UNITED STATES NUCLEAR REGULATORY COMMISSION

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE INC.

DOCKET NO. 50-388

NOTICE OF CONSIDERATION OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE AND PROPOSED NO SIGNIFICANT HAZARDS

CONSIDERATION DETERMINATION AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of an amendment to Facility Operating License No. NPF-22 issued to Pennsylvania Power and Light Company and Allegheny Electric Cooperative, Inc. (the licensees), for operation of Susquehanna Electric Station, Unit 2, located in Luzerne County, Pennsylvania.

The proposed amendment would change the Technical Specifications (TS) in support of the ensuing Cycle 5 reload. This notice supersedes in its entirety the notice published in the Federal Register on December 12, 1990 (56 FR 1183).

Before issuance of the proposed amendment, 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 request for amendment 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 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:

The following three questions are addressed for each of the proposed Technical Specification changes:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

3. Does the proposed change involve a significant reduction in a margin of safety?

Specification 3/4.2.3. Minimum Critical Power Ratio

The changes to this specification provide new operating limit MCPR curves based on cycle-specific transient analyses.

1. No. Limiting core-wide transients were evaluated with ANF's COTRANSA code (see Summary Report Reference 29) [See incoming application for Summary Report] and this output was utilized by the XCOBRA-T methodology (see Summary Report Reference 30) to determine delta CPRs. Both COTRANSA and XCOBRA-T have been approved by the NRC in previous license amendments. All core-wide transients were analyzed deterministically (i.e., using bounding values as input parameters).

   Two local events, Rod Withdrawal Error and Fuel Loading Error, were analyzed in accordance with the methods described in XN-NF-80-19 (A) Vol. 1 (see Summary Report Reference 7). This methodology has been approved by the NRC.

   Based on the above, the methodology used to develop the new operating limit MCPRs for the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The methodology described can only be evaluated for its effect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in 1. above.

3. No. As stated in 1. above, and in greater detail in the attached Summary Report, the methodology used to evaluate core-wide and local transients is consistent with previously approved methods and meets all pertinent regulatory criteria for use in this application.

   Based on the above, the use of the methodology utilized to produce the U2C5 MCPR operating limits will not result in a significant decrease in any margin of safety.

Specification 3/4.4.1.1.1. Recirculation System (Two Loop Operation)

The changes to this specification (i.e., Figure 3.4.1.1.1-1) reflect cycle-specific stability analyses.

1. No. COTRAN core stability calculations were performed for Unit 2 Cycle 5 to determine the decay ratios at predetermined power/flow conditions. The resulting decay ratios (See Summary Report, Reference 5) were used to define operating regions which comply with the interim requirements of NRC Bulletin No. 88-07, Supplement 1 "Power Oscillations in Boiling Water Reactors," (See Summary Report, Reference 21). As in the previous cycle, Regions B and C of the NRC Bulletin have been combined into a single region (i.e., Region II), and Region A of the NRC Bulletin corresponds to Region I.

   Region I has been defined such that the decay ratio for all allowable power/flow conditions outside of the region is less than 0.90. To mitigate or prevent the consequences of instability, entry into this region requires a manual reactor scram. Region I for Unit 2 Cycle 5 has been calculated to be slightly different than Region I for the previous cycle.

   Region II has been defined such that the decay ratio for all allowable power/flow conditions outside of the region (excluding Region I) is less than 0.75. For Unit 2 Cycle 5, Region II must be immediately exited if it is inadvertently entered. Similar to Region I, Region II is slightly different than in the previous cycle.

   In addition to the region definitions, PP&L has performed stability tests in SSES Unit 2 during initial startup of Cycles 2, 3, and 4 to demonstrate stable reactor operation with ANF 9X9 fuel. The test results for U2C2 (See Summary Report, Reference 22) show very low decay ratios with a core containing 324 ANF 9X9 fuel assemblies.

   Figure 3/4.1.1.1-1 is also referenced by Specification 3/4.4.1.1.2, which governs Single Loop Operation (SLO). The evaluation above applies under SLO conditions as well.

   Based on the above, operation within the limits specified by the proposed changes will ensure that the probability and consequences of unstable operation will not significantly increase.

2. No. The methodology described above can only be evaluated for its effect on the consequences of unstable operation; it cannot create new events. The consequences were evaluated in 1. above.

3. No. PP&L believes that the use of Technical Specifications that comply with NRC Bulletin 88-07 Supplement 1, and the tests and analyses described above, will provide assurance that SSES Unit 2 Cycle 5 will comply with General Design Criteria 12, Suppression of Reactor Power Oscillations. This approach is consistent with the SSES Unit 1 Cycle 6 method for addressing core stability (See Summary Report, References 2 and 4).

Specifications 3/4.4.1.1.2. Recirculation System (Single Loop Operation), and 3/4.4.1.2. Jet Pumps

The changes to these specifications are corrections of typographical errors.

1. No. The following typographical errors are proposed to be corrected:

   - 4.4.1.1.2.5 : "stop" should be "stops".

   - 4.4.1.1.2.6 : "operable loop" should be "inoperable loop".

   - 4.4.1.1.2.6b : This surveillance is being restored after being inadvertently deleted in a previously issued amendment.

   - 4.4.1.2, footnote "**" : The reference to Specification 4.4.1.1.2.9 should be 4.4.1.1.2.6, based on renumbering which occurred in a previously issued amendment.

   Due to their editorial nature, these changes are of no safety significance.

2. No. See 1. above.

3. No. See 1. above.

Specification 5.3.1. Fuel Assemblies

This section has been changed to describe the actual core configuration for U2C5, which includes one inert (i.e., solid zircaloy-2) rod.

1. No. The insert Rod was used to repair a fuel assembly that failed during U2C2. This repaired assembly was analyzed and found to be acceptable in support of U2C4 operation, which was approved by the NRC (See Reload Summary Report Reference 3). Based on the above, use of the repaired assembly does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. See 1. above.

3. No. See 1. above.

Specification 5.3.2. Control Rod Assemblies

The changes to this specification are provided in order to recognize the replacement blade design being introduced in U2C5.

1. No. The main differences between the replacement Duralife 160C control blades and the original equipment control blades are:

   a. the Duralife 160C control blades utilize three solid hafnium rods at each edge of the cruciform to replace the three B_4C rods that are most susceptible to cracking and to increase control blade life;

   b. the Duralife 160C control blades utilize improved B_4C tube material (i.e. high purity stainless steel vs. commercial purity stainless steel) to eliminate cracking in the remaining B_4C rods during the lifetime of the control blade;

   c. the Duralife 160C control blades utilize GE's crevice-free structure design, which includes additional B_4C tubes in place of the stiffeners, an increased sheath thickness, a full length weld to attach the handle and velocity limiter, and additional coolant holes at the top and bottom of the sheath;

   d. the Duralife 160C control blades utilize low cobalt-bearing pin and roller materials in place of stellite which was previously utilized;

   e. the Duralife 160C control blade handles are longer by approximately 3.1 inches in order to facilitate fuel moves within the reactor vessel during refueling outages at Susquehanna SES; and

   f. the Duralife 160C control blades are approximately 16 pounds heavier as a result of the design changes described above.

   The Duralife 160C control blade has been evaluated to assure it has adequate structural margin under loading due to handling, and normal, emergency, and faulted operating modes. The loads evaluated include those due to normal operating transients (scram and jogging), pressure differentials, thermal gradients, seismic deflection, irradiation growth, and all other lateral and vertical loads expected for each condition. The Duralife 160C control blade stresses, strains, and cumulative fatigue have been evaluated and result in an acceptable margin to safety. The control blade insertion capability has been evaluated and found to be acceptable during all modes of plant operation within the limits of plant analyses. The Duralife 160C control blade coupling mechanism is equivalent to the original equipment coupling mechanism, and is therefore fully compatible with the existing control rod drives in

the plant. In addition, the materials used in the Duralife 160C are compatible with the reactor environment. The impact of the increased weight of the control blades on the seismic and hydrodynamic load evaluation of the reactor vessel and internals has been evaluated and found to be negligible.

With the exception of the crevice-free structure and extended handle, the Duralife 160C control blades are equivalent to the NRC approved Hybrid I Control Blade Assembly (See Reload Summary Report Reference 10). The mechanical aspects of the crevice-free structure were approved by the NRC for all control blade designs in Reload Summary Report Reference 11. A neutronics evaluation of the crevice-free structure for the Duralife 160C design was performed by GE using the same methodology as was used for the Hybrid 1 control blades in Reload Summary Report Reference 10. These calculations were performed for the original equipment control blades and the Duralife 160C control blades described above assuming an infinite array of ANF 9X9 fuel. The Duralife 160C control blade has a slightly higher worth than the original equipment design, but the increase in worth is within criterion for nuclear interchangeability. The increase in blade worth has been taken into account in the appropriate U2C5 analyses. However, as stated in Reload Summary Report Reference 10, the current practice in the lattice physics methods is to model the original equipment all B_4C control blade as non-depleted. The effects of control blade depletion on core neutronics during a cycle are small and inherently taken into account by the generation of a target k-effective for each cycle. As discussed above, the neutronics calculations of the crevice-free structure show that the non-depleted Duralife 160C control blade has direct nuclear interchangeability with the non-depleted original equipment all B_4C design. The Duralife 160C also has the same end-of-life reactivity worth reduction limit as the all B_4C design. Therefore, the Duralife 160C can be used without changing the current lattice physics models as previously approved for the Hybrid I control blades (Reload Summary Report Reference 10).

The extended handle and the crevice-free structure features of the Duralife 160C control blades result in a one pound increase in the control blade weight over that of the Hybrid I blades, and a sixteen pound increase over the Susquehanna SES original equipment control blades. In Reload Summary Report Reference 10, the NRC approved the Hybrid I control blade which weighs less (by more than one pound) than the D lattice control blade. The basis of the Control Rod Drop Accident analysis continues to be conservative with respect to control rod drop speed since the Duralife 160C control blade weighs less than the D lattice control blade, and the heavier D lattice control rod drop speed is used in the analysis. In addition, GE performed scram time analyses and determined that the Duralife 160C control blade scram times are not significantly different than the original equipment control blade scram times. The current Susquehanna SES measured scram times also have

considerable margin to the Technical Specification limits. Since the increase in weight of the Duralife 160C control blades does not significantly increase the measured scram speeds and the safety analyses which involve reactor scrams utilize the Technical Specification limit scram times, the operating limits are applicable to U2C5 with Duralife 160C control blades.

Since the Duralife 160C control blades contain solid hafnium rods in locations where the B_4C tubes have failed, and the remaining B_4C rods are manufactured with an improved tubing material (high purity stainless steal vs. commercial purity stainless steel), boron loss due to cracking is not expected. Therefore, the requirements of IE Bulletin 79-26, Revision 1 do not apply to the Duralife 160C control blades. However, PP&L plans to continue tracking the depletion of each control blade and discharge any control blade prior to a ten percent loss in reactivity worth.

Based on the discussion above, the new control blades proposed to be utilized in U2C5 do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. No. The replacement blades can only be evaluated for their effectiveness as part of the overall reactivity control system, which is evaluated in terms of analytical consequences in 1. above. Since they do not cause any significant change in system operation or function, no new events are created.

3. No. The analyses described in 1. above indicate that the replacement blades meet all pertinent regulatory criteria for use in this application, and are expected to eliminate the boron loss concerns expressed in IE Bulletin 79-26, Revision 1. Therefore, the proposed change does not result in a significant decrease in any 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. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and should cite the publication date and page number of this FEDERAL REGISTER notice. Written comments may also be delivered to Room P-223, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland, from 7:30 a.m. to 4:15 p.m. C[illegible] of written comments received may be examined at the NRC Public Docume[illegible]e Gelman Building, 2120 L Street, N.W., Washington, D.C. The filing [illegible]s for hearing and petitions for leave to intervene is discussed below.

By April 18, 1991 , the licensee may file a request for a hearin[illegible] with respect to issuance of the amendment to the subject facility oper[illegible] 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 petition for leave to intervene. Request for a hearing and petitions 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, N.W., Washington, D.C. 20555 and at the Local Public Document Room located at the Osterhout Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania, 18701. 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 fifteen (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 fifteen (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 request for amendment involves no significant hazards consideration, the Commission may issue the amendment and make it effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

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

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 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.

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, D.C. 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C., by the above date. Where petitions are filed during the last ten (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) 325-6000 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Walter R. Butler, Director, Project Directorate I-2, Division of Reactor Projects I/II: (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, D.C. 20555, and to Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037, 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 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 March 7, 1991, which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. 20555 and at the Local Public Document Room located at Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Dated at Rockville, Maryland, this 18th day of March 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

James J. Raleigh

James J. Raleigh, Acting Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation