August 23, 1994

Docket Nos. 50-277
and 50-278

Mr. George A. Hunger, Jr.
Director-Licensing, MC 52A-5
PECO Energy Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Hunger:

SUBJECT: NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR HEARING, PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3 (TAC NOS. M86826 AND M86827)

Enclosed is a copy of the subject notice for your information. This notice relates to your application dated June 23, 1993, pertaining to an increase in the maximum authorized power limit for Peach Bottom Atomic Power Station, Units 2 and 3, to 3458 MWt.

This notice has been forwarded to the Office of the Federal Register for Publication.

Sincerely,
Original signed by:
Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Joseph W. Shea, Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure:
Notice

cc w/enclosure:
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UNITED STATES NUCLEAR REGULATORY COMMISSION

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-277 AND 50-278

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSE, PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION, AND OPPORTUNITY FOR A HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License Nos. DPR-44 and DRP-56 issued to the Philadelphia Electric Company (the licensee) for operation of the Peach Bottom Atomic Power Station, Units 2 and 3, located in York County, Pennsylvania.

The proposed amendments would revise the facility operating license and Appendix A and B of the operating license to change the maximum core power limit from 3293 MWt to 3458 MWt.

Before issuance of the proposed license amendments, the Commission will have made findings required by the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations.

The Commission has made a proposed determination that the amendment request involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would 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

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reduction in a margin of safety. As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1) The proposed OL changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed power rerate imposes only minor increases in the plant operating conditions. Plant systems, components, and structures have been verified to be capable of performing their intended functions under rerated conditions. When necessary, some components will be modified or replaced prior to implementation of the Power Rerate Program to accommodate the revised operating conditions. No new component or system interactions that could lead to an accident are created. As discussed below, no transient events result in a new sequence of events which could lead to a new accident scenario.

   Anticipated Transients Without Scram (ATWS) Analysis

   The changes to plant parameters are consistent with the results in NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991. Therefore, the response to an ATWS event at rerated power will be consistent with the generic response and is acceptable.

   ECCS-LOCA Analysis

   The current ECCS-LOCA performance analysis already bounds the rerated power conditions. The peak clad temperature for rerated conditions is 1,516°F which is below the 10CFR50.46 required limit of 2,200°F. Therefore, the analysis demonstrates that PBAPS, Units 2 and 3 will continue to comply with 10CFR50.46 and 10CFR50, Appendix K at rerated conditions.

   Abnormal Operating Transient Analysis

   The results of the evaluation of transients indicate that the margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR) is unchanged for the 8x8 array fuel types such as the GE9 product line currently in the Unit 2 and Unit 3 cores, and will increase by 0.01 for the GE11 fuel design. The fuel thermal-mechanical limits at power rerate conditions are within the specific design criteria for the GE fuels currently loaded in the PBAPS Unit 2 Cycle 10 core.

Also, the power-dependent and flow-dependent MCPR and Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits developed as part of the core performance improvement program are applicable to rerated conditions. The peak RPV bottom head pressure is still within the ASME requirement for RPV overpressure protection.

The analysis performed focused on the most limiting transient events in each disturbance category selected specifically for the power rerated evaluations. The results demonstrate that PBAPS Units 2 and 3 core thermal power output can be safely increased to the power rerate level without significant impact on the plant safety during a postulated transient event.

a) Events Resulting in a Nuclear System Pressure Increase

i) Main Generator Load Rejection with No Steam Bypass

At rerated conditions, the fuel transient thermal and mechanical overpower results remain below the NRC accepted design criteria.

ii) Main Turbine Trip with No Steam Bypass

The fuel transient thermal responses are less severe than for the Generator Load Rejection event. Therefore, at rerated conditions, this event remains bounded by the Generator Load Rejection event.

iii) Main Steam Isolation Valve Closure, Flux Scram

The peak RPV bottom head pressure for rerated conditions is slightly higher than the RPV bottom head pressure at current rated conditions due to the higher initial system pressure. However, the resultant pressure is still below the ASME overpressure limit of 1,375 psig by a margin of 68 psi.

b) Events Resulting in a Reactor Vessel Water Temperature Decrease

i) Inadvertent HPCI Actuation

For the condition analyzed, both the high water level setpoint and the high RPV steam dome pressure SCRAM setpoint are not reached. Based on the peak average fuel surface heat flux results, the HPCI actuation event will be bounded by the limiting pressurization event with respect to delta Critical Power Ratio ([delta] CPR) considerations. In addition, the fuel transients thermal and mechanical overpower limits remain within the allowable NRC accepted design values.

ii) Feedwater Controller Failure-Maximum Demand

The [delta] CPR calculated for this event at rerated conditions is about 0.01 higher than the corresponding value for the current rated power. However, the trend for the Feedwater Controller Failure-Maximum Demand event is consistent with the analysis for the current rated power. This event continues to be the limiting event at the low core flow condition and is bounded by the limiting Generator Load Rejection event. The fuel thermal margin results are within the acceptable limits for the fuel type analyzed.

iii) Loss of Feedwater Heating

The [delta] CPR for this event at the rerated conditions is bounded by the result estimated for this event at the current rated power level, and remains significantly less than the cycle operating MCPR limit. Because of the round-off process, there is no change between the [delta] CPR results for high and low core flow conditions. However, the results at low core flow conditions are actually slightly higher than for the high core flow condition because of increased inlet coolant subcooling into the reactor core. The calculated thermal and mechanical overpower limits for this event at power rerate conditions also meet the fuel design criteria.

c) Event Resulting in a Positive Reactivity Insertion

i) Rod Withdrawal Error (RWE)

The [delta]CPR calculated for this event at rerated conditions is slightly less than the value for this event at current rated power and is bounded by the generic RWE limits of 0.13 based on implementation of the APRM-Rod Block Monitor TS (ARTS) changes. Therefore, the generic ARTS-based RWE analysis [delta]CPR result is verified for applicability to PBAPS power rerate conditions.

d) Event Resulting in a Reactor Vessel Coolant Inventory Decrease

i) Loss of Feedwater Flow

This transient event does not pose any direct threat to the fuel in terms of a power increase from the initial conditions. However, it is included in the power rerate evaluation to provide assurance that sufficient water make-up capability is available to keep the core covered when all normal feedwater is lost.

The generic analysis results in NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991, show that at power rerate conditions, the minimum water level is reduced by about 1.5 feed from that previously calculated for current rated power, but a large amount of water, more than 5 feet, remains above the top of the active fuel. The sensed water level outside of the core shroud has also been checked to show adequate operational flexibility exists for setting the Level 1 RPV water level setpoint so that it is not expected to be reached even in the conservative case of a HPCI failure. Therefore, PBAPS, Units 2 and 3 will maintain adequate reactor water level during a postulated Loss of Feedwater Flow event at power rerate conditions.

e) Event Resulting in a Core Coolant Flow Decrease

i) Recirculation Pump Seizure

The recirculation pump seizure assumes instantaneous stoppage of the pump motor shaft of one recirculation pump. As a result, the core flow decreases rapidly. The RPV water level swell due to the rapid core flow reduction reaches the high RPV water level setpoint, causing a feedwater pump trip, a turbine trip and subsequently a reactor SCRAM on turbine stop valves closure. The peak neutron flux and average fuel

surface heat flux do not increase significantly above the initial conditions; therefore, no impact on the fuel thermal margin is postulated to occur.

f) Event Resulting in a Core Coolant Flow Increase

i) Recirculation Flow Controller Failure Increasing Flow

The results of this transient for PBAPS, Units 2 and 3 power rerate remain non-limiting as compared with other more severe pressurization events.

g) Performance Improvements

i) Main Turbine Bypass Out-of-Service

The main turbine steam bypass out-of-service condition is included in the input assumptions used in the Abnormal Operating Transient Occurrences analyses for power rerate application. The transient analyses results at power rerate conditions reflect the plant response accounting for this condition.

ii) Single Loop Operation (SLO)

The safety analysis for rerated conditions shows that the SLO mode is valid for power rerate conditions and remains unchanged from the current rated power conditions.

iii) Final Feedwater Temperature Reduction

Final Feedwater Temperature Reduction is a cycle extension mode of operation, used in conjunction with increased core flow (ICF) at the end of a normal operating cycle. The analyses show that for a temperature reduction up to 55°F, this mode of operation is applicable for operation of PBAPS, Units 2 and 3 at the power rerate conditions.

h) Other evaluations

These evaluations included the effect of power rerate on the radiological consequences of accidents presented in UFSAR Subsections 5.2, 14.6 and 14.9. The following bounding analyses were performed: 1) Loss-of-Coolant Accident (LOCA); 2) Main Steam Line Break (MSLB) Accident; 3) Fuel

Handling Accident; 4) Control Rod Drop Accident; and 5) Instrument Line Break Accident.

The analyses shows the offsite radiological consequences for the bounding accidents increase, but remain well within the guidelines of 10CFR100 as discussed in the UFSAR Section 14.9 and the NRC Safety Evaluation Reports for PBAPS, Units 2 and 3. In general, offsite doses are expected to increase proportionally with reactor power. However, a direct comparison between the original analyses and rerate values has limited meaning because the original analyses could not be fully reconstituted. For the fuel handling accident, control rod drop accident, and instrument line break accident, the offsite doses increase by less than 1 rem. For the MSLB accident, the whole body dose remains less than 1 rem and the thyroid dose increases by only 3% from 85 rem to 88 rem. For the LOCA, a re-evaluation of the original analysis was performed. The resultant thyroid dose increased by 19% from 201 rem to 239 rem; however, only about 3% of the increase is due to rerated conditions and 16% due to changes in the analysis model reconstitution. Whole body dose increases slightly to 3.9 rem.

Accident radiological consequences in the Control Room and Technical Support Center (TSC) were also evaluated. The results show doses are well below the 30-day limit of GCC 19 of Appendix A to 10CFR50 (i.e., 5 rem whole body and 30 rem thyroid). A re-evaluation of the original analysis was performed. The highest dose consequence is from a main steam line break which results in a dose of 18 rem thyroid compared to 1.5 rem in the UFSAR. However, only about 3% of this increase is due to rerated conditions and 16% is due to analysis model reconstitution. All whole body doses are less than 1 rem.

An evaluation was performed to address the impact of power rerate on accident mitigation features, structures, systems, and components within the balance of plant. The results are as follows.

- Auxiliary systems such as primary containment chilled water, building Heating, Ventilation, and Air Conditioning (HVAC) systems, reactor building closed loop cooling, service water and emergency service water, high pressure service water, spent fuel pool cooling, process auxiliaries such as instrument air and makeup water and the post-accident sampling system were confirmed to operate acceptably under normal and accident conditions at rerated conditions.

- Combustible gas control systems were confirmed to be capable of maintaining oxygen concentrations inside the primary containment within limits under post accident conditions after implementation of the Power Rerate Program.

- The secondary containment and standby gas treatment system were confirmed to be able to adequately contain, process, and control the release of normal and post-accident levels of radioactivity at rerated conditions.

- Instrumentation was reviewed and confirmed to be capable of performing its control and monitoring functions under rerated conditions.

- Electric power systems including the turbine generator and switchgear components were verified as being capable of providing the electrical load as a result of the rerated power levels. No safety-related electrical loads were affected which would impact the emergency diesel generators.

- Piping systems were evaluated for the effect of operation at higher power levels, including transient loadings. The evaluation confirmed that with few exceptions piping and supports are adequate to accommodate the increased loadings resulting from operation at rerated power conditions. In a few cases, piping supports will be modified to accept the higher forces due to rerated conditions.

- The effect of rerated conditions on high energy line break (HELB) for all NSSS and BOP systems were evaluated. The evaluation confirmed structures, systems, and components important to safety are capable of accommodating the effects of jet impingement and blowdown forces and the environmental effects resulting from HELB events at rerated conditions.

- Control room habitability was evaluated. Post-accident Control Room and TSC doses at rerated conditions were confirmed to be within the limits of GDC 19 of 10CFR50, Appendix A.

- Doses for normal operation at rerated conditions were reviewed and confirmed to remain within the limits of 10CFR20 and 10CFR50, Appendix I. The impact on post-accident sampling activities and post-accident access to vital areas was also confirmed to be acceptable.

- The environmental qualification of equipment important to safety was evaluated for the impact of normal and accident operating conditions at rerated power levels. The majority of equipment remains qualified for the new conditions. For equipment not qualified, corrective actions will be taken to ensure the plant equipment will perform their intended functions under rerated conditions. No new equipment will be added for power rerate which would increase the potential for component failure. The Preventative [Preventive] Maintenance Program (PMP) is not power dependent and will continue to provide for equipment repair or replacement at rerated power conditions.

- The impact of operation at rerated power levels was evaluated for Station Blackout and fire safe shutdown area heat-up concerns. The evaluation confirmed there is no adverse impact from rerated conditions on the ability of the plant to achieve safe shutdown under these conditions.

The consequences of all transients and special events (i.e., ATWS and Station Blackout) remain within NRC accepted criteria for rerated conditions. Concurrent malfunctions assumed to occur during accidents have been accounted for in the safety analyses for rerated conditions. The consequences of these equipment malfunctions do not change with implementation of the Power Rerate Program.

All equipment "Important to Safety" is capable or will be modified/replaced to be capable of performing its intended function. The availability of redundant systems to provide safety functions in the event of component malfunction is not impacted as a result of rerated conditions.

Furthermore, the impact of power rerate on the consequences of abnormal transients and accident conditions which are a result of component malfunctions has been shown to be acceptable.

The probability (i.e., frequency of occurrence) of DBAs occurring is not affected by the increased power level, as the applicable regulatory criteria established for plant equipment (e.g., ANSI Standard B31.1, ASME code, NRC Regulatory Guides) will still be followed as the plant is operated at the rerated power level. Reactor SCRAM setpoints will be established such that there is no significant increase in scram frequency due to rerated conditions. No new challenges to safety-related equipment will result from power rerate.

The changes in consequences of hypothetical accidents which would occur from 102% of the rerated power, compared to those previously evaluated, are in all cases not significant, because the accident evaluations from a power rerate to 105% of original rated power will not result in exceeding the applicable NRC approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and have been determined to meet the current regulatory criteria for PBAPS, Units 2 and 3 at rerated conditions. The offsite doses resulting from DBAs are calculated to increase only a few percent (i.e., approximately 3%) because of the rerated power level and remain below 10CFR100 limits. In the area of core design, the fuel operating limits will still be met at the rerated power level, and fuel reload analyses will show plant transients meet the criteria accepted by the NRC as specified in NEDO-24011, "GESTAR II."

Challenges to fuel or ECCS performance were evaluated and shown to still meet the criteria of 10CFR.46 and 10CFR50, Appendix K. Challenges to the containment have been evaluated and still meet 10CFR50, Appendix A GDC 38, Long Term Cooling, and GDC 50, Containment. Radiological Release events have been evaluated and shown to meet the guidelines of 10CFR100. Therefore, the proposed OL changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) The proposed OL changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

All actions to ensure that safety-related structures, systems, and components will remain within their design allowable values and ensure they can perform their intended functions under rerated conditions will be taken prior to implementation of power rerate. Power rerate does not increase challenges to or create any new challenge to safety-related equipment or other equipment whose failure could cause an accident. No new equipment is added as a result of implementing the Power Rerate Program which would create the possibility of a new type of accident. In addition, power rerate does not create any new sequence of events or failure modes that lead to a new type of accident.

No new operating mode, safety-related equipment lineup, accident scenario, or equipment failure mode was identified as resulting

from the implementation of the Power Rerate Program. The full spectrum of accident considerations defined in NRC Regulatory Guide 1.70 have been evaluated for rerated conditions and no new or different kind of accident has been identified. implementation of the Power Rerate Program uses already-developed technology and applies it within the capabilities of already existing plant equipment in accordance with presently existing regulatory criteria to include applicable NRC approved codes, standards, and methods. GE has designed BWRs of higher power levels than the rerated power of any of the currently operating BWR fleet and no new power dependent accidents have been identified.

Therefore, the proposed OL changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) The proposed OL changes do not involve a significant reduction in a margin of safety.

Power rerate will not involve a significant reduction in a margin of safety, as plant equipment and reactions to transients and hypothetical accidents will not result in exceeding the presently approved NRC acceptance limits.

For systems addressed in the TS Sections 2.1, 2.2, 3.1, 3.2, 3.4, 3.5, 3.6, and 3.7 (i.e., RPS, Protective Instrumentation, SLCS, HPCI, RCIC, Primary System Boundary and Containment Systems) all components will be operable and capable of performing their intended functions under power rerate conditions such that the existing margin of safety is not impacted.

For TS Bases 3.7.A and 4.7.A, the impact of rerated conditions affects LOCA offsite radiological consequences discussed in that section. A re-evaluation of the original analysis was performed. The resultant offsite thyroid dose increased by 19% from 201 rem to 239 rem; however, only about 3% of the increase is due to rerated conditions and 16% is due to the analysis model reconstitution. This preserves adequate margin between expected offsite doses and 10CFR100 guidelines.

The events (i.e., transients and accidents) that form the TS Bases (e.g. TS Bases 2.1, 3.1) were evaluated for rerated conditions. Although some changes to the TS are required for power rerate, no NRC acceptance limit will be exceeded. Therefore, the margins of safety to the safety limits and other TS limits will be maintained.

Therefore, the proposed OL changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received. Should the Commission take this action, it will publish in the FEDERAL REGISTER a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, from

7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555.

The filing of requests for hearing and petitions for leave to intervene is discussed below.

By , the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Proceedings" in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at the Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is

aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch,

or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to Mohan C. Thadani, Acting Director, Project Directorate I-2: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this FEDERAL REGISTER notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to J. W. Durham, Sr., Esquire, Sr. V.P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101, attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated June 23, 1993, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC 20555 and at the local public document room located at the Government Publications Section, State Library of Pennsylvania, (REGIONAL

DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Dated at Rockville, Maryland, this 23rd day of August 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Joseph W. Shea, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

cc:

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