September 12, 1995

Mr. Robert G. Byram
Senior Vice President-Nuclear
Pennsylvania Power and Light Company
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2
(TAC NOS. M91663 and M91666)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 154 to Facility Operating License No. NPF-14 and Amendment No. 124 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments are in response to your letter dated February 2, 1995.

These amendments change the Technical Specifications for the two Susquehanna units to increase the licensed discharge fuel assembly for SPC 9X9-2 fuel from 40 to 45 GWD/MTU. This change is consistent with the Commissions approval of Topical Report PL-NF-94-005-P, "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES," documented in a letter to PP&L dated December 15, 1994.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,
original signed by
Chester Poslusny, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-387/50-388

Enclosures: 1. Amendment No. 154 to License No. NPF-14
2. Amendment No. 124 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Chester Poslusny, Senior Project Manager
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-387

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. NPF-14

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-14 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 154 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 12, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Appendix A Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
6-20b	6-20b

CORE OPERATING LIMITS REPORT (Continued)

9. XN-NF-84-97, Revision 0, "LOCA-Seismic Structural Response of an ENC 9x9 Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., December 1984.

10. PLA-2728, "Response to NRC Question : Seismic/LOCA Analysis of U2C2 Reload," Letter from H.W. Keiser (PP&L) to E. Adensam (NRC), September 25, 1986.

11. XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup Supplement 1 Extended Burnup Qualification of ENC 9x9 Fuel," May 1988.

12. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling Water Reactors : Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., March 1983.

13. XN-NF-524(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., November 1983.

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

16. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

17. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

18. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," September 1994.

19. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES", January, 1995.

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 124
License No. NPF-22

1. The Nuclear Regulatory Commission (the Commission or the NRC) having found that:

   A. The application for the amendment filed by the Pennsylvania Power & Light Company, dated February 2, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of the Facility Operating License No. NPF-22 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 124 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PP&L shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days after its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 12, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 124

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Appendix A Technical Specifications with enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
6-20b	6-20b

CORE OPERATING LIMITS REPORT (Continued)

14. XN-NF-512-P-A, Revision 1 and Supplement 1, Revision 1, "XN-3 Critical Power Correlation," October, 1982.

15. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

16. NE-092-001A, Revision 1, "Licensing Topical Report for Power Uprate With Increased Core Flow," Pennsylvania Power & Light Company, December 1992.

17. NRC SER on PP&L Power Uprate LTR (November 30, 1993).

18. PL-NF-90-001, Supplement 1-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Loss of Feedwater Heating Changes and Use of RETRAN MOD 5.1," September 1994.

19. PL-NF-94-005-P-A, "Technical Basis for SPC 9x9-2 Extended Fuel Exposure at Susquehanna SES", January, 1995.

6.9.3.3 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits and accident analysis limits) of the safety analysis are met.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.154TO FACILITY OPERATING LICENSE NO. NPF-14

AMENDMENT NO.124 TO FACILITY OPERATING LICENSE NO. NPF-22

PENNSYLVANIA POWER & LIGHT COMPANY

ALLEGHENY ELECTRIC COOPERATIVE, INC.

SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2

DOCKET NOS. 50-387 AND 388

1.0 INTRODUCTION

By letter dated February 2, 1995, the Pennsylvania Power and Light Company (the licensee) submitted a request for changes to the Susquehanna Steam Electric Station, Units 1 and 2, Technical Specifications (TS). The requested changes would add to the list of references in Technical Specification 6.9.3.2, the NRC approved PP&L Licensing Topical Report PL-NF-94-005-P-A, "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES," dated January 1995. This effectively would implement the Commission's approval to increase the licensed discharge fuel assembly exposure for Susquehanna Steam Electric Station, Units 1 and 2 from 40 gigawatt days per metric ton (GWD/MTU) of uranium to 45 GWD/MTU.

2.0 EVALUATION

Pennsylvania Power and Light Company (PP&L) (the licensee), on May 31, 1994, submitted to the Commission for review, Topical Report PL-NF-94-005-P, "Technical Basis for SPC 9X9-2 Extended Fuel Exposure at Susquehanna SES." This report provided a technical justification for the increased fuel burnup and the staff subsequently approved the use of this report on Susquehanna Steam Electric Station (SSES), Units 1 and 2, as indicated in its letter to PP&L dated December 15, 1994. As the staff discussed in that report, there were five criteria used to confirm the performance of the extended exposure demonstration assemblies. The inspection of the four assemblies in October, 1994 showed that all five criteria were met, as indicated below:

Extended Exposure Performance Criteria (46.848 GWD/MTU)	Inspection Results	Criteria Met?
Maximum rod oxide thickness is less than 3 mils (78 microns)	41 microns	yes
Fuel rod engaged in upper tie plate	Minimum rod engagement = 0.643 inch	yes
Fuel rod diameter and ovality consistent with SPC data base	Average creepdown = 0.4210 inch (0.70%) Average ovality = 0.0014 inch	yes
Fuel Channel engaged with lower tie plate seal	Channel engagement = 0.3 inch	yes
Rod-to-rod spacing shows no unusual gap closure	No unusual gap closure observed	yes

In the December 15, 1994, letter referenced above, the staff requested that PP&L review the applicable Final Safety Anaylsis Report (FSAR) design basis events in order to determine the impact of the higher fuel assembly exposure and Linear Heat Generation Rate (LHGR). PP&L indicated in its submittal that increasing the maximum fuel exposure from 40 to 45 GWD/MTU will have a small impact on the core design (i.e., increased enrichment, number, and placement of fuel assemblies). The licensee also stated that this small change in core design will in turn have a minimal impact on the design basis events.

For each reload cycle, analyses are performed by PP&L to assure that the new core configuration will meet the appropriate fuel-related safety limits. As stated in PL-NF-94-005-P-A, all of the SPC fuel design limits are met for the higher exposure and LHGR. The cycle-specific design basis events as described in the NRC approved PP&L Licensing Topical Report PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis" (July 1992), will be analyzed each cycle using the higher fuel exposure and LHGR. Small increases in fuel exposure and LHGR do not invalidate either the current approach for selection of the limiting events or the assumptions used in the analysis of specific limiting events.

As part of the SSES power uprate effort, General Electric (GE) analyzed the Loss of Coolant Accident (LOCA) with the higher fuel exposure and LHGR (NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis for Susquehanna Units 1&2," May 1992); therefore PP&L stated that reanalysis of the LOCA event is not required. The staff agrees with this position.

The licensee indicated in its submittal that design basis accidents which result in a radiological release (e.g., LOCA, main steamline break (MSLB), control rod drop accident (CRDA), and refueling accident) were evaluated in GE NEDC-32161P, "Power Uprate Engineering Report for Susquehanna Steam Electric Station Units 1 and 2," December 1993. The calculated fission product inventory used in these radiological release evaluations was based on a reference core in which all of the fuel is assumed to operate continuously for 3 years at 4.9 MW per bundle (1.09% of core average). GE stated that the

resulting source term is conservative for end-of-cycle core average exposures which are not substantially greater than 29 GWD/MTU. Since the introduction of the extended exposure fuel will not produce end-of-cycle core average exposures substantially greater than 29 GWD/MTU, the source term will remain valid.

The LOCA analysis used the above source term and hence, remains valid for extended exposure fuel. The MSLB analysis assumed coolant activities based on maximum allowable Technical Specification values (and no fuel failures are assumed as a result of the event). Thus, the MSLB analysis is valid for extended exposure fuel. The CRDA and refueling accidents used the above mentioned source term and assumed a 1.5 radial peaking factor. Since the above source term is valid for extended exposure fuel and a radial peaking factor of 1.5 is also conservative for core designs with extended exposure 9X9-2 fuel, the power uprate CRDA and refueling accident analyses also apply to extended exposure fuel.

As part of the NRC's concern on the fuel rod failure threshold for high burnup fuel, the CRDA was reevaluated for a failure threshold of 30 cal/gm (well below the current criterion of 170 cal/gm). The results of the reevaluation, which were approved by the NRC in support of PL-NF-94-005-P-A, show that the radiological releases will remain well within 10 CFR Part 100 limits.

The staff has evaluated the information discussed above and, in addition, reviewed a publication which was prepared for the NRC entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988. The NRC contractor, Pacific Northwest Laboratory (PNL) of Battelle Memorial Institute, examined the changes that could result in the NRC design basis accident (DBA) assumptions, described in the various appropriate SRP sections and/or Regulatory Guides (RG), that could result from the use of extended burnup fuel (up to 60 MWD/MTU). The staff agrees that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimated that I-131 fuel gap activity in the peak fuel rod with 60 MWD/MTU burnup could be as high as 12%. This value is approximately 20% higher that the value normally used by the staff in evaluating fuel handling accidents RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors").

For the fuel handling accident, PNL concluded that the use of RG 1.25 procedures for the calculation of accident doses for extended burnup fuel may be utilized. These procedures give conservative estimates for noble gas release fractions that are above calculated values for peak rod burnups of 60,000 MWD/MTU. Iodine-131 inventory, however, may be up to 20% higher than that predicted by RG 1.25 procedures.

In its evaluation for SSES Units 1 and 2 issued in April 1981 (NUREG-0776), the staff conservatively estimated offsite doses due to radionuclides released to the atmosphere from a fuel handling accident. The staff concluded that the plant mitigative features would reduce the doses for this DBA to below the doses specified in Standard Review Plan (SRP) Section 15.7.4.

Since the licensee intends to utilize extended burnup fuel, the staff reanalyzed the fuel handling DBA for this case. According to PNL increasing the fuel burnup rate to 5.0 weight percent U-235 with a maximum burnup of 60,000 MWD/MTU increases the doses for a fuel handling accident by a factor of 1.2. The licensee proposes to increase the fuel burnup rate from 40,000 to 45,000 MWD/MTU. The 1.2 factor increase in dose displayed in Table 1 below, bounds the dose consequences of the licensee's proposal. In Table 1, the new and old DBA doses are presented and compared to the guidelines doses in SRP Section 15.7.4 (established on the basis of 10 CFR Part 100).

Table 1

Radiological Consequences of Fuel Handling Design Basis Accident (rem)

	Thyroid	
	Exclusion Area	Low Population Zone
Staff Evaluation April 1981 (NUREG-0776)	12	<1
Bounding Estimates for Extended Burnup Fuel	14.4	<1.2
Regulatory Requirement (NUREG-0800 Section 15.7.4)	75	75

The staff concludes that the only potential increased doses resulting from the fuel handling accidents with extended burnup fuel is the thyroid doses; these doses remain well within the dose limits given in NUREG-0800 and are, therefore, acceptable.

2.1 Summary

Based on the information provided in the licensee's submittal and the staff's evaluation of the increase in potential doses, the staff finds that the proposed TS change and implementation of the approved increase in fuel burnup rate from 40 to 45 MWD/MTU to be acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on September 12, 1995 (60 FR 47402). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of this amendment will not have significant effect on the quality of the human environment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: C. Poslusny

Date: September 12, 1995