BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	:	
	:	Docket No. 50-277 50-278
PHILADELPHIA ELECTRIC COMPANY	:	

APPLICATION FOR AMENDMENT

OF

FACILITY OPERATING LICENSES

DPR-44 & DPR-56

Edward G. Bauer, Jr.
Eugene J. Bradley

2301 Market Street
Philadelphia, Pennsylvania 19101

Attorneys for
Philadelphia Electric Company

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Philadelphia Electric Company, Licensee under Facility Operating Licenses DPR-44 and DPR-56 for Peach Bottom Atomic Power Station Units 2 and 3, hereby requests that the Technical Specifications contained in Appendix A of the Operating Licenses be amended as indicated by a vertical bar in the margin of the attached pages 31, 80, 148, 180 and 185.

The Licensee proposes to (1) revise the Section 1.2 BASES with regards to the description of the values and codes utilized in establishing the pressure safety limit of the reactor recirculation system, and revise the design pressure of the suction piping resulting from the installation of recirculation system piping which has been analyzed to a later version of the ASME Code (Units 2 and 3); (2) revise Tables 3.7.1, 3.7.4, and 4.2.A to reflect the removal of the Reactor Vessel Head Spray primary containment isolation valves MO-10-32 and 33 (Unit 3); and (3) revise the Surveillance Requirements of Section 4.6.E to reflect the removal of the recirculation system cross-tie piping and equalizer valves (Units 2 and 3).

Confirmatory Order dated March 20, 1986 from the Nuclear Regulatory Commission contained a requirement for the submittal of plans for inspection and/or modification to the recirculation and other reactor coolant pressure boundary piping systems during the next refueling outage for Unit 3. The decision to perform the pipe replacement was confirmed by letter from Licensee to the Nuclear Regulatory Commission dated December 22, 1986 and a description of plans for the recirculation and Residual Heat Removal System piping replacement was submitted to the Nuclear Regulatory Commission in a letter dated May 29, 1987. This matter was discussed with the Nuclear Regulatory Commission staff during a September 9, 1987 meeting with Licensee.

The refueling and pipe replacement outage for Unit 3 was begun in October, 1987. During this outage, the existing Type-304 stainless steel recirculation system piping and safe ends, and portions of the Residual Heat Removal System piping, will be replaced with Type-316NG stainless steel as described in the previously submitted plans. The existing Type-304 stainless steel has been determined to be susceptible to Intergranular Stress Corrosion Cracking (IGSCC). In order to protect against IGSCC, low carbon Type-316NG stainless steel will be used in the replacement piping. NUREG-0313, Revision 1 has identified this material as suitable for use in reactor pressure boundary applications.

A similar pipe replacement was performed on Unit 2 during an outage which began in April, 1984 and ended July, 1985.

This Application proposes changes to the Technical Specifications to reflect these pipe replacements.

In addition to the replacement of IGSCC susceptible piping, other piping design enhancements will be undertaken during the Unit 3 outage. The enhancements which directly affect the Technical Specifications are the removal of the containment isolation valves associated with the removal of the Reactor Vessel Head Spray piping of the Residual Heat Removal System (Unit 3) and the elimination of the cross-tie piping and equalizer valves which connect recirculation Loop A and Loop B (Units 2 and 3).

Nuclear Regulatory Commission approval and issuance of this amendment is needed prior to the startup of Units 2 and 3.

Category of Proposed Changes

The proposed changes to the Technical Specifications proposed herein may be classified into three categories or types of changes:

(1) Changes to the Units 2 and 3 Technical Specifications Section 1.2 BASES which revise the description of the values and codes utilized in obtaining the pressure safety limit of the reactor recirculation system, and Technical Specification changes which revise the design pressure of the suction piping resulting from the design and analysis of the recirculation system piping to an updated code (Category A).

(2) Changes to the Unit 3 Technical Specifications to reflect the removal of the containment isolation valves associated with the Reactor Vessel Head Spray piping removal (Category B).

(3) Changes to the Units 2 and 3 Technical Specifications to reflect the elimination of the cross-tie piping and equalizer valves between Recirculation Loop A and Loop B (Category C).

Discussion of Category A Changes (Units 2 and 3)

The pipe replacement modification for Units 2 and 3 replaces the existing Type-304 stainless steel recirculation system piping and safe ends, and portions of the Residual Heat Removal System and Reactor Water Cleanup System with Type-316NG stainless steel piping.

This modification also includes the removal of the Reactor Vessel Head Spray piping inside containment (Unit 3 only) and removal of recirculation system cross-tie piping and the associated equalizer valves.

The design pressure for the Type-304 Stainless Steel piping being removed was established in accordance with the requirements of ANSI-B31.1.0, 1967 Edition. The design pressure for the Type-316NG replacement recirculation piping and safe ends, and the Residual Heat Removal suction and return piping has been established in accordance with ASME Section III, Subsection NB, 1980 Edition up to and including the Winter 1981 Addenda.

The Category A changes to Section 1.2 <u>BASES</u> reflect the revised design pressure for the Type-316NG stainless steel suction piping as established by the ASME Section III, Subsection NB, 1980 Edition up to and including the Winter 1981 Addenda. Additionally, Section 1.2 <u>BASES</u> has been modified to revise the description of the values and codes utilized in obtaining the pressure safety limit of the reactor recirculation system.

<u>List of Category A Changes (Units 2 and 3)</u>

a. The existing section 1.2 <u>BASES</u> (Page 31) states "The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator assures not exceeding 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel (1250 psig at 575 degrees F) and coolant system piping (suction piping: 1148 psig at 562 degrees F; discharge piping: 1326 psig at 562 degrees F)." It is proposed to add the word "limiting" to the second sentence to clarify that the Technical Specification refers to the "limiting design pressures". Addition of the word "limiting" only provides additional clarity and does not establish a different value than intended from the previous value represented by the "design pressure".

b. Additionally, "coolant system piping" will be replaced with "recirculation system piping" to correctly identify the portion of the primary coolant system that serves as the bases for the limiting design pressure. This proposed change only clarifies the location of "limiting design pressure" and does not change the location from the previously evaluated location which is the recirculation system suction piping. It is also proposed to eliminate reference to the discharge piping because the discharge piping does not now, nor did before, represent the location of the limiting design pressure. Therefore, it is proposed to eliminate the words "discharge piping: 1326 psig at 562 degrees F".

c. To reflect the change in the limiting design pressure of the suction piping which resulted from the use of the later version of the ASME Code, it is also proposed to replace "1148 psig" and "562 degrees F" with revised values of "1250 psig" at "575 degrees F". The proposed Technical Specification would state "The 1375 psig value is derived from the limiting design pressures of the reactor pressure vessel (1250 psig at 575 degrees F) and recirculation system piping (suction piping: 1250 psig at 575 degrees F)".

d. It is proposed to eliminate the discussion of the selection of the pressure safety limit and the reference to the ANSI B31.1.0 Code. The limiting design pressure is based on the recirculation system suction piping which is analyzed to the ASME Boiler and Pressure Vessel Code Section III. Reference to the ANSI B31.1.0 Code is no longer appropriate because the recirculation system piping analyzed to the ANSI B31.1.0 Code is being removed. Additionally, a discussion which identifies that the pressure safety limit is based upon two different codes is no longer applicable. Therefore, it is proposed to eliminate the words "The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and ANSI B31.1.0 for the reactor coolant system piping."

e. It is also proposed to eliminate the words "The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the ANSI Code permits pressure transients up to 20% over the design pressure (120% X 1148 = 1378 psig; 120% X 1326 = 1591 psig)." These words will be replaced with the words "The pressure safety limit is set in accordance with the ASME Boiler and Pressure Vessel Code Section III to limit the maximum pressure to less than 110% of the design pressures for the reactor vessel (110% X 1250 = 1375 psig) and the recirculation system piping (suction: 110%

X 1250 = 1375 psig)". The proposed revision described above will clarify that the pressure safety limit is set in accordance with ASME Boiler and Pressure Vessel Code Section III and that the ANSI code is no longer applicable because the limiting design pressures are based on ASME Boiler and Pressure Vessel Code Section III. Additionally, the existing values designated for the design pressure of the suction piping upon which the pressure safety limit is based, are proposed to be changed from "120%", "1148 psig" and "1378 psig", to "110%", "1250 psig" and "1375 psig".

The reference to the discharge piping values ("120% X 1326 = 1591 psig") is proposed to be eliminated because the pressure safety limit is now, and was before, based on the suction piping and not the discharge piping of the recirculation system.

<u>Safety Assessment for Category A Changes (Units 2 and 3)</u>

The proposed revisions described above concerning the pressure safety limit do not alter the previously established value of the pressure safety limit and therefore do not affect the safety of the plant.

Additional clarifications are proposed to be added to the Section 1.2 <u>BASES</u> to state that the pressure safety limit is derived from the design pressures of the reactor pressure vessel and the suction piping of the recirculation system piping. This change only provides clarification and does not alter the previously evaluated location. References to the discharge piping also have been eliminated because the pressure limits for the discharge piping are not limiting.

The reference in the Technical Specification to ANSI B31.1.0 piping are proposed to be deleted since the B31.1.0 piping has been replaced with piping designed, fabricated, and installed in accordance with ASME Section III, Subsection NB requirements. The pressure safety limit is based on the limiting design pressures of the reactor pressure vessel and the new Type-316NG recirculation system suction piping. Therefore, references to ANSI B31.1.0 are no longer applicable because the limiting design pressure for the Type-316NG stainless steel has been established in accordance with the ASME Section III, Subsection NB, 1980 Edition up to and including the Winter 1981 Addenda.

<u>Significant Hazards Consideration for Category A Changes</u>

The proposed revision of the Section 1.2 <u>BASES</u> with regard to the description of the values and codes utilized in establishing the pressure safety limit of the reactor coolant system does not involve a Significant Hazards Consideration. In order to support a "No Signification Hazards Consideration" determination, the necessary background supporting information is provided below along with an evaluation of each of the three standards set forth in Title 10 CFR Section 50.92. Operation of the plant under the proposed Technical Specification would not:

i) <u>Involve a significant increase in the probability or consequences of an accident previously evaluated.</u>

This change reflects the use of ASME Section III as the design code for the new recirculation and Residual Heat Removal System piping. The increased design pressures are consistent with currently accepted criteria for nuclear piping. No change in the reactor system "over pressure set point" is required as a result of this change; the reactor vessel and the recirculation suction piping remain the limiting components in the system. Consequently, the probability or consequences of any accident previously evaluated in Chapter 14 of the Final Safety Analysis Report have not been increased. Additionally, removal of the references to the ANSI B31.1.0 Code in the Section 1.2 BASES is appropriate because the new recirculation system piping pressure limits have been established in accordance with the ASME Section III Code recognized in the Commission's regulations. Removal of the references to the discharge piping is appropriate because the pressure safety limit is based upon the suction piping and not the discharge piping. The removal of the reference to the discharge piping provides greater clarification to the Section 1.2 BASES. Consequently, the probability or consequences of any accident previously evaluated in Chapter 14 of the Final Safety Analysis Report have not been increased.

ii) Create the possibility of a new or different kind of accident from any previously evaluated.

The revised design pressure of the recirculation system suction piping is in the conservative direction and will not create a new or different accident then previously evaluated in Chapter 14 of the Final Safety Analysis Report. Additionally, removal of the references to the ANSI B31.1.0 Code and discharge piping will not create a new or different accident than previously evaluated in Chapter 14 of the Final Safety Analysis Report.

iii) Involve a significant reduction in a margin of safety.

The use of Type-316NG stainless steel recirculation system suction piping has resulted in a design pressure which allows greater margin of safety above normal operating pressure. Additionally, removal of the references to the ANSI B31.1.0 Code and discharge piping will not affect the margin of safety, nor affect any previous accident analysis evaluated in Chapter 14 of the Final Safety Analysis Report since the current design of the recirculation system reflects current requirements specified in the Commission's regulations.

Conclusion

Based on the three standards discussed above, the changes to the Technical Specifications involve no significant hazards considerations.

Discussion of Category B Changes (Unit 3)

As part of the piping modifications for Unit 3 , the Reactor Vessel Head Spray piping of the Residual Heat Removal System and the associated containment isolation valves are proposed to be removed to eliminate a portion of the primary coolant system that is susceptible to IGSCC. The Category B changes to the Technical Specifications reflect these modifications.

List of Category B Changes (Unit 3)

a. Existing Table 4.2.A (Page 80) contains a subsection entitled "Logic System Functional Test (4)(6)". Item 2) of this subsection includes the words "Head Spray". The logic system involves the isolation logic for the Head Spray valves. It is proposed to delete the existing words "Head Spray". This revision recognizes the proposed removal of the Head Spray primary containment isolation valves.

b. Existing Table 3.7.1 (Page 180) entitled "Primary Containment Isolation Valves" contains the words "RHRS Reactor Vessel Head Spray Isolation Valves" and their associated references in this table. It is proposed that these words and the associated table references be deleted. This change is necessary to reflect the deletion of the Head Spray primary containment isolation valves.

c. Existing Table 3.7.4 (Page 185) entitled "Primary Containment Testable Isolation Valves" contains the motor operated valves "MO-10-32; MO-10-33". It is proposed to eliminate these valves and the associated penetration numbers from the table. This change is appropriate to reflect the removal of the Head Spray primary containment isolation valves.

Safety Assessment for Category B Changes (Unit 3)

The Reactor Vessel Head Spray has no safety function and no credit for its use has been taken in the accident or transient analyses described in the Peach Bottom Final Safety Analysis Report or in the emergency operating procedures. The Reactor Vessel Head Spray System was intended for use during shutdown cooling to increase the reactor vessel head cooldown rate, however, experience at Peach Bottom Unit 3 and other Boiling Water Reactors has shown that the rapid cooldown of the reactor head is not necessary to speed refueling activities. In fact, elimination of the Reactor Vessel Head Spray piping should actually reduce refueling critical path time and reduce personnel radiation exposure by eliminating the need to disassemble and reassemble the piping during reactor pressure vessel head removal and installation. Removal of the Reactor Vessel Head Spray System would neither increase or decrease Residual Heat Removal System reliability since it has no impact on other operating modes of the system.

The containment isolation valves associated with the head spray piping are proposed to be removed and the penetration capped on the inboard and outboard side of the containment to maintain the pressure boundary. Integrity of this modified penetration will be assured by the integrated leak test performed on the containment.

Consequently, it is concluded that the removal of the Head Spray function and its isolation valves have no adverse impact on plant safety since the system has no safety function, and capping of this penetration assures containment integrity.

Significant Hazards Consideration for Category B Changes (Unit 3)

The proposed removal of the Reactor Vessel Head Spray piping of the Residual Heat Removal System and the associated containment isolation valves does not involve a Significant Hazards Consideration. In order to support a "No Significant Hazards Consideration" determination, the necessary background supporting information is provided below along with an evaluation of each of the three standards set forth in 10 CFR Section 50.92. Operation of the plant under the proposed Technical Specifications would not:

(i) Involve a significant increase in the probability or consequences of an accident previously evaluated.

This change eliminates direct supply of cooling water to the vessel head region during shutdown. Additionally, the associated containment isolation valves will be removed and the containment penetration capped. The Residual Heat Removal System, independent of the head spray feature, is capable of reducing the reactor vessel to temperatures below 125 degrees F. within approximately 20 hours after inserting the control rods. The Reactor Vessel Head Spray System is merely an additional feature which was intended to expedite the shutdown cooling process and routine refueling. However, experience at Peach Bottom has shown that this capability has not been utilized because rapid head cooling is not needed to expedite the start of refueling activities. Because Reactor Vessel Head Spray is not required for achieving or maintaining shutdown cooling, no credit is taken for this capability in any of the Final Safety Analysis Report Chapter 14 analyses. The containment isolation valves will be removed and the penetration will be capped on the inboard and outboard side of the containment maintaining the pressure boundary. Therefore, the removal of the Reactor Vessel Head Spray System and the associated containment isolation valves does not increase the probability or consequences of any accident previously evaluated in Chapter 14 of the Final Safety Analysis Report.

(ii) <u>Create the possibility of a new or different kind of accident from any previously evaluated.</u>

The removal of Reactor Vessel Head Spray System and the associated containment isolation valves would neither increase or decrease Residual Heat Removal System reliability or impact on any other operating mode of the Residual Heat Removal System. As discussed previously, the Head Spray System does not perform a safety function. Removal of the containment isolation valves and capping the penetration establishes a passive primary containment boundary not subject to the effects of isolation valve degradation and malfunction. Elimination of a nonsafety function and the replacement of the containment isolation valves with a passive containment boundary provides protection at least equivalent to the present level and does not create the possibility of a new or different kind of accident.

(iii) Involve a significant reduction in a margin of safety.

The plant safety design basis is not affected by removal of the Reactor Vessel Head Spray piping and the associated containment isolation valves. The Reactor Vessel Head Spray System has no safety function and no credit is taken for its presence in Chapter 14 of the Final Safety Analysis Report analyses. However, the Technical Specifications must be amended to reflect the deletion of the Head Spray isolation valves. Since the system function is being removed and the associated pipe which contains these valves is being removed, there is no longer a surveillance requirement for these valves. Removal of the Head Spray piping and associated valves eliminates a portion of the primary coolant system that is susceptible to IGSCC degradation; therefore, a potential location for a primary system pipe break. Additionally, removal of the valves eliminates the potential for degradation of containment integrity due to valve malfunction. The containment penetration will be capped and the pressure boundary maintained. Consequently, the margin of safety is enhanced.

Conclusion

Based on the three standards discussed above, the changes to the Technical Specifications involve no significant hazards considerations.

Discussion of Category C Changes (Units 2 and 3)

As part of the piping modifications for Unit 3, the recirculation loop cross-tie piping and the associated equalizer valves (MO-65A, MO-65B and MO-66A, MO-66B) are being removed to eliminate a portion of the primary coolant system that is susceptible to IGSCC degradation. This modification was implemented on Unit 2 during a previous outage.

The original plant design provided a cross connect line with two normally closed valves, between the two recirculation loops (A & B) (Final Safety Analysis Report Section 1.6.1.3.3).

The design intent of this cross-tie line was to provide, when in single loop operation (SLO), a means of having the capability to promote equal flow distribution through loops A and B headers. However, the Peach Bottom Units 2 and 3 safety analyses for SLO were performed assuming the cross-tie valves (equalizer valves) to be closed, and not used. SLO as analyzed does not permit the use of the cross-tie line.

The current Units 2 and 3 Technical Specifications, Surveillance Requirements, Section 4.6.E.2 ("Jet pumps") reflects the non-use of the cross-tie line and equalizer valves when operating with one recirculation pump.

Summary of the Category C Change (Units 2 and 3)

a. The existing Surveillance Requirement 4.6.E.2 states "Additionally when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure ...". The proposed Technical Specification would eliminate the words "with the equalizer valves closed" since removal of the equalizer valves and cross-tie piping eliminate the need for this restriction. Therefore, the proposed Technical Specification would state "Additionally when operating with one recirculation pump, the diffuser to lower plenum differential pressure..."

Safety Assessment for the Category C Change (Units 2 and 3)

The piping modifications include the removal of the recirculation loop cross-tie piping and the associated equalizer valves. Based on the above discussion, the cross-tie piping and equalizer valves serve no safety or operational function. Consequently, the removal of the recirculation loop cross-tie piping and the associated equalizer valves and respective bypass lines will have no adverse impact on plant safety. The purpose of Surveillance Requirement 4.6.E.2 is to establish additional surveillance and operability requirements when operating with only one recirculation pump with the equalizer valves closed. Removing the equalizer valves does not impact the ability to comply with this Surveillance Requirement since the removal of the cross-tie piping and equalizer valves is equivalent to the equalizer valves being in the closed position.

Significant Hazards Consideration for Category C Changes (Units 2 and 3)

The removal of the cross-tie piping and equalizer valves does not involve a Significant Hazards Consideration. In order to support a "No Significant Hazards Consideration" determination, the necessary background supporting information is provided below along with an evaluation of each of the three standards set forth in 10 CFR Section 50.92. Operation of the plant under the proposed Technical Specifications would not:

(i) involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposed change reflects the elimination of the cross-tie piping and equalizer valves which provide a function previously identified as not being required for the safe operation of Peach Bottom Units 2 and 3. The two equalizer valves in the line are maintained in the locked-closed position during power operations. The cross-tie line was intended to provide the capability to promote equal flow distribution through Loops A and B during single loop operation (SLO). The Nuclear Steam System Supplier previously concluded that adequate core flow can be obtained during SLO with one recirculation pump operating and the cross-tie line closed. The Peach Bottom Units 2 and 3 safety analyses for SLO was performed assuming the valves are closed and not used. No credit has been taken for use of the cross-tie piping and equalizer valves in any Chapter 14 analysis. Therefore, the removal of the recirculation cross-tie piping and valves does not increase the probability or consequences of an accident previously evaluated in Chapter 14 of the Final Safety Analysis Report.

(ii) Create the possibility of a new or different kind of accident from any previously evaluated.

The equalizer valves have never been used during reactor power operations. As mentioned above, the safety analysis for SLO assumed that the valves are close and not used. Elimination of the cross-tie line, previously deemed not to be required for the safe operation of the plant, does not create a new or different kind of accident.

(iii) <u>Involve a significant reduction in a margin of safety.</u>

The cross-tie line has no safety function and no credit for its use has ever been taken in any accident or transient analysis or the emergency operating procedures. Removal of the cross-tie line would neither increase or decrease recirculation reliability since it has no impact on the recirculation system. Removal of the cross-tie line is beneficial in that it removes a potential location for a primary system pipe break and consequently maintains or enhances the margin of safety. The purpose of Surveillance Requirement 4.6.E.2 is to establish additional surveillance and operability requirements when operating with only one recirculation pump with the equalizer valves closed. Removing the equalizer valves does not impact the ability to comply with this Surveillance Requirement since the removal of the cross-tie piping and equalizer valves is equivalent to the equalizer valves being in the closed position.

Conclusion

Based on the three standards discussed above, the operation of the facility after making the proposed Category C changes to the Technical Specifications involve no significant hazards considerations.

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 unreviewed safety questions or involve Significant Hazards Considerations and will not endanger the health and safety of the public.

Respectfully submitted,
PHILADELPHIA ELECTRIC COMPANY

By JW Gallagher
Vice President

COMMONWEALTH OF PENNSYLVANIA :

: ss.

COUNTY OF PHILADELPHIA :

J. W. Gallagher, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company, and that he has read the foregoing Application for Amendment of Facility Operating License 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.

JW Gallagher

Vice President

Subscribed and sworn to before me this 16th day of March, 1988

Melanie R. Campanella

Notary Public

MELANIE R. CAMPANELLA
Notary Public, Philadelphia, Philadelphia Co.
My Commission Expires February 12, 1990

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Application were served on the following by deposit in the United States mail, first-class postage prepaid, on the 21st day of March, 1988.

William T. Russell, Regional Administrator
U. S. Nuclear Regulatory Commission
Region 1
631 Park Avenue
King of Prussia, PA 19406

T. P. Johnson, Resident Inspector
U. S. Nuclear Regulatory Commission
Peach Bottom Atomic Power Station
P. O. Box 399
Delta, PA 17314

Mr. Thomas Gerusky, Director
Bureau of Radiological Protection
Department of Environmental Resources
P. O. Box 2063
Harrisburg, PA 17120

Eugene J. Bradley

Attorney for
Philadelphia Electric Company