in curve B_{BH} for Figure 3.6.2 referenced above are based on the Unit 2 surveillance capsule withdrawal and test results discussed in a technical report submitted to the NRC on May 13, 1988 (GE Nuclear Energy, SASR 88-24).

Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens (Generic Letter 91-01)

Included in this amendment request is a proposed administrative change to remove the schedule for withdrawal of reactor vessel material specimens from the Technical Specifications.

The PBAPS Units 2 and 3 surveillance requirements specify the withdrawal schedule for the reactor vessel material specimens. Recently, the NRC approved a request to remove this schedule from the Technical Specifications of the Joseph M. Farley Nuclear Plant. The NRC has determined that the placement of this schedule in the Technical Specifications duplicates the controls on changes to this schedule that have been established by 10 CFR 50, Appendix H. Therefore, the staff concluded that, because this duplication is unnecessary, the removal of this Technical Specification schedule as a line-item improvement is consistent with the Commission Policy Statement on Technical Specification Improvements.

This change is being done in accordance with the guidance provided in Generic Letter 91-01 ("Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications").

This administrative change is discussed below as a Category 2 change.

DESCRIPTION OF PROPOSED CHANGES

The Category 1 changes are technical in nature and involve the reactor vessel pressure-temperature limits. The Category 2 changes are administrative.

Category 1 Changes:

A. Licensee proposes to replace the Unit 3 pressure-temperature limit curves in Figures 3.6.1, 3.6.2 and 3.6.3 (pages 164, 164a, and 164b, respectively) with new curves which are based on the Unit 3 neutron flux surveillance specimen test results. The new curves for Unit 3 represent less restrictive operating limits than the current curves, but will still provide sufficient margin to prevent brittle fracture of reactor coolant pressure boundary material. These curves are valid to 32 EFPY.

   Also included on Figure 3.6.2 (page 164a) of Units 2 and 3 is a new second curve, B_{BH}, for the bottom head limits. Since the temperature of the bottom head can lag behind the rest of the vessel under certain non-nuclear heatup/cooldown situations, this second curve is being added to the Technical Specifications.

B. Licensee proposes to delete Unit 3 Figure 3.6.4 (page 164c) which provides information on estimating the shift in nil-ductility transition temperature (RT_{NDT}) relative to

fluence. This figure was for information only and did not establish any Technical Specification requirement.

C. Licensee proposes to reduce the Unit 3 minimum temperature of the vessel head flange and vessel head at which the head bolting studs may be under tension (as stated in Specification 3.6.A.3, page 144). Currently, the temperature must be "greater than 100°F"; we propose that the temperature must be "greater than 70°F". This change is recommended by the reactor vessel supplier and is consistent with Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5), NUREG-0123, Rev. 3. The current 100°F limit is overly restrictive. The ASME Code to which the vessel was built* only required a 70°F (RT_{NDT} +60°) limit, and the current Code permits an even lower limit.

Category 2 Changes:

A. Licensee proposes to delete Figure 3.6.4 ("Transition Temperature Shift vs. Fluence") from the Unit 3 "List of Figures" (page iv). Additionally, Licensee proposes to delete the content of Unit 3 page iva which is duplicated at the bottom of page iv.

B. Licensee proposes to correct a sentence in Unit 3 Specification 3.6.A.2 by adding a comma. The proposed Specification states: "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..."

C. Licensee proposes to correct the "Applicability" portion of Unit 3 Specification 4.6 (page 143). Currently, this Specification makes reference to the "reactor cooling system". The proposed Specification would change this wording to the "reactor coolant system".

* ASME Boiler & Pressure Vessel Code, Section III, its interpretations, and applicable requirements including 1965 Winter Addendum for Class A vessels as defined therein.

D. Licensee proposes to reword Units 2 and 3 Specification 4.6.A.2 (page 143) to more accurately describe the test specimens installed in the reactor vessel. Currently, this Specification states: "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..." The proposed Specification states: "Test specimens of the reactor vessel base, weld and heat affected zone metal were installed..." The existing words could be interpreted to mean that the installed specimens represent vessel locations that receive only the highest fluence and neutron energies greater than 1 Mev. In fact, the installed specimens represent vessel locations that receive a variety of neturon energies and fluence levels.

E. Licensee proposes to delete the last paragraph of Unit 3 Specification 3.6.A.2 (page 144) which states that Figures 3.6.1, 3.6.2, and 3.6.3 will be updated prior to nine (9) effective full power years of operation. This Technical Specification amendment is updating Figures 3.6.1, 3.6.2 and 3.6.3 prior to nine effective full power years of operation. Therefore, this Specification is no longer needed.

F. Licensee proposes to reword Unit 3 Specification 3.6.A.3 (page 144) to more accurately describe the vessel materials and appurtenances involved. Currently, this Specification states "... the temperature of the vessel head flange and the head is..." The proposed Specification states "...the temperatures of the closure flanges and adjacent vessel and head materials are..."

G. Licensee proposes to replace the reference in the Unit 3 Specification 4.6.A.2 (page 144) from "neutron flux specimens" to "surveillance specimens", which is the more common term. The capsules contain specimens for material property evaluation in addition to the "neutron flux" wires.

H. Licensee proposes to remove from the Units 2 and 3 Specification 4.6.A.2 (page 144) the specimen withdrawal schedule (identified as a footnote to Specification 4.6.A.2) and insert the proposed words "in accordance with 10 CFR 50, Appendix H". This change is in accordance with the guidance provided in Generic Letter 91-01 ("Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens for Technical Specifications").

I. Licensee proposes to add to Unit 3 Specification 4.6.A.2 (page 144) the words "and irradiation embrittlement" which clarify the purpose of testing the surveillance specimens.

J. Licensee proposes to remove from Unit 3 Specification 4.6.A.2 (page 144) the words "... for Figure 3.6.4," and replace them with "... for Figures 3.6.1, 3.6.2, and 3.6.3, and the figures shall be updated based on the results." This change clarifies that the testing of the surveillance specimens is used to determine and update Figures 3.6.1, 3.6.2 and 3.6.3.

K. Licensee also proposes that the Bases of Unit 3 Specifications 3.6.A and 4.6.A (pages 152, 152a) be revised to provide current information about the surveillance program.

SAFETY ASSESSMENT

Category 1 Changes:

Section 4.2 of the Final Safety Analysis Report (FSAR) states that the safety design bases of the reactor vessel and appurtenances are to "withstand adverse combinations of loadings and forces resulting from operation under abnormal and accident conditions" and to "minimize the possibility of brittle fracture failure of the nuclear system process barrier." The revised thermal and pressurization limits will not compromise these safety objectives because they were developed in accordance with NRC Regulations and the latest NRC guidance, which do support these safety objectives.

The original analysis of the reactor vessel material specimens in conjunction with the surveillance specimen program ensures that the reactor pressure boundary will behave in a non-brittle manner during plant testing, startup, and operation. The revised pressure-temperature limit curves were conservatively generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, as supplemented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. The proposed minimum allowable temperature at which the head bolting studs may be under tension is also in accordance with 10 CFR 50, Appendix G as supplemented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. The RT_{NDT} used to evaluate the new pressure-temperature limits for the beltline material was based on Regulatory Guide 1.99, Revision 2, which is the latest guidance on RT_{NDT} determinations.

The proposed changes to the thermal and pressurization limitations displayed in new Figures 3.6.1, 3.6.2, and 3.6.3 for Unit 3 were developed considering the most limiting conditions of the discontinuity regions and the irradiated beltline region in

order to bound all operating conditions. The limiting regions of the vessel affecting the curve's shapes are the feedwater nozzles, bottom head and closure flange regions. Since the temperature of the bottom head can lag behind the rest of the vessel under certain non-nuclear heatup/cooldown situations, Figure 3.6.2 for both units contains a second curve, (B_{BH}) for the bottom head limits. Curve B on Figure 3.6.2 will be used for the feedwater nozzle and vessel flange limits whereas Curve B_{BH} will be used for the bottom head CRD penetration limits. Curve B_{BH} was generated in accordance with the fracture toughness requirements of 10CFR50, Appendix G, as supplemented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. The predicted irradiation shifts for the beltline materials are low enough that the beltline is not predicted to be limiting through 32 EFPY of operation.

The new curve for the bottom head limits will also meet the safety design basis of the reactor vessel and appurtenances as cited previously in Section 4.2 of the Final Safety Analysis Report.

Category 2 Changes:

The Category 2 changes are administrative because they do not impact plant equipment or systems, plant operations, or testing. These administrative changes will improve the Technical Specifications by correcting typographical errors, updating and improving terminology and information, and deleting unnecessary material.

The removal of the schedule for the withdrawal of the reactor vessel material specimens would be considered the removal of unnecessary material. As stated in Generic Letter 91-01, the removal from the Technical Specifications of the schedule for the withdrawal of reactor vessel material surveillance specimens will not result in any loss of regulatory control because changes to this schedule are controlled by the requirements of Appendix H to 10 CFR Part 50. In addition, to ensure that the surveillance specimens are withdrawn at the proper time, the surveillance requirements in the Technical Specification on pressure and temperature limits indicate that the specimens shall be removed and examined to determine changes in their material properties, as required by Appendix H. As stated in the Generic Letter, a request for a license amendment to remove this table from the Technical Specifications may be made based upon this guidance. As also stated in the Generic Letter, "the licensee should commit to maintain the NRC-approved version of the specimen withdrawal schedule in the UFSAR." The PBAPS UFSAR will be revised to incorporate the specimen withdrawal schedule.

Therefore, the above administrative changes are of no safety significance.

NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATIONS

Category 1 Changes:

The Category 1 changes requested herein do not involve a significant hazards consideration based on the foregoing Safety Assessment for the following reasons:

i) The proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated because the revised thermal and pressurization limits prohibit conditions where brittle fracture of reactor vessel materials is possible. Consequently, there will be no increase in the probability or consequences of previously evaluated accidents since the primary coolant pressure boundary integrity will be maintained as assumed in the safety design analyses.

The RT_{NDT} used to evaluate the new Unit 3 pressure-temperature limits for the beltline material and the Units 2 and 3 bottom head limits was based on the guidance in Regulatory Guide 1.99, Revision 2, which is the latest guidance on RT_{NDT} determinations. The revised Unit 3 pressure-temperature limit curves and bottom head curve for Units 2 and 3 were conservatively generated in accordance with the fracture toughness requirements of 10 CFR 50, Appendix G, as supplemented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. The proposed Unit 3 minimum allowable temperature at which head bolting studs may be under tension is also in accordance with 10 CFR 50, Appendix G, as supplemented by Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code.

Removal of Figure 3.6.4 is of no safety significance because it was for information only and is no longer appropriate.

ii) The proposed revisions do not create the possibility of a new or different kind of accident from any accident previously evaluated because the revised Unit 3 thermal and pressurization limits and the addition of the Units 2 and 3 bottom head curve do not create any new kind of operating mode or introduce any new potential failure mode. Conditions where brittle fracture of primary coolant pressure boundary materials is possible will be avoided by use of the revised and new curves.

iii) The proposed revisions do not involve a significant reduction in a margin of safety because the proposed pressure-temperature limits provide sufficient safety margin. The revised Unit 3 pressure-temperature limits and the new Units 2 and 3 head curves, were established in accordance with current regulations and the latest regulatory guidance on RT_{NDT} determinations. Although there is some reduction in safety margin, operation within the new limits will ensure that the reactor vessel materials will behave in a non-brittle manner and will remain conservative in that the original safety design bases will be preserved.

Category 2 Changes:

The NRC provided guidance concerning the application of the standards for determining whether license amendments involve significant hazards considerations by providing examples in 51 FR 7751. An example (Example i) of a change that involves no significant hazards considerations is "a purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature." The Category 2 changes requested herein conform to this example and do not involve a significant hazards consideration based on the foregoing Safety Assessment for the following reasons:

i) The proposed revisions do not involve a significant increase in the probability or consequences of an accident previously evaluated because they do not affect operations, equipment, or any safety-related activity. Thus, these administrative changes cannot affect the probability or consequences of any accident.

ii) The proposed revisions do not create the possibility of a new or different kind of accident from any accident previously evaluated because these changes are purely administrative and do not affect the plant. Therefore, these changes cannot create the possibility of any accident.

iii) The proposed revisions do not involve a significant reduction in a margin of safety because the changes do not affect any safety related activity or equipment. These changes are purely administrative in nature and increase the probability that the Technical Specifications are correctly interpreted by adding clarifying information, deleting inappropriate information, and correcting errors. Thus, these changes cannot reduce any margin of safety.

ENVIRONMENTAL IMPACT ASSESSMENT

An environmental impact assessment is not required for the changes requested by this Application because the requested changes conform to the criteria for "actions eligible for categorical exclusion" as specified in 10 CFR 51.22(c)(9). The requested changes have been shown by this Application not to adversely affect the objective of the primary coolant pressure boundary to act as a radioactive material barrier. The Application involves no significant hazards consideration as demonstrated in the preceding sections. The Application involves no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and there will be no significant increase in individual or cumulative occupational radiation exposure.

CONCLUSION

The Plant Operations Review Committee and the Nuclear Review Board have reviewed these proposed changes to the Technical Specifications and have concluded that they do not involve an unreviewed safety question and will not endanger the health and safety of the public.