BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of	:	Docket Nos. 50-277
	:	50-278
PHILADELPHIA ELECTRIC COMPANY	:	

APPLICATION FOR AMENDMENT

OF

FACILITY OPERATING LICENSES

DPR-44 & DPR-56

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Philadelphia Electric Company, Licensee under Facility Operating Licenses DPR-44 and DPR-56 for the Peach Bottom Atomic Power Station Unit No. 2 and Unit No. 3, respectively, hereby requests that the Technical Specifications contained in Appendix A of the Operating License be amended by revising certain sections as indicated by a vertical bar in the margin of the attached pages 11a, 12, 15, 21, 61, 63, 72, 79, 80, 89, 182, 199 and 252. The Application requests the following changes: (1)

correct errors and establish consistency in the reactor water level setpoint values, (2) lower the main steam line isolation valve low water isolation setpoint from low-low to low-low-low reactor water level, and (3) revise the audit frequency of the Facility Emergency Plan and implementing procedures to conform with the Commission's regulations. A discussion of each of the requested changes is set forth below.

1) The following revisions are requested in the reactor water level setpoints specified in the Technical Specifications.

   a) The current Technical specifications specify reactor water level setpoints to three different reference points, e.g., in terms of inches above top of active fuel, inches above vessel zero, and instrument indicated level. This inconsistency results in multiple values that are confusing to the reader and may result in possible misinterpretation of the specifications. The proposed changes on pages 12, 63, 72 and 89 would establish consistency by identifying the setpoint in terms of the "instrument indicated level", with reference to the "inches above vessel zero". The "instrument indicated level" was selected to enhance operator proficiency since this represents the values displayed on the reactor level instruments. The reference to the "inches above vessel zero" establishes

the level setpoints to a fixed physical elevation and therefore reduces the chances of an inadvertent change in the setpoint. The reference to the "top of active fuel" is deleted.

b) The current Technical Specification on page 12 identifies the Core Spray and LPCI actuation setpoint as minus 159.5 inches indicated level (low-low-low level) and 378 inches above vessel zero. Tables 7.4.2, 7.4.3 and 7.4.4 of the Peach Bottom Final Safety Analysis Report, as well as design information provided by General Electric Company, confirms the 378 inch value. However, since the level instrumentation reference zero is set at 538 inches above vessel zero, the indicated level should be minus 160 inches, not 159.5 inches. General Electric Company also confirms this minor error. The same setpoint actuation level is correctly shown as minus 160 inches on page 64 of the Technical Specifications. The error identified above has existed since the issuance of the original license and appears to be the result of an error during the preparation of the original license while the original proposed Technical Specifications were being revised to conform with the Technical Specifications of another Licensee within a limited time schedule. The proposed change

would specify the correct value of minus 160 inches on page 12 for the Core Spray and LPCI actuation setpoint.

c) The current Technical Specifications on pages 12, 61 and 79 identifies either minus 49.5 or minus 49.0 inches indicated level (low-low level) as the setpoints for HPCI/RCIC actuation, recirculation pump trip, and primary containment isolation. According to Tables 7.3.2 and 7.4.1 of the Peach Bottom Final Safety Analysis Report, as well as design information provided by General Electric Company, the correct value should be minus 48 inches indicated level which corresponds with 490 inches above vessel zero. The low-low level on page 64 of the Technical Specifications was originally specified at minus 49 inches and subsequently corrected to minus 48 inches by License Amendment 69/68, dated May 16, 1980, in response to an Application submitted April 15, 1980. The origin of these errors appears to be error during the preparation of the original license. The proposed change would specify the correct value of minus 48 inches indicated level on pages 12, 61 and 79 of the Technical Specifications.

d) The proposed change on page 21 of the Technical Specification Bases would correctly state the relationship between the scram setting and the normal

operating level. The normal operating level is at plus 23 inches indicated level, which is approximately the midpoint between the reactor high water level trip (trips High Presure Coolant Injection, Reactor Core Isolation Cooling, and Turbine Stop Valves) set at plus 45 inches indicated level and the reactor low water level trip (scram setpoint) set at zero inches indicated level, in order to minimize spurious trips due to feed water transients. Therefore, the scram setting is approximately 23 inches, not 31 inches, below the normal operating range.

e) The final change involving reactor water levels specified in the Technical Specification deals with the minimum water level required in the shutdown condition as shown on page 11a. The Standard Technical Specifications for General Electric Boiling Water Reactors (NUREG-0123, Rev. 3, page 2-2) requires the reactor water level to be maintained above the top of the active irradiated fuel in the shutdown mode. The Peach Bottom Technical Specifications had conservatively set this safety limit to correspond with the low-low-low level setpoint (378 inches above vessel zero). Based on the original top of active fuel elevation of 360.3 inches above vessel zero, a valve of 17.7 inches above the active fuel was specified. While the Bases of page

15 correctly identified the safety value as 17.7 inches above the top of the fuel, the specification on page 11a erroneously specifies 17.1 inches as a result of an apparent typographical error. The proposed change would correct this error, and identify the safety limit in terms of the format described in change (a) above, i.e. inches indicated level. Consequently, the safety limit is specified as minus 160 inches indicated level (378 inches above vessel zero). A discussion of the safety implications of the new fuel design on the reactor water level setpoints is provided below.

Safety Evaluation

Originally, the reactor cores at Peach Bottom Units 2 and 3 employed a 7 x 7 fuel rod array containing fuel rods of 144 inches fuel column length. Subsequently, fuel assemblies containing an 8 x 8 fuel rod array of both 144 and 146 inch fuel column lengths were introduced. More recently, fuel assemblies containing an 8 x 8 fuel rod array of 150-inch fuel column length were introduced such that the reactor cores contained a mix of fuel assemblies containing the above described fuel column lengths. For the current operating cycles, both Peach Bottom Units 2 and 3 are employing only the 150-inch fuel column length (6 inches of natural uranium

added to top of fuel column). These fuel changes were described in previous fuel reload amendment applications.

The proposed changes to the reactor water level setpoints maintain the low-low level and low-low-low levels at 490 and 378 inches above vessel zero, respectively. The dimensional change associated with the new fuel designs does not impact the Emergency Core Cooling System (ECCS) evaluations for Peach Bottom 2 and 3 since the values for low-low and low-low-low setpoints used in the ECCS analysis are 486.5 and 366.4 inches above vessel zero, respectively.

The proposed changes revise the reactor water low-level setpoints in order to achieve consistency throughout the Technical Specifications and correct errors in setpoint values. Both the current and proposed low-level setpoints are more conservative than the setpoint values assumed for the ECCS analysis performed as the bases for licensing the facility after each reactor fuel reload. Consequently, the proposed changes do not involve a significant hazards consideration since the request does not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety.

2) As part of the Licensee's response to NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.K.3.16, the Licensee proposes to reduce the instrument setpoint which closes the Group 1 Isolation Valves of the Primary Containment Isolation System (PCIS) on reactor low water level from "low-low" to "low-low-low" reactor water level. The modification will reduce the Safety Relief Valve (SRV) challenge rate, and consequently the probability that a relief valve will fail, due to the reduced MSIV isolation demands resulting from the lower level isolation setpoint. This conclusion is supported by an evaluation submitted to the NRC on March 31, 1981, by the BWR Owners' Group in response of NUREG-0737, "Clarification of TMI Action Plan Requirements", Item II.K.3.16. Licensee endorsed this modification in correspondence dated April 2, 1981 (J. W. Gallagher, Philadelphia Electric Co. to D. G. Eisenhut, NRC) as a means of implementing the NRC requirement in NUREG-0737, Item II.K.3.16, to reduce challenges to the SRV's. Furthermore, the change will mitigate a portion of the induced loads of SRV subsequent actuations (second pops) during a postulated small break LOCA.

The change has no effect on ECCS performance and will have no effect on large break LOCA performance because of the rapid inventory loss and reactor depressurization which result for large break accidents. The change only affects the small

break LOCA by increasing the small break peak cladding temperature (PCT). However, as substantiated in General Electric Company's analysis titled, "Evaluation of Mark I SRV Load Cases C3.2 and C3.3 for Peach Bottom Atomic Power Station Units 2 and 3", document NEDC-24367, dated September 1981, this increase will result in a PCT of approximately 1553 degrees F which is substantially less than both the PCT resulting from large break accidents and the 2200 degrees F PCT limit given in Appendix K of 10 CFR 50.

The probabilty of an accident or malfunction of equipment important to safety is not increased. Although the PCT for small breaks is increased by this modification, analysis shows that these consequences are still less severe than the limiting design basis accident (DBA) break and within the 10 CFR 50, Appendix K limits.

Accordingly, Licensee requests that the instrumentation for initiating primary containment isolation as specified in Table 3.2.A on page 61 and in Table 4.2.A on page 80 be changed from "Reactor Low-Low Water Level" to "Reactor Low-Low-Low Water Level". Likewise, Licensee requests that the conditions for actuation of the valves in Group 1 of the Primary Containment Isolation System (PCIS) as specified under the Notes for Table 3.7.1 on page 182 and in the Bases for Sections 3.7.D and 4.7.D on page 199 be changed from

"Reactor Vessel Low-Low Water Level" to "Reactor Vessel Low-Low-Low Water Level".

Additionally, Licensee requests that the Limiting Safety System Setting, as specified in Section 2.1.K on page 12, be changed from "$\geq$ minus 48 in. indicated level ($\geq$ 490 inches above vessel zero)" to "$\geq$ minus 160 in. indicated level ($\geq$ 378 inches above vessel zero)" and the trip level setting, as specified in Table 3.2.A on page 61, be changed from "at or above -48" indicated level" to "at or above -160" indicated level". Licensee also requests that the instrument setpoint referenced in Note 4 under the Notes for Table 3.2.A on page 63 be changed from "490 inches above vessel zero" to "384 inches above vessel zero". Furthermore, Licensee requests that the Bases for Section 3.2, as specified in the last paragraph on page 89, be revised to reflect the proposed change in the reactor low-water level instrument setpoint which closes the Main Steam Line Isolation Valves, Main Steam Drain Valves and Recirc. Sample Valves from "low-low reactor water level" to "low-low-low reactor water level".

Since the change reduces the probability of a severe transient without impacting the limiting design basis parameters, it is deemed as not involving a significant hazards consideration as defined in 10 CFR 50.92, and as

clarified in example (vii) of 48 FR 14870 for actions not likely to involve a significant hazards consideration.

Licensee requests that the pages 11a, 12, 15, 21, 61, 63, 72 and 79 of the proposed Technical Specifications take effect immediately upon approval of this Amendment application to Facility Operating Licenses DPR-44 and DPR-56, and that pages 80, 89, 182 and 199 of the proposed Technical Specifications take effect upon completion of the modifications, authorized by approval of this Amendment Application, to lower the actuation of the Group 1 Primary Containment Isolation Valves from -48 inches indicated level (low-low reactor water level) to -160 inches indicated level (low-low-low reactor water level). Likewise, footnotes have been added to pages 12, 61 and 63 of the proposed Technical Specifications to reflect the fact that those sections identified by the footnotes are not to take effect until the modifications authorized by approval of this Amendment Application are completed.

3) Licensee hereby requests that the requirement for auditing of the Facility Emergency Plan and Implementing Procedures, as specified in Section 6.5.2.8.e on page 252 of Appendix A of the Operating Licenses, be revised to specify performing the audit at least "once per year" rather than "once per two years". The proposed revision would bring the Licensees existing plant - specific technical specifications into

compliance with the requirements of 10 CFR 50.54(t) as requested by the Commission's Generic Letter No. 82-17 (D. G. Eisenhut, NRC, to Licensee) dated October 20, 1982. Although the existing Technical Specifications indicate performanace of the review at least once every two years, the Licensee has implemented independent review of its emergency preparedness program on an annual basis in order to conform with 10 CFR 50.54(t). Since the change is administrative in nature and is requested to conform the Peach Bottom Technical Specifications with the Commission's regulations, it is deemed as not involving a significant hazards consideration as defined in 10 CFR 50.92.

Pursuant to 10 CFR Section 170.22, "Schedule of Fees for Facility License Amendments", Philadelphia Electric Company proposes that this Application for Amendment be considered a Class III Amendment for Unit 2 and a Class I Amendment for Unit 3, since the proposed changes are either administrative in nature or have acceptability for issues clearly identified by an NRC position or involve a single safety issue and are deemed not to involve a significant hazards consideration.

The Plant Operation Review Committee and the Nuclear Review Board (off-site safety review committee) have reviewed the proposed changes to the Technical Specifications and have concluded that they do not involve an unreviewed safety question

cr a significant hazard consideration and will not endanger the health and safety of the public.

Respectfully submitted,
PHILADELPHIA ELECTRIC COMPANY

By ______________________
Vice President

COMMONWEALTH OF PENNSYLVANIA :
: ss.
COUNTY OF PHILADELPHIA :

S. L. Daltroff, being first duly sworn, deposes and says:

That he is Vice President of Philadelphia Electric Company, the Applicant herein; that he has read the foregoing Application for Amendment of Facility Operating Licenses 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.

S L Daltroff
____

Subscribed and sworn to
before me this 9th day
of April, 1984

Patricia A. Jones
____
Notary Public

PATRICIA A. JONES
Notary Public, Phila., Phila. Co.
My Commission Expires Oct. 13, 1986

CERTIFICATE OF SERVICE

I certify that service of the foregoing Application was made upon the Commonwealth of Pennsylvania, by mailing a copy thereof, via first-class mail, to Thomas R. Gerusky, Director, Bureau of Radiological Protection, P. O. Box 2063, Harrisburg, PA 17120, all this 19th day of April, 1984.

Eugene J. Bradley

Attorney for
Philadelphia Electric Company