

July 6, 2001

Mr. Robert G. Byram
Senior Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
2 North Ninth Street
Allentown, PA 18101

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE
OF AMENDMENT RE: 1.4 PERCENT POWER UPRATE (TAC NOS. MB0444
AND MB0445)

Dear Mr. Byram:

The Commission has issued the enclosed Amendment No. 194 to Facility Operating License No. NPF-14 and Amendment No. 169 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 30, 2000, as supplemented by letters dated February 5, May 22, May 31, and June 26, 2001.

These amendments increase the licensed power level by approximately 1.4 percent from 3,441 megawatts thermal (MWt) to 3,489 MWt. The changes are anticipated to increase each unit's net electrical output by approximately 14 MWe. The request is based on the installation of the Caldon LEFM[✓]™ ultrasonic flow measurement system with its ability to achieve increased accuracy in measuring reactor feedwater flow.

In approving the enclosed amendments, the NRC staff has relied on the commitments included in your May 22, 2001, supplemental letter. With regard to your commitment to revise the pressure-temperature limit curves in the TS, the staff has found that reasonable controls for the implementation and subsequent evaluation of proposed changes pertaining to this commitment are best provided by your commitment management program. With regard to your commitment to modify the Unit 1 standby liquid control (SLC) system by the Spring 2002 refueling outage, the staff has found that prior NRC approval is required for subsequent changes to this commitment. Therefore, implementation of this amendment for Unit 1 has been conditioned on modification of the SLC system as described in your May 22, 2001, letter and the enclosed NRC staff safety evaluation.

A copy of the related Notice of Issuance for publication in the *Federal Register* is enclosed. Also enclosed are nonproprietary and proprietary versions of our related safety evaluation. The nonproprietary version of the safety evaluation will be placed in the Nuclear Regulatory

**NOTE: THIS DOCUMENT CONTAINS PROPRIETARY INFORMATION. THIS DOCUMENT
BECOMES NONPROPRIETARY UPON REMOVAL OF ENCLOSURE 4.**

R. Byram

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Commission public document room and added to the Agencywide Documents Access and Management Systems Publicly Available Records System (ADAMS PARS) Library. The Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/RA/

Robert G. Schaaf, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures: 1. Amendment No. 194 to
License No. NPF-14
2. Amendment No. 169 to
License No. NPF-22
3. Safety Evaluation (nonproprietary)
4. Safety Evaluation (proprietary)
5. Notice of Issuance

cc w/encls: See next page

**NOTE: THIS DOCUMENT CONTAINS PROPRIETARY INFORMATION. THIS DOCUMENT
BECOMES NONPROPRIETARY UPON REMOVAL OF ENCLOSURE 4.**

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PPL SUSQUEHANNA, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC.
DOCKET NO. 50-387
SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 1
AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 194
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 PPL Susquehanna, LLC, dated October 30, 2000, as supplemented by letters dated February 5, May 22, May 31, and June 26, 2001, 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 Facility Operating License and 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. 194 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC 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 shall be implemented upon startup following the Unit 1 12th Refueling and Inspection Outage, currently scheduled for Spring 2002. Implementation shall include modification of the standby liquid control system as described in PPL Susquehanna, LLC's, May 22, 2001, letter and the NRC staff's safety evaluation dated July 6, 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: July 6, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 194

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

INSERT

Facility Operating License:

Page 3

Page 3

Technical Specifications:

1.1-6
5.0-21
5.0-22
5.0-23
5.0-24

1.1-6
5.0-21
5.0-22
5.0-23
5.0-24

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
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 PPL Susquehanna, LLC, dated October 30, 2000, as supplemented by letters dated February 5, May 22, May 31, and June 26, 2001, 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 Facility Operating License and 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. 169 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. PPL Susquehanna, LLC 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 shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachment: Changes to the Facility
Operating License and
Technical Specifications

Date of Issuance: July 6, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE

INSERT

Facility Operating License:

Page 3

Page 3

Technical Specifications:

1.1-6
5.0-21
5.0-22
5.0-23
5.0-24

1.1-6
5.0-21
5.0-22
5.0-23
5.0-24

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 194 TO FACILITY OPERATING LICENSE NO. NPF-14
AND AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. NPF-22
PPL SUSQUEHANNA, LLC
ALLEGHENY ELECTRIC COOPERATIVE, INC.
SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2
DOCKET NOS. 50-387 AND 50-388

1.0 INTRODUCTION

By letter dated October 30, 2000, as supplemented February 5, May 22, May 31, and June 26, 2001, PPL Susquehanna, LLC (PPL or the licensee), submitted a request for changes to the Susquehanna Steam Electric Station (SSES), Units 1 and 2, Facility Operating Licenses (FOLs) and Technical Specifications (TSs). The requested changes would increase the licensed power level for each unit by approximately 1.4 percent from 3441 megawatts thermal (MWt) to 3489 MWt. This power uprate request is based on the Caldon, Inc., Engineering Report (ER) 80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM[✓]™ System," Revision 0 (Reference 1); its supplement, ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM[✓]™ System" (Reference 2); and PPL licensing topical report (LTR) NE-2000-001P, "Power Uprate Resulting from Increased Feedwater Measurement Accuracy," Revision 1 (Reference 3). The supplemental letters contained clarifying information and did not expand the scope of the original *Federal Register* notice.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specific core thermal power level. The power level is indicated in the control room by neutron flux instrumentation that is calibrated to correspond to core thermal power. Core thermal power is validated by a nuclear steam supply system (NSSS) energy balance calculation. The accuracy of this calculation depends primarily upon the accuracy of feedwater flow, temperature, and pressure measurements.

The thermal power levels assumed in a plant's design-basis transient and accident analyses must bound the potential range of power levels at which the plant could be operated. The

NOTE: Redaction of proprietary information is denoted by brackets (e.g., []).

uncertainty of calculating values of core thermal power is factored into the allowable thermal power levels to reduce the likelihood of exceeding the power levels assumed in the analyses. At one time, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix K, required licensees to base their transient and accident analyses on an assumed power level of at least 102 percent of the licensed thermal power level. This was to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties). The 2 percent power margin uncertainty value was intended to address uncertainties related to heat sources in addition to instrument measurement uncertainties. The U.S. Nuclear Regulatory Commission (NRC) concluded, at the time of the original emergency core cooling system (ECCS) rulemaking, that the 2 percent power margin requirement was based solely on considerations associated with power measurement uncertainty as is reflected in Appendix K to 10 CFR Part 50.

Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2 percent margin, notwithstanding that the instruments used to calibrate the neutron flux instrumentation may be more accurate than originally assumed in the ECCS rulemaking. On June 1, 2000, the NRC published a final rule in the *Federal Register* (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty by using more accurate instrumentation to calculate the reactor thermal power and thereby calibrate the neutron flux instrumentation. Another objective of the final rule was to avoid unnecessary exemption requests by eliminating the need for licensees to obtain exemptions.

SSES, Units 1 and 2, were originally licensed to operate at a rated thermal power (RTP) of 3293 MWt. Amendment No. 143 to the Unit 1 FOL, issued on March 22, 1995, authorized a power uprate for Unit 1 to 3441 MWt. Amendment No. 103 to the Unit 2 FOL, issued on April 11, 1994, authorized a power uprate for Unit 2 to 3441 MWt. These uprates were based on the licensee's topical report NE-092-001 (Reference 4), which was approved by the NRC staff in its safety evaluation (SE) dated November 30, 1993 (Reference 5). The licensing basis analyses performed for the previous power uprate included a 2 percent power margin uncertainty, as required at that time by Appendix K to 10 CFR Part 50, such that analyses performed for the previous power uprate were performed at a power level of 3510 MWt.

In its current application, PPL requested approval to increase the SSES units' licensed thermal power levels to 3489 MWt based on the installation of the Caldon LEFM[✓]™ measurement system. The Caldon system is designed to improve the accuracy of feedwater flow rate measurement, which is used, in part, to calculate reactor thermal power. The improved flow measurement instrumentation would allow PPL to operate the SSES units with a reduced margin between the actual power level and the previously analyzed power level used in the licensing basis ECCS analyses of 3510 MWt.

3.0 EVALUATION

The NRC staff's review of PPL's 1.4 percent power uprate license amendment application is presented in the following subsections:

- 3.1 Reactor Core and Fuel Performance
- 3.2 Reactor Coolant System and Connected Systems
- 3.3 Engineered Safety Features

- 3.4 Instrumentation and Control
- 3.5 Electrical Power and Auxiliary Systems
- 3.6 Power Conversion Systems
- 3.7 Radwaste Systems and Radiation Sources
- 3.8 Reactor Safety Performance Features
- 3.9 General Issues
- 3.10 Facility Operating License (FOL) and TS Changes

3.1 Reactor Core

3.1.1 Thermal-hydraulic Design and Fuel Performance

The core thermal-hydraulic design and fuel performance characteristics are evaluated for each fuel cycle in accordance with NRC-approved Siemens' design criteria, analytical models and methods described in ANF-91-048(P)(A) (Reference 6).

The following sections describe the effects of the power uprate on the fuel design performance, thermal limits, the power/flow map, and reactor stability. The NRC staff's evaluation of these effects is considered as part of its evaluation of the ECCS design, ECCS performance, and reactor transients, discussed in Sections 3.3.2, 3.3.3, and 3.8.1, respectively, of this SE.

3.1.1.1 Fuel Design and Operation

Fuel bundles are designed to ensure that, (1) the fuel bundles are not damaged during normal steady state operation and anticipated operational occurrences (AOOs), (2) the damages to the fuel bundles are not so severe as to prevent control rod insertion when required, (3) the number of fuel rod failures during accidents are not underestimated, and (4) the coolability of the core is always maintained. For each fuel vendor, NRC-approved fuel design acceptance criteria and analysis methodologies that assure the fuel bundles comply with the objectives of Sections 4.2 and 4.3 of the Standard Review Plan (SRP) (Reference 7) and the applicable general design criteria (GDC) of 10 CFR Part 50, Appendix A. The fuel vendors perform thermal-mechanical, thermal-hydraulic, neutronic, and material analyses to ensure that the fuel system design can meet the fuel design limits during steady state, AOO, or accident conditions.

Because the uprated SSES Units 1 and 2 cores will consist exclusively of the Siemens ATRIUM-10™ fuel bundles, the fuel design criteria are based on the NRC-approved methodology described in ANF-89-98(P)(A) (Reference 8). The licensee stated that a new mechanical fuel design is not needed to achieve the 1.4 percent power uprate, even though new fuel designs may be used in the future to obtain additional operating flexibility or to maintain the fuel cycle length. The current ATRIUM-10™ fuel meets the NRC-approved acceptance criteria and any new fuel designs that do not comply with the NRC-approved fuel design criteria given in ANF-89-98(P)(A) will require NRC review and approval.

The slightly higher operating power and the increased steam void content will affect the core and fuel performance. Moreover, the licensee may change the power distribution in the reload design to achieve more operating flexibility or to maintain the fuel cycle length. This would also affect the core and fuel performance. However, the steady state and transient design linear heat generation rate limits for each fuel bundle ensure that the fuel plastic strain design limit or the fuel centerline melt limit will not be exceeded. The thermal-hydraulic design and the

operating limits will also ensure that the probability of boiling transition fuel failures will not increase at the uprated conditions. Fuel burnup may increase for the power uprate operation; however, the licensee cannot exceed the NRC-approved limit for the Siemens Power Corporation (SPC) boiling-water reactor (BWR) fuel. In the April 20, 1995, SE approving ANF-89-98(P), the NRC staff stated the maximum approved burnup for SPC BWR fuel cannot exceed the 60,000 MWD/MTU peak pellet limit without NRC review and approval.

The uprated reload core design and analyses will take into account the impact of the power uprate on the core thermal-hydraulics and fuel performance, and will also establish the fuel design limits.

3.1.1.2 Thermal Limits Assessment

GDC 10 of 10 CFR Part 50, Appendix A, requires that the reactor core and the associated control and instrumentation systems be designed with appropriate margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation, including AOOs. Operating limits are established to assure that regulatory and/or safety limits are not exceeded for a range of postulated events (transients and accidents). The safety limit minimum critical power ratio (SLMCPR) protects 99.9 percent of the fuel rods from boiling transition during steady state operation. The operating limit minimum critical power ratio (OLMCPR) assures that the SLMCPR will not be exceeded as result of an AOO. The operating linear heat generation rate (LHGR) is the core operating limit that assures the fuel thermal-mechanical performance limit (i.e. the 1 percent fuel plastic strain design limit or the no-fuel-centerline melt limit) will not be exceeded as a result of an AOO.

The SLMCPR is calculated for every reload at the rated thermal power throughout the cycle using NRC-approved methodologies (References 6, 8, 9, 10, 11, and 12). In a May 22, 2001, supplement, the licensee provided the draft Final Safety Analysis Report (FSAR) Chapter 15 update, which included the uprated reload analysis for Unit 2, Cycle 11 (U2C11). The supplement discussed the effect of the 1.4 percent power uprate and the transition from a mixed core with resident FRA-ANP 9X9-2 fuel to all ATRIUM-10™ fuel on the SLMCPR for U2C11 and Unit 1, Cycle 13 (U1C13). According to the licensee, the uprated core design slightly increased the SLMCPR. In the previous cycle, the resident fuel consisted of high-exposure/low-power assemblies, which do not contribute to the 0.1 percent of the total number of pins that are predicted to be in boiling transition. Therefore, PPL stated that the transition from a mixed core to exclusively ATRIUM-10™ fuel does not affect the calculated SLMCPR. The licensee pointed out that any increase in the core power flattens the radial power distribution so that more bundles have a peaking factor close to the maximum boiling transition. PPL added that since power increases by 1.4 percent, the effect on core power distribution is more limited, resulting in only a slight change in the SLMCPR. Thus, the licensee has calculated the SLMCPR for Unit 2 at the uprated condition, ensuring that 99 percent of the fuel pins in the core will not experience boiling transition. The licensee plans to uprate Unit 1 in the spring of 2002 (Cycle 13), at which time a similar reload analysis will establish SLMCPR values that will provide equivalent protection against boiling transition during rated and other than rated operating conditions throughout the cycle.

The licensee stated that the OLMCPR is determined on a cycle-specific basis from the results of the reload transient analysis and this approach will not change. The FSAR updates, in the May 22 supplement, also contained the U2C11 results for limiting transients. AOOs are

analyzed at various points in the allowable operating domain, depending on the type of transient. The licensee has analyzed the power-dependent transients at 100.6 percent of the uprated thermal power. The change in the minimum critical power ratio (MCPR) is combined with the SLMCPR to establish the OLMCPR, which ensures that 99 percent of the rods will not reach boiling transition in the event of an anticipated transient. The licensee has established the OLMCPR at the uprated condition for Unit 2 and will establish the OLMCPR for Unit 1 in the next Unit 1 reload analysis.

The steady-state and transient LHGR limits are established for every fuel design to protect against fuel centerline melt throughout the operating cycle. The licensee will determine the LHGR limits for the uprated cycle in the reload analysis, and these limits will be maintained during operation.

The maximum planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting loss-of-coolant accident (LOCA) and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46. For every new fuel type, the licensee performs LOCA analyses to confirm compliance with the LOCA acceptance criteria, and for every reload, the licensee confirms that the MAPLHGR operating limit for each reload fuel bundle design remains applicable.

Thus, the licensee has calculated the OLMCPR, the SLMCPR, the LHGR, and the MAPLHGR for the uprated conditions for both cycles using NRC-approved methodologies. The licensee will specify these limits in the TSs and/or the core operating limits report (COLR).

3.1.1.3 Reactivity Characteristics

The licensee stated that operation at higher power could reduce the excess reactivity. According to PPL, the loss of reactivity may affect the ability to manage the power distribution needed to meet the target power through the cycle, but the uprated core can be designed with sufficient excess reactivity to maintain the cycle length. The increase in the hot reactivity may result in higher hot-to-cold reactivity difference, reducing the shutdown margin. The licensee stated that the uprated core design will account for the loss of margin; if necessary a bundle design with improved shutdown margin characteristics can be used for future cycle. The licensee added that the reload analysis will ensure that the minimum shutdown margin requirements are met for each core design.

3.1.1.4 Power/Flow Operating Map

The licensee stated that the power uprate will not increase the licensed maximum core flow or the operating domain of the power/flow map, but the associated control and protective systems, which are based on percent power and percent flow, will be rescaled to the uprated thermal power. SSES Units 1 and 2 are licensed to operate with an increased core flow of $108 \times 10^6 \text{ lb}_m/\text{hr}$ and an extended load line limit analysis (ELLLA) region. The proposed power uprate will extend the ELLLA region to the 100 percent uprated power level. Thus, the power/flow map will have a smaller range of core flows at 100 percent uprated power.

3.1.1.5 Stability

PPL has installed an oscillation power range monitor (OPRM) to automatically detect and suppress instability, in accordance with NRC Bulletin 88-07 (Reference 13) and Supplement 1 to the bulletin (Reference 14). However, the OPRM system is not armed and is undergoing proof testing. PPL has implemented procedures that restrict plant operation in the high-power/low-flow region of the power/flow operating map. During a controlled or inadvertent entry into the instability region, specific operator actions prescribe clear instructions for operators in exiting the restricted region. The licensee has determined that neither the restricted region nor the required operator actions will be affected by the proposed power uprate. Therefore, the 1.4 percent power uprate will not significantly affect the licensee's capability to detect and suppress instabilities.

3.1.2 Control Rod Drives (CRDs) and CRD Hydraulic System

The CRD system controls gross changes in core reactivity by positioning neutron-absorbing control rods within the reactor. The CRD system is also required to scram the reactor by rapidly inserting withdrawn rods into the core. The licensee stated that the scram and rod insertion/withdrawal functions of the CRD system depend on [

] The licensee added that since the steam dome pressure remains the same as in the previous power uprate evaluation, [

] PPL, therefore, determined that the CRD system is capable of performing its design functions of rapid rod insertion (scram) and rod positioning (insertion/withdrawal).

The NRC staff concludes that the proposed power uprate will not have a significant impact on the CRD system for the following reasons:

1. The design operating dome pressure will not change, and []
2. The proposed power uprate may minimally affect the scram timing during transient overpressure conditions, but after the initial delayed scram time, the reactor pressure will assist in the scram.
3. There must be a minimum pressure differential of 250 psid between the hydraulic control unit and the vessel bottom head for normal CRD insertions and withdrawals (FSAR Section 4.6.1.1.2.4.1). Because the design operating dome pressure will not increase, the power uprate will have little impact on the CRD pump capacity.

Therefore, the NRC staff finds that the CRD system will continue to perform all its safety-related functions at the proposed uprated conditions.

The licensee evaluated the structural integrity of the control rod drive mechanisms' (CRDMs) by comparing the proposed parameters shown in Tables 1-1 and 1-2 of NE-2000-001P to those in

the design-basis analysis. The licensee indicated that the input parameters used in the existing design-basis analysis remain bounding, and concluded that the CRDM will continue to perform its function and maintain its structural integrity under the proposed power uprate condition. The NRC staff finds that the existing maximum calculated stress and fatigue usage factor previously provided by the licensee in NE-092-001 have large margins compared to the allowable limits. Therefore, the NRC staff finds that the CRDM will continue to meet its design-basis and performance requirements for the proposed 1.4 percent power uprate.

3.2 Reactor Coolant System and Connected Systems

The NRC staff reviewed the effects of the power uprate on the structural and pressure boundary integrity of the NSSS and balance-of-plant (BOP) systems. The review focused on the effects of the power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components, and the BOP piping systems.

The proposed 1.4 percent power uprate will increase the RTP level from 3441 MWt to 3489 MWt. The maximum core flow rate (108×10^6 lb_m/hr) and reactor vessel dome design pressure of 1050 psia remain unchanged, the dome temperature increases from 550.2 °F to 550.35 °F (an increase of 0.15 °F) and the steam flow rate increases from 14.139×10^6 lb_m/hr to 14.370×10^6 lb_m/hr (an increase of approximately 1.6 percent).

3.2.1 Nuclear System Pressure Relief

The safety/relief valves (SRVs) provide overpressure protection for the NSSS during abnormal operational transients. The licensee stated that the change in steam flow associated with the 1.4 percent power uprate will be accomplished by opening the turbine control valves slightly. This will result in a slight increase in the operating steam dome pressure from 1048 psia to 1049 psia, which is within the analytical steam dome pressure of 1050 psia. No changes to the SRV setpoints contained in the TSs have been proposed in support of the proposed power uprate.

Topical Report NE-2000-001P Tables 1-1 and 1-2 provide the reactor heat balance parameters for the current and the proposed uprated conditions. The tables show that for a core flow of 108×10^6 lb_m/hr, the steam flow rate increases by 1.6 percent for the uprated conditions. Considering that the steam flow will increase by 1.6 percent, that the SRVs will actuate at the current setpoints, and that the current American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* overpressure protection analysis is based on 102 percent of the currently licensed power, the NRC staff finds acceptable the licensee's assessment that the SRVs will have sufficient capacity to handle the increased steam flow associated with the proposed uprate.

3.2.2 Reactor Vessel Overpressure Protection

The ASME Code allowable peak pressure for the reactor vessel is 1375 psig (110 percent of the design pressure of 1250 psig), which is the acceptance limit for pressurization events. The licensee analyzes AOOs that may result in the largest overpressure transient on a cycle-specific basis, taking into account the power uncertainty. The most limiting overpressure transient event for SSES, Units 1 and 2, is the main steam isolation valves closure (MSIVC)

with failure of the valve position scram. The licensee has analyzed the MSIVC at 3510 Mwt (100.6 percent of the uprated power), 108×10^6 lb_m/hr core flow, and a steam dome pressure of 1050 psia. The number of SRVs assumed out of service in the analysis is consistent with the number specified in the TSs. The licensee determined that the increased operating power produces higher vessel peak pressure, but the peak pressure remains below the ASME Code limit of 1375 psig. The NRC staff has reviewed the Unit 2 Cycle 10 ASME overpressure analysis results in the draft FSAR updates, and finds the licensee's analyses acceptable based on the peak pressure remaining below the ASME Code limit.

3.2.3 Reactor Vessel and Internal Components

In assessing the impact of the proposed 1.4 percent power uprate on the reactor pressure vessel and its internal components, the licensee relied primarily on the review that was performed for its 1992 request for a 4.5 percent power uprate for SSES, Units 1 and 2 (topical report NE-092-001, Reference 5). In its current submittal, the licensee stated that the comprehensive review in topical report NE-092-001 remains valid, since the previous uprate evaluated the RPV and reactor vessel internals at 3510 MWt, which bounds the conditions resulting from the newly proposed increase in RTP. New pressure and temperature (P-T) limit curves were generated in 1992 based on conservative fluence values. The licensee stated that the current P-T limit curves for Units 1 and 2 remain unchanged, and are valid for 32 effective full-power years (EFPY). The NRC staff granted approval of the 1992 topical report by letter dated November 30, 1993.

The very low copper (Cu) and nickel (Ni) content of the reactor vessel beltline materials (e.g., 0.04 percent Cu and 0.990 percent Ni for a typical reactor vessel beltline weld), as documented in the licensee's letter of July 8, 1992 (Reference 15), results in low reactor vessel embrittlement. As indicated in FSAR Tables 5.3-4a and 5.3-4b, SSES Units 1 and 2 are not limited by the RPV beltline materials. The limiting materials are non-beltline (i.e., nozzles).

The NRC staff has reviewed the information provided by the licensee and determined that the P-T limit curves and upper shelf energy (USE) analyses for each unit are acceptable based on the analyses meeting the requirements of Appendix G to 10 CFR Part 50. In evaluating the effect of the power uprate on the shift in adjusted reference temperature (ART) and the need for new P-T limit curves, the staff applied the methodology for evaluating radiation embrittlement, found in Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," and evaluated the beltline and non-beltline curves for 32 EFPY. The SSES Unit 1 beltline curves were based on the highest ART in the beltline, which is 65.6 °F for lower intermediate shell plate 22-3. The SSES Unit 2 beltline curves were based on the highest ART in the beltline, which is 58.3 °F for lower intermediate shell plate 22-1. The current beltline and non-beltline P-T limit curves for both units remain bounding. However, the NRC staff has not completed its technical review and approval of the methodology used to derive the fluence values used in the proposed licensing action. The staff must complete its review and approval of this methodology in order to justify applying the fluence values for a full 32 EFPY. As an interim solution, the licensee committed to submit an amendment request by August 30, 2001, to limit application of the P-T curves to two operating cycles.

The NRC staff finds that reasonable controls for the implementation and subsequent evaluation of proposed changes pertaining to the above regulatory commitment are best provided by the licensee's administrative processes, including its commitment management program. The NRC

staff concludes that the above regulatory commitment does not warrant the creation of a regulatory requirement (items requiring prior NRC approval of subsequent changes).

The P-T limit curves and USE evaluations meet the requirements of Appendix G to 10 CFR Part 50. In addition, the power uprate does not necessitate a change in the SSES, Units 1 and 2, 10 CFR Part 50 Appendix H RPV surveillance program. SSES Unit 2 is included in the Boiling Water Reactor Vessel and Internals Project (BWRVIP) integrated surveillance program (ISP), which is currently under review by the NRC staff. The BWRVIP ISP was developed in order to integrate surveillance data for the benefit of those BWR vessels that do not have unirradiated baseline data. The BWRs that are part of the ISP will be able to use baseline and irradiated surveillance data to measure changes in RPV material embrittlement.

The licensee evaluated the reactor vessel and internal components considering the effects of changes in the design input parameters shown in NE-2000-001P Tables 1-1 and 1-2, as well as the applicable loads due to the proposed 1.4 percent power uprate.

The licensee indicated that the proposed power uprate will not change the design reactor steam dome pressure or core flow. There is no change in SRV, fuel lift, or seismic loads due to the uprate. LOCA loads for the normal operating, upset, emergency, and faulted conditions were evaluated previously based on the maximum core flow rate of 108×10^6 lb_m/hr and thus, remain bounding for the proposed 1.4 percent power uprate. The licensee concluded that the design-basis stresses and fatigue usage factors for the reactor vessel and internal components will remain unchanged for the proposed 1.4 percent power uprate. The NRC staff finds that the calculated stresses and cumulative usage factors provided in the previous power uprate (NE-092-001) have sufficient margin to accommodate the change in the reactor vessel temperature due to the proposed power uprate. Therefore, the reactor vessel and internal components will continue to be within the Code allowable limits and are acceptable.

The licensee assessed the flow-induced vibration for the proposed power uprate. The licensee indicated that the maximum core flow and maximum recirculation pump speed will remain unchanged following the proposed power uprate. The licensee also indicated that the potential for the flow-induced vibration in the reactor vessel internals has been evaluated at the maximum design condition, which remains bounding, and concluded that the flow-induced vibration level will remain within the design limits for the proposed 1.4 percent power uprate. The NRC staff accepts the licensee's conclusion.

3.2.4 Reactor Recirculation System

The power uprate will be accomplished by operating along extensions of rod and core flow lines on the power/flow map. SSES is currently licensed to operate at up to a maximum core flow of 108×10^6 lb_m/hr. The power uprate does not require an increase in the maximum allowable core flow. Therefore, the reactor recirculation flow will be maintained according to the existing power/flow map, with 100 percent power corresponding to the uprated power level. The cycle-specific reload analysis will consider the full range of the power and flow operating region.

The licensee stated that the recirculation inlet flow temperature [] which has negligible effect on the available net positive suction head (NPSH) available for the recirculation pumps. [

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The licensee concluded that since the reactor pressure and core flow rates used for the recirculation evaluation do not change as a result of the power uprate, the recirculation pump drive flow stops' setpoints and the power required by the recirculation pump motors do not change. Based on the above, the NRC staff concludes that the changes associated with the 1.4 percent power uprate will have an insignificant impact on the function of the recirculation system and, therefore, the system will continue to perform its design-basis functions at the uprated power level.

3.2.5 Reactor Coolant and BOP Piping

The licensee evaluated the reactor coolant and the BOP piping systems by comparing the system design input parameters shown in NE-2000-001P Tables 1-1 and 1-2, to those in the existing design-basis analysis. The licensee indicated that because the system pressure and temperature are not changed as a result of the increased operating power level, the effects of the power uprate are confined to the piping systems that experience an increased flow rate as a result of the proposed power uprate. However, the licensee stated in its letter dated May 31, 2001, that the increase in feedwater and main steamline flow is bounded by the current analysis. The licensee concluded that the 1.4 percent power increase does not affect the existing design-basis analysis for the reactor coolant pressure boundary and the BOP piping systems. The NRC staff agrees with the licensee's conclusion that the piping, components, and their supports at SSES, Units 1 and 2, will continue to meet the requirements of the code of record following the proposed 1.4 percent power uprate.

The licensee's evaluation of the reactor coolant piping confirmed that changes in the flow parameter associated with the power uprate would have no significant effects on the potential for flow-induced erosion/corrosion in those systems that might be susceptible to the phenomenon (e.g., feedwater or main steam systems). The NRC staff has reviewed the licensee's evaluations regarding the effect of the power uprate on the reactor coolant and BOP piping, and concludes that the licensee has bounded the effects of the power uprate. The proposed power uprate will not cause an adverse increase in erosion/corrosion, and no change to the SSES, Units 1 and 2, erosion/corrosion program is necessary.

3.2.6 Main Steam Isolation Valves

The MSIVs are engineered safety features (ESFs) for the reactor coolant pressure boundary. Within the TS-defined time limit (usually 3 to 5 sec), the MSIVs close to isolate the reactor vessel during postulated transient and accident conditions.

The licensee stated that the MSIV design pressure (1250 psig), temperature (575 °F) and flow (3.72×10^6 lb_m/hr) bound the maximum uprated operating conditions (1050 psia, 550 °F and 3.6×10^6 lb_m/hr). Therefore, the design conditions for the MSIVs bound the expected proposed operating conditions. The NRC staff agrees that the operating changes due to the power uprate will have little effect on the closure function of the MSIVs because the uprated steady state operating conditions are bounded by the MSIV normal design conditions. The 1.6 percent increase in steam flow may slightly increase the pressure drop across the MSIVs. However, the MSIVs are designed to close against a much higher pressure differential at a higher steam flow.

In addition, various TS surveillances require routine monitoring of MSIV closure time and leakage to ensure that the original licensing basis for the MSIVs is preserved. Based on the above, the NRC staff concludes that the MSIVs will continue to perform their design-basis function at the uprated power level.

3.2.7 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for starting a low-pressure core cooling system. According to FSAR Section 5.4.6, the RCIC system is designed to provide rated flow over a range of reactor pressures from approximately 165 psia up to 1201 psia. As noted in FSAR Section 15.2.7, the loss-of-feedwater flow transient assumes that the RCIC system will maintain sufficient water level inside the core shroud to ensure that the top of the active fuel will be covered throughout the event. The transient analysis also assumes that the low-setpoint SRVs would remove the stored and decay heat because MSIV closure on low water level isolates the reactor from the main condenser. The transient is a power-dependent transient and is more severe at a higher initial power, because there is more stored and decay heat to be dissipated and the water level drops faster.

In the generic 5 percent uprate topical (LTR1, Reference 16), [

] In
the 4.5 percent power uprate, the licensee evaluated the RCIC capability to provide core flow to the reactor vessel during a loss of feedwater flow event. In its June 26, 2001, supplement, the licensee stated that RCIC is designed to provide [] and, at [] the RCIC evaluation demonstrates that the system can provide sufficient water inside the core shroud to ensure that the top of the active fuel will be covered in the event of loss-of-feedwater flow.

Because the proposed 1.4 percent power uprate does not increase the design operating steam dome pressure or the SRV actuation setpoints and the RCIC injection capability was evaluated at [] the NRC staff concludes that RCIC performance will not be adversely affected by the proposed power uprate.

3.2.8 Residual Heat Removal System

The residual heat removal system (RHR) is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal after reactor

shutdown for both normal and post-accident conditions. The RHR system is designed to operate in low pressure coolant injection (LPCI) mode, shutdown cooling (SDC) mode, suppression pool cooling mode, and containment spray cooling (CSC) mode. The effects of the power uprate on the SDC, suppression pool cooling, and CSC modes are discussed in the following subsections. The LPCI mode of operation is discussed in Section 3.3.2.

3.2.8.1 Shutdown Cooling Mode

As stated in FSAR Section 5.4.7.1.1.1, the functional design-basis of the SDC mode is to reduce the reactor coolant temperature after reactor shutdown to 125 °F in approximately 20 hours using two RHR loops. Licensing topical report NE-092-001 evaluated the capability of the SDC mode of operation at an initial thermal power of 3510 Mwt. The SDC evaluation was performed to support the previous 4.5 percent power uprate and 108×10^6 lb_m/hr increased core flow operation. Therefore, evaluation of the SDC mode for the 1.4 percent power uprate is not necessary because the evaluation at the higher power (3510 Mwt) remains bounding and applicable.

The PPL licensing report NE-092-001 stated that the decay heat increases proportionally to the operating reactor power level; therefore, at the uprated power, the time required to reduce the reactor temperature to the shutdown conditions increases slightly. RG 1.139, "Guidance for Residual Heat Removal," provides an alternative approach to demonstrate SDC capability: the RHR system can reduce the reactor coolant temperature to 200 °F within 36 hours. The previous 4.5 percent uprate licensing report stated that the SDC capability evaluation, based on the RG 1.139 requirement, demonstrated that SSES can be brought to cold shutdown (200 °F) in 28 hours. Because the 4.5 percent power uprate SDC evaluation was performed at 102 percent of the current rated thermal power and the evaluation demonstrated that SSES can be brought to cold shutdown in 28 hours, the NRC staff finds that the previous shutdown cooling evaluation remains bounding.

3.2.8.2 Suppression Pool and Containment Spray Cooling Modes

The power uprate slightly increases the heat input to the suppression pool during a LOCA, which results in a negligible increase in peak suppression pool temperature. However, the proposed increase in core thermal power operation will have no effect on the CSC function of the RHR system. In its SE dated November 30, 1993, the NRC staff approved the licensee's analysis of suppression pool cooling and CSC modes of RHR operation at a reactor power level of 3510 MWt. This analysis is still valid, and bounds the proposed power uprate to 3489 MWt. Therefore, the NRC staff finds that operation of the RHR in suppression pool cooling and CSC modes at the 1.4 percent increase in thermal power level is acceptable.

3.3 Engineered Safety Features

3.3.1 Containment System

Primary containment temperature and pressure response following a postulated LOCA is important in determining the potential for offsite release of radioactive material, ECCS pump NPSH requirements, and environmental qualification requirements for safety-related equipment located inside the primary containment. Short- and long-term containment analyses for conditions following a large break inside the drywell are presented in the FSAR. The short-term

analysis is directed primarily at determining the peak drywell pressure responses during the initial blowdown of the reactor vessel inventory to the containment following a design-basis accident (DBA). The long-term analysis is directed primarily at determining the peak suppression pool temperature response.

FSAR Section 6.2 indicates that containment analyses were performed at 102 percent of the current licensed power level (3510 MWt) and an initial reactor pressure of 1050 psia. In its SE dated November 30, 1993, the NRC staff approved the licensee's analysis of the containment systems at the above uprated thermal power conditions, which bound the proposed RTP increase to 3489 MWt. Therefore, the NRC staff finds that operation of the containment systems at the proposed increase in RTP is acceptable.

3.3.2 Emergency Core Cooling Systems (ECCS)

The ECCS is designed to provide protection in the event of a LOCA due to a rupture of the primary system piping. Although DBAs are not expected to occur during the lifetime of a plant, plants are designed and analyzed to ensure that the radiological dose from a DBA will not exceed the 10 CFR Part 100 limits. For a LOCA, 10 CFR 50.46 specifies design acceptance criteria based on (1) the peak cladding temperature, (2) local cladding oxidation, (3) total hydrogen generation, (4) coolable core geometry, and (5) long-term cooling. The LOCA analysis considers a spectrum of break sizes and locations, including a rapid circumferential rupture of the largest recirculation system piping. Assuming a single-failure of the ECCS, the LOCA analyses identify the break sizes that severely challenge the ECCS systems and the primary containment. The MAPLHGR operating limit is based on the most limiting LOCA analysis, and the licensees perform LOCA analyses for each new fuel type to demonstrate that the 10 CFR 50.46 acceptance criteria can be met.

The ECCS for SSES includes the high-pressure coolant injection system (HPCI), the LPCI mode of the RHR system, the low-pressure core spray (CS) system and the automatic depressurization system (ADS). The ADS system is discussed in Section 3.3.2.1, and the ECCS performance in Section 3.3.3, below.

SPC performed the LOCA analysis for SSES Units 1 and 2 at a design reactor vessel dome pressure of 1050 psia and a bounding power of 102 percent of the current rated thermal power (3510 MWt). Because these initial conditions do not change for the proposed 1.4 percent power uprate, the licensee stated that the current LOCA analysis remains applicable. According to the licensee, the LOCA analyses of record demonstrate that the HPCI system, the LPCI mode of RHR and the CS system have the capabilities to provide core cooling during LOCA. These capabilities do not change for operation at the uprated condition, therefore, the ECCS will continue to meet the ECCS-LOCA analysis assumptions and design criteria at the uprated condition.

Because the LOCA analysis is based on NRC-approved methodology and codes, and the assumed reactor vessel dome pressure and power are bounding, the NRC staff finds that the licensee's assessment that the ECCSs will perform as designed and analyzed at the uprated conditions is acceptable.

3.3.2.1 Automatic Depressurization System

As stated in FSAR Sections 6.3.1.2.4 and 7.3.1, the ADS uses the safety/relief valves to reduce reactor pressure following a small-break LOCA with HPCI failure, allowing LPCI and CS to provide cooling flow to the vessel. The plant design requires a minimum flow capacity for the SRVs and after a time delay, the ADS initiate either on low water level with high drywell pressure, or on low water level alone. FSAR Table 6.3-2 indicates that the LOCA analyses (for the spectrum of accidents requiring ADS actuation) were performed at a power level of 3510 MWt and a reactor vessel steam dome pressure of 1050 psia. The NRC staff concludes that the current power uprate does not affect the capability of the ADS to perform its design function because the system will continue to operate within the previously accepted design limits.

3.3.3 Emergency Core Cooling System Performance

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and the analysis models must satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K.

As indicated in FSAR Section 6.3.3, the ECCS performance is evaluated under all LOCA conditions using approved 10 CFR Part 50, Appendix K, models to demonstrate compliance with the requirements of 10 CFR 50.46. The results of the ECCS-LOCA analyses for GE fuel were summarized in Table 1 of the previous SSES power uprate submittal (Reference 4). The results for Siemens ATRIUM-10™ fuel are provided in topical reports EMF-96-160(P) and EMF-96-161(P) (References 17 and 18, respectively).

Because the SSES Unit 1 and 2 cores for the power uprate will consist exclusively of ATRIUM-10™ fuel, the SPC ECCS-LOCA analysis will be the applicable analysis of record. The NRC staff finds acceptable PPL's ECCS performance evaluation because the analytical models and codes are based on NRC-approved methodology described in ANF-91-048(P)(A) and the ECCS-LOCA analyses are based on bounding power and flow conditions.

3.3.4 Standby Gas Treatment System

The standby gas treatment system (SGTS) is designed to ensure controlled and filtered release of particulates and halogens from primary and secondary containment to the environment during abnormal and accident conditions in order to maintain offsite doses within the limits specified by 10 CFR Part 100. The SGTS consists of two 100 percent capacity, parallel, redundant flow trains. Each train is sized to change one secondary containment (SC) air volume per day, while maintaining the SC at a slight negative pressure of 0.25-inch water gauge with respect to the outside atmosphere. Maintaining this negative pressure prevents unfiltered release of radioactive material from the SC to the environment. The licensee determined that the proposed slight increase in power (1.4 percent) will not impair the capability of the SGTS to meet this design objective, since the original analysis was completed at an initial power level of 3616 MWt, a value just over 103 percent of the proposed power level (3489 MWt). In addition, the previous evaluation demonstrated that the SGTS has a 500 percent excess capacity. Therefore, operation at the proposed RTP would have an insignificant impact on the ability of the SGTS to meet its intended design function.

On the basis of its review of the licensee's rationale and the experience gained from its review of power uprate applications for similar BWR plants, the NRC staff finds that operation at the proposed uprated power level does not change the design aspects or operation of the SGTS.

3.3.5 Other Engineered Safety Features Systems

3.3.5.1 Main Steam Isolation Valve Leakage Control System (MSIV-LCS)

The MSIV-LCS was designed to control any leakage of contaminated steam through redundant isolation valves on each main steamline from reactor to the turbine in the event of a LOCA. In its SE dated August 15, 1995 (Reference 19), the NRC staff approved the licensee's request to eliminate the MSIV-LCS from its TSs. The approved analysis takes advantage of the large volume in the main steamlines and main condenser to provide hold-up and plate-out of fission products that may leak from closed MSIVs. In addition, the alternative approach will continue to mitigate the consequences of an accident at the proposed power uprate conditions which could result in potential offsite exposures comparable to 10 CFR Part 100.

Based on the licensee's rationale, approval of the analysis (Reference 19), and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff finds that operation at the proposed uprated power level does not change the design and operational aspects of the alternative leakage path for the plants MSIVs; and will have an insignificant impact on the alternative MSIV leakage control method.

3.3.5.2 Post-LOCA Combustible Gas Control

The combustible gas control system is designed to control the hydrogen concentrations of the drywell and containment atmospheres below the flammability limit of 4.0 volume percent (v/o) following a LOCA. The system design is based on evolution of hydrogen from three sources, including (1) metal-water reaction of active fuel cladding, (2) corrosion of zinc and aluminum exposed to water during the LOCA, and (3) radiolysis of water. As a result of the proposed power uprate, only the post-LOCA production of hydrogen by radiolysis will increase in proportion to the power. The licensee's analysis basis used in the determination of hydrogen generated by radiolysis was a power level of 3510 MWt. The NRC staff approved this analysis (Reference 5), and the volume of hydrogen used for analysis purposes bounds the volume of hydrogen generated at the proposed uprated conditions. Therefore, the NRC staff finds that post-LOCA combustible gas control at the uprated power level is acceptable.

3.4 Instrumentation and Control

3.4.1 Nuclear Steam Supply System (NSSS) I&C

Increases in core thermal power and steam flow affect some of the instrument setpoints, and these setpoints may need to be adjusted. The setpoints are calculated based on NRC-approved methodologies and the setpoint adjustment must assure sufficient margin between the analytical limits and the system setting. The following section discusses the affect of the power uprate on instrument setpoints.

3.4.1.1 Neutron Monitoring

PPL stated that the averaged power range monitor (APRM) power signal will be rescaled to the uprated power, but the percentage setpoint will not change, i.e., indicated power will remain 100 percent of the uprated power. [

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In the May 22, 2001, draft FSAR update, the licensee provided the results of the RWE analysis at the uprated conditions for SSES Unit 2 Cycle 11. Similarly, the licensee has reanalyzed the limiting transients at the uprated conditions using rescaled analytical setpoints. Therefore, the staff finds the proposed changes to the neutron monitoring setpoints and rod block setpoints to be acceptable, because the reload analyses at the uprated conditions use the adjusted setpoints. The proposed setpoint changes are also consistent with the assessments in the NRC-approved power uprate generic topical reports (References 16 and 20).

3.4.2 Caldon LEFMTM Feedwater Flow Measurement System

Caldon Topical Report ER-80P (Reference 1) and its supplement, ER-160P (Reference 2), provide the generic basis for increasing power by as much as 1.4 percent. The staff approved these documents in SEs dated March 8, 1999 (Reference 21), and November 24, 2000 (Reference 22), respectively. The licensee's LTR NE-2000-001P (Reference 3) provides a plant-specific justification for the proposed 1.4 percent power uprate at SSES on the basis of the two Caldon topical reports. This SE addresses the licensee's plant-specific justification for the proposed SSES power uprate.

3.4.2.1 Background

As discussed in Section 2.0 of this SE, an accurate measurement of feedwater flow and temperature is necessary for calibrating nuclear instrumentation to represent core thermal power. In addition, the uncertainty of calculating core thermal power values determines the probability of exceeding the power levels assumed in the design-basis transient and accident analyses. In this regard, to allow for uncertainties in determining thermal power (e.g., instrument measurement uncertainties), Appendix K to 10 CFR Part 50 requires that LOCA and ECCS analyses must assume that the reactor has operated continuously at a power level that is at least 102 percent of the licensed thermal power. The 2 percent margin was intended to address uncertainties related to heat sources, as well as instrument measurement uncertainties. Later, the NRC concluded that, at the time of the original ECCS rulemaking, the 2 percent power margin requirement appeared to be based solely on considerations associated

with power measurement uncertainty. Appendix K to 10 CFR Part 50 did not require demonstration of the power measurement uncertainty and mandated a 2 percent margin, notwithstanding that the instruments may be more accurate than originally assumed in the ECCS rulemaking.

The Commission's final rule, published in the *Federal Register* (65 FR 34913) on June 1, 2000, gives licensees the option to justify and apply a reduced margin between the licensed power level and the assumed power level for ECCS evaluation, or to maintain the current power margin of 2 percent. Under the amended rule, licensees may use a reduced margin to gain benefits from operation at higher power, or to relax ECCS-related TSs (e.g., pump flows). Licensees can also realize another potential benefit in modifying fuel management strategies (e.g., possibly by altering core power peaking factors). However, the final rule, by itself, does not allow increases in licensed power levels. Because the licensed power level for a plant is specified in the FOL and TS, proposals to raise the licensed power level must be reviewed and approved under the license amendment process.

The instrumentation for measuring feedwater flow rate typically uses a venturi, an orifice plate, or a flow nozzle to generate a differential pressure that is proportional to the feedwater velocity in the pipe. The most common instrumentation for measuring flow rates is the venturi flow meter in the feedwater system piping, as is used at SSES. The major advantage of a venturi flow meter over the other two flow measurement designs is the relatively low head loss that is created as the feedwater passes through the device. The major disadvantage of the venturi flow meter is the effect of venturi fouling on flow meter instrument accuracy. Fouling causes a venturi flow meter to indicate higher differential pressures for equivalent flow velocities, thereby resulting in an output signal that represents a higher-than-actual flow rate. Because feedwater flow rate is directly proportional to calorimetric power, this error in feedwater flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher-than-actual core power.

Calibrating the nuclear instrumentation to indicate higher-than-actual core power is conservative with respect to reactor safety, but causes the licensee to generate electrical power proportionately lower than would be the case if the nuclear instrumentation was calibrated to indicate the actual core power. To eliminate the effects of venturi fouling, the venturi flow meter device must be removed, cleaned, and calibrated. The high cost of flow meter calibration and the need to improve flow instrumentation accuracy prompted the nuclear industry to assess other flow measurement techniques. Use of a leading edge flow meter (LEFM) implementing transit time technology was found to be a viable alternative.

The basis of using transit time technology to measure fluid velocity is that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than in the opposite direction. Consequently, the difference in the upstream and downstream traversing times is proportional to the velocity of the fluid in the pipe that has been traversed by the ultrasonic pulses. Additionally, the average of the upstream and downstream transit times is proportional to the average density of the traversed fluid, which is a function of the average fluid temperature and pressure.

The Caldon Chordal LEFM[✓]™ is a software-controlled digital system that consists of an electronic cabinet in the auxiliary instrument room and a measurement section, or a spool piece, permanently mounted in each of the feedwater pipes. The LEFM[✓]™ measures four line

integral velocities at precise locations with respect to the pipe center line. The system numerically integrates the four measured velocities to determine the average (bulk) feedwater flow rate and the bulk feedwater fluid temperature. The plant's computer then uses these processed measurements to determine the reactor thermal power.

The NRC staff's safety evaluation of the licensee's submittal is discussed in the following section.

3.4.2.2 Evaluation

Caldon Topical Report ER-80P and its supplement ER-160P (both previously approved by the NRC staff) describe the improved LEFM✓™ system for the measurement of feedwater flow and temperature to determine reactor thermal power and provide a basis for a 1.4 percent uprate of the licensed reactor power. The topical report stated that the LEFM✓™ is superior to the venturi-based instrumentation currently in use on the basis of the following:

1. The elements of LEFM✓™ accuracy can be verified on-line,
2. The LEFM✓™ measurement accuracy results in an uncertainty ± 0.6 percent of thermal power, with a 95 percent confidence limit, whereas the measurement uncertainty of the current venturi flow element instrumentation is ± 1.4 percent.

The licensee used an approved setpoint methodology to calculate the plant-specific total power measurement uncertainty of the LEFM✓™. The calculation was done with two standard deviations to determine, with a 95 percent confidence level (probability of operation within bounds), that the calculated power measurement uncertainty bounds of the LEFM✓™ were less than ± 0.6 percent.

In approving Caldon Topical Report ER-80P, the NRC staff included four additional criteria to be addressed by a licensee requesting a power uprate. The licensee addressed each of the four criteria as follows:

1. The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

The staff reviewed the process by which the LEFM✓™ will be calibrated prior to shipment to SSES and found the process to be consistent with calibration processes for safety-related instrumentation. Therefore, the NRC staff finds the initial calibration process to be acceptable.

By letter dated May 22, 2001, the licensee provided supplemental information in response to questions from the NRC staff regarding implementation of the LEFM✓™ at SSES. The licensee states in this supplemental information that the design of the LEFM✓™ is such that the operator can detect suspect or inoperable LEFM✓™ components. If the system cannot be restored to operability within the time period specified in the TS, the licensee states that the backup feedwater flow measurement system (the current venturi signals) will be used as input to the core thermal power

calculation. Additionally, the licensee will reduce the allowed operating power level to the current RTP of 3441 MWt when the venturi system is used as input to the core thermal power calculation. The licensee states that these requirements will be included in the Technical Requirements Manual (TRM) and the plant procedures for alarm response. The licensee further states that plant procedures will be revised to this effect prior to operation at the increased RTP. These actions are in accordance with the amended ECCS rule and, therefore, are acceptable.

Instrumentation sensors for reactor pressure, feedwater flow, control rod drive flow, feedwater temperature, recirculation pump power, reactor water cleanup system temperature and flow, and total core flow will provide inputs into the reactor heat balance calculation. The licensee stated that these instruments will be calibrated and maintained by either preventative maintenance activities and/or by surveillance activities. The licensee defined preventative maintenance activities as those activities that extend equipment service life or prevent equipment failure and are based on engineering judgement and manufacturer recommendations. Surveillance activities were defined as those activities that are performed to satisfy TS or TRM requirements. These definitions are consistent with industry definitions and, therefore, are acceptable.

The licensee stated in the supplemental information that, for the instrumentation used in the heat balance calculations, loop calibrations are scheduled and performed in accordance with the SSES "Routine Task System," "Surveillance Testing Program," and the "Maintenance and Control of Installed Instrumentation" procedure. The licensee stated that these programs and procedures are in accordance with SSES FSAR Section 17.2, "Quality Assurance During the Operation Phase." This program addresses the establishment and conformance of the SSES physical configuration and associated documentation with SSES design and licensing requirements. The NRC staff finds that loop calibration controls will be maintained through a formal quality assurance program and, therefore, are acceptable.

Although the LEFM[✓]™ function is not nuclear safety-related because it provides feedwater flow and temperature inputs only to the calorimetric calculation, the LEFM[✓]™ software has been developed and will be maintained under a software quality assurance (SQA) program. The SQA program has been applied to all system software and hardware, and includes a detailed code review.

The licensee stated that the LEFM[✓]™ software configuration will be controlled by a combination of processes that consist of the following:

- the PPL Process Computer SQA program and lower-tier instructions, which manage the software design, configuration, and control of supplier services
- the PPL Modification process, which controls design, configuration changes, and installation
- the PPL Corrective Action process and the Work Order process, which are used to conform the system to its design function

In addition to these design programs, the licensee states that the LEFM[✓]™ Computer System SQA plan is written to specifically and uniquely prescribe the processes that are used for this system, include the following elements:

- system identification
- responsibilities
- assumptions, dependencies, and constraints
- system resource requirements
- standards
- software classification
- documentation configuration management
- software configuration management
- software change control
- configuration maintenance
- access control

Furthermore, the licensee stated that development of the LEFM[✓]™ system was controlled under the Caldon Quality Assurance program, which is in compliance with industry SQA standards and the PPL SQA program. The NRC staff finds these software development and maintenance control areas acceptable.

The licensee stated that the hardware configuration of plant systems and components will be controlled in accordance with a configuration management program that is pursuant to SSES FSAR Section 17.2, "Quality Assurance During the Operation Phase." These programs are applied to the equipment that affects the total power uncertainty used in the licensee's power uprate application, and are acceptable.

The licensee's existing deficiency control program (Condition Report Process) focuses on prompt identification, documentation, and correction of conditions that are adverse to quality or safety. The program contains provisions for tracking and trending conditions, as well as provisions for identifying and analyzing precursors to conditions adverse to quality. Corrective actions are performed in accordance with appropriate plant programs. The NRC staff finds this approach for addressing corrective actions to be acceptable.

The licensee stated that part/equipment deficiencies identified at SSES will be documented using the Condition Report Process. As part of the investigation, the work group responsible for resolving the condition report will contact the manufacturer, as required. The Condition Report Process includes process steps that require evaluation for reportability concerns. The reportability evaluation process includes the consideration of reporting requirements specified by 10 CFR Part 21. This program is applied to the equipment that affects the total power uncertainty described in the licensee's power uprate application. The NRC staff finds this approach for reporting deficiencies to the manufacturer to be acceptable.

The licensee stated that it implements a comprehensive Industry Event Review Program (IERP). The purpose of the program is to collect lessons learned information from the nuclear industry to preclude similar events at SSES. Notices such as those received

from the NRC, 10 CFR Part 21 reports, manufacturer/vendor notices, etc., are evaluated by the IERP.

If the IERP determines that a notice is applicable to SSES, the Condition Report Process is entered and used to control the evaluation, prioritization, and tracking of any warranted corrective actions. This program is applied to the equipment that affects the total power uncertainty described in the licensee's power uprate application. The NRC staff finds this approach for receiving and addressing manufacturer deficiency reports acceptable.

2. For plants that currently have a LEFM✓™ installed, the licensee should provide an evaluation of the operational and maintenance history of the installation, and confirm that the installed instrumentation is representative of the LEFM✓™ system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

The licensee will install the LEFM✓™ system as part of this licensed power uprate. Therefore, the requirement for an evaluation of the operational and maintenance history of the installation is not applicable to SSES for this power uprate.

3. The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM✓™ in comparison to the current feedwater instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternate methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installation for comparison.

The SSES licensing basis setpoint methodology provides the reactor coolant system control system uncertainties and the reactor trip system and ESF trip setpoints used in the plant safety analysis. The setpoint methodology is based on SSES plant setpoint methodology. The licensee stated that the current instrumentation setpoint calculation is not affected by the 1.4 percent power uprate of SSES.

The licensee's evaluation of the total uncertainty in the SSES overall core thermal power is 0.551 percent, which is bounded by the 0.6 percent value included in the Caldon topical report and its supplement, which is an estimated total power uncertainty based on an LEFM✓™ measurement of flow in a generic two-loop BWR system. Accordingly, the uncertainty calculation performed by the licensee is acceptable.

4. Licensees for plant installations where the ultrasonic meter (including the LEFM✓™) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show either that the meter installation is independent of the plant-specific flow profile for the stated accuracy or that the installation can be shown to be equivalent to known calibrations and the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM✓™ installation and calibration assumptions.

The LEFM✓™ spool pieces will be installed in existing sections of piping in each of the three feedwater flow lines. The sections of piping in which the spool pieces will reside were chosen to ensure that the piping configurations bound the original LEFM✓™ installation and calibration assumptions, thereby ensuring that the calculated error analysis for the instrumentation is not compromised. This installation configuration is consistent with the guidance approved in ER-80P and, therefore, is acceptable.

The NRC staff finds that the licensee's responses to these criteria have sufficiently resolved the plant-specific concerns about LEFM✓™ maintenance and calibration, hydraulic configuration, processes and contingencies for an inoperable LEFM✓™, and the methodology for the plant-specific calculations of the LEFM✓™ power measurement uncertainty.

3.4.2.3 NRC Staff Conclusion

The staff evaluation found the calculation of the power calorimetric measurement uncertainty for the SSES power uprate to be acceptable. Based on its review of the licensee's plant-specific LEFM✓™ error band calculation, the NRC staff finds that the SSES LEFM✓™ thermal power measurement uncertainty is less than 0.6 percent of the actual reactor thermal power, and can support the proposed 1.4 percent uprate of the SSES licensed thermal power. The NRC staff also finds that the licensee has sufficiently addressed the four additional criteria outlined in the staff's SE of Caldon Topical Report ER-80P. Therefore, the NRC staff finds that the licensee's request for a 1.4 percent thermal power uprate is acceptable.

3.5 Electrical Power and Auxiliary Systems

3.5.1 Electrical Power Systems

The offsite power system includes transmission facilities to customers and interconnections to other utilities. Unit 1 supplies the 230kV transmission system and Unit 2 supplies the 500 kV transmission system.

The licensee evaluated the power transmission system and determined that all switchyard equipment, with the exception of the Transformer #21 230kV line 3000 amp gang-operated circuit breakers in the 230kV switchyard, can support operation at increased RTP. Transformer #21 interconnects the 500kV system with the 230kV system. Operation at increased power will not impact thermal loading or voltage, but will affect the stability of the grid under a particular faulted condition identified as Fault Test Case N-10.

To remedy this situation, the licensee is planning to replace the two Transformer #21 230kV line 3000 amp gang-operated circuit breakers with 3000 amp independent pole operated circuit breakers. This is acceptable since this replacement will result in a stable grid under all analyzed conditions, and the evaluation of this case in the FSAR will remain unchanged.

The main alternating current (AC) generator supplies power to the auxiliary transformers for onsite power distribution, as well as the main step-up transformers for offsite distribution.

Station loads under normal operation and distribution conditions are computed-based on equipment nameplate data. Operation at the uprate using existing equipment will be at or

below the nameplate rating. Therefore, under normal conditions, the electrical supply and distribution components are adequate.

Station loads under emergency operation and distribution conditions using emergency diesel generators (EDGs) are based on equipment nameplate data except for emergency service water and RHR pumps where the operating point is used, and the CS pumps where high-flow brake horsepower (BHP) is used. Operation at increased RTP will be achieved by using existing equipment operating at or below the nameplate rating and within the calculated BHP for the stated pumps. Therefore, under emergency conditions, the electrical supply and distribution components are adequate.

The licensee evaluated the electrical loads that increased as a result of the uprate, and found that all electrical load increases can be supported by the present electrical distribution system configurations.

Safety-related load changes associated with increased RTP operation have been reviewed by the licensee against the existing plant voltage model and have been found to be bounded. The licensee has work ongoing to upgrade the plant's voltage model to incorporate these load changes prior to operating under the new RTP.

The licensee evaluated the main step-up transformers for offsite distribution, and found them to be capable of handling the increased power for Unit 1. The Unit 2 transformers were replaced prior to operation at 3441 MWt, and will support operations at increased RTP. Therefore, both Units' transformers are acceptable for operation at the increased RTP.

The licensee's evaluation of the EDGs and auxiliaries confirmed that the loads on the 4kV 1E buses will remain within the original design. The bus voltage and under-voltage trip setpoints will not change, and no changes are required in the operating modes, control logic or EDG start timing. Therefore, the EDGs and their auxiliaries will continue to be adequate for operation at the increased RTP.

The direct current (DC) systems at SSES, Units 1 and 2, consist of battery banks, battery chargers, load centers, distribution panels and motor control centers. The DC power systems will not be affected by operation at increased RTP, because the uprate will not increase the design loads or operation of the DC systems.

The generator stator cooling and the hydrogen cooling systems are designed for operation of the turbine generator at valve-wide-open steam flow conditions, and the flow rate will not be affected by the uprate. The licensee's evaluation determined that no modifications of the stator and hydrogen cooling systems are necessary for the uprate, and the cooling systems will support operation at the proposed increased RTP.

The licensee evaluated the safety-related electrical equipment in accordance with Institute of Electrical and Electronics Engineers (IEEE) standard IEEE 323, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," to ensure that the electrical equipment inside the containment is qualified for normal and accident conditions resulting from a main steamline break or a DBA-LOCA. The licensee's evaluations revealed that the DBA-LOCA are not changed and the existing environmental qualifications for safety-related equipment inside

the containment will not be affected by the uprate during normal operation and accident conditions.

The equipment outside the containment is qualified for temperature, pressure, and humidity environments resulting from a main steamline break in the pipe tunnel, DBA-LOCA, or other high-energy line breaks, whichever is limiting for each plant area. The licensee analyzed these events, and concluded that equipment qualification remains bounded by existing analyses, and will not be changed as a result of the uprate. Therefore, the present environmental qualification for the safety-related equipment outside the containment did not change.

Based on the above evaluation, the NRC staff concludes that the licensee has provided assurance that the 1.4 percent power uprate will not affect the safety functions and environmental qualifications of the electric power systems at SSES, Units 1 and 2. The replacement of the gang-operated circuit breakers by independent pole-mounted circuit breakers will ensure the grid stability at the higher power level for the same series of electrical faults. The licensee indicated that this modification will be performed prior to the second unit (i.e., Unit 1 in Spring 2002) completing the power uprate. This is consistent with GDC 17, and therefore, the proposed change is acceptable.

3.5.2 Fuel Pool Cooling and Cleanup System

The licensee stated that the fuel pool cooling system will remain within the design heat removal capacity of the heat removal systems for both normal and emergency loads at the proposed uprated power conditions. In addition, a margin of approximately 16 percent, through the RHR fuel pool cooling assist mode, remains on the fuel pool cooling system at the proposed uprated conditions.

Based on the review of the licensee's rationale and the experience gained from its review of power uprate applications for similar BWR plants, the NRC staff finds that plant operations at the proposed uprated power level do not change the design aspects and operation of the fuel pool cooling system.

3.5.3 Water Systems

3.5.3.1 Safety Related Cooling Water Systems

The licensee evaluated the impact of operating at 101.4 percent of current rated thermal power on the following safety-related cooling water systems:

- emergency service water system
- residual heat removal service water system
- ultimate heat sink

These systems provide cooling water to various plant systems and components during normal and accident conditions. In its SE dated November 30, 1993, the NRC staff found that operation at an analyzed power level of 102 percent of the current RTP (3510 MWt) had little or no impact on the performance of these systems. Consequently, the analysis performed at 3510 MWt bounds the proposed uprated conditions, and the NRC staff finds that operation of the above systems at the uprated power level is acceptable.

3.5.3.2 Non-Safety Related Cooling Water Systems

The licensee evaluated the impact of operating at 101.4 percent of current RTP on the following non-safety related cooling water systems:

- non-safety related service water systems
- reactor building closed cooling water system
- turbine building closed cooling water system
- gaseous radwaste recombiner closed cooling water system
- river water makeup
- chilled water systems

The licensee determined that the power uprate had little or no impact on the performance of these systems.

However, the NRC staff did not review the impact of plant operations at the uprated power level on the design and performance of these systems, because they perform no safety-related function. Additionally, the failure of these systems will not affect the performance of any safety-related system or component.

3.5.4 Standby Liquid Control (SLC) System

The shutdown capability of the SLC system and the boron solution necessary are evaluated each reload cycle. As indicated in FSAR Section 9.3.5, the SLC system is designed to inject 82.4 gpm at a maximum reactor pressure equal to the minimum SRV setpoint pressure. Because the SRV setpoints are not changed for the proposed power uprate, the uprate will have no effect on the rated flow injection. The capability of the SLC system to provide its backup shutdown function and meet the requirements of 10 CFR 50.62 with the current SRV setpoints was evaluated and found acceptable for the previous power uprate.

Subsequently, in response to NRC inspection (Inspection Report (IR) 05000387/2001-004 and 05000388/2001-004, Reference 23), PPL modified the SSES Unit 2 SLC system during refueling outage 10 in the spring of 2001. The system modification ensures that the SLC system can inject the required flow during a loss-of-offsite-power (LOOP) anticipated transient without scram (ATWS) event without exceeding the SLC system relief valve setpoint. In its May 22, 2001, supplement, the licensee stated that these modifications ensure conformance with the ATWS analysis assumptions. Moreover, the licensee stated, "Regarding Unit 1, prior to implementation of the power uprate on Unit 1 in the Spring 2002, PPL commits to implement modifications on Unit 1 SLC system so that the SLC ATWS analysis also remains valid for Unit 1."

The NRC staff concludes that the above regulatory commitment warrants the creation of a regulatory requirement requiring prior NRC approval of subsequent changes. Therefore, implementation of this amendment for Unit 1 will be conditioned on the licensee's completion of modifications to the SLC system as committed to in its May 22, 2001, supplemental letter. The NRC staff concludes that, provided the SLC system design modifications which ensure the SLC system can inject the required flow during the LOOP/ATWS transient are implemented before

operation at the uprated conditions, the SLC system will be capable of performing its design and licensing basis functions at the uprated power level.

3.5.5 Heating, Ventilation, and Air Conditioning (HVAC) Systems

The licensee evaluated the impact of higher process fluid temperature in piping for all HVAC systems including the drywell cooling system, reactor building HVAC system, control structure HVAC system, radwaste building HVAC system, turbine building ventilation system, engineered safeguards service water pump house heating and ventilation system, and the diesel generator building ventilation system.

The licensee performed an evaluation which confirmed that power uprate will have no significant effect on the above HVAC systems. This evaluation was completed at a thermal power level of 3510 MWt that the NRC staff approved in a SE dated November 30, 1993. The previous analysis bounds the proposed increase in RTP to 3489 MWt. Therefore, the NRC staff finds that operation of these HVAC systems at the uprated power level is acceptable.

3.5.6 Fire Protection System

The licensee stated that the operation of the plant at the uprated power level will not affect the fire-suppression or the fire-detection systems. The uprate will not result in any changes to the physical plant configuration or combustible load, and the safe-shutdown systems and equipment used to achieve and maintain cold-shutdown conditions will not change, and will remain adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire-protection systems and analyses are not affected by power uprate.

Based on its review and experience gained from review of power uprate applications for similar BWR plants, the NRC staff finds that the licensee's evaluation that the 1.4 percent power uprate will not change the design and operational aspects of the fire protection system is acceptable.

3.6 Power Conversion Systems

The steam and power conversion systems and associated components (e.g., the turbine/generator, condenser and steam jet air ejectors, turbine steam bypass, feedwater and condensate systems) were originally designed to utilize the energy available from the NSSS and to accept the system and equipment flows resulting from continuous operation at 105 percent of the currently licensed rated power. Therefore, these systems will not be affected by the proposed power uprate.

The NRC staff did not review the impact of plant operations at the uprated power level on the design and performance of these systems, because they perform no safety-related function. Additionally, the failure of these systems will not affect the performance of any safety-related system or component.

3.7 Radwaste Systems and Radiation Sources

The licensee evaluated the effects resulting from plant operations at 101.4 percent of the current licensed RTP on the liquid, gaseous, and solid radwaste systems. The increase in core thermal power operation will increase the liquid radwaste influents by an proportional amount, and the airborne effluent activity released through building vents is not expected to increase significantly at the proposed uprated power level. Both the liquid and gaseous radwaste systems were previously analyzed (Reference 4) at an increased RTP of 3510 MWt and were found to continue to meet the requirements of 10 CFR Part 20 and Appendix I to 10 CFR Part 50. The previous analysis completed by the licensee and approved by the NRC staff on November 30, 1993, bounds the proposed increase in RTP to 3489 MWt, and is still valid. Therefore, the NRC staff finds that operation of the liquid and gaseous radwaste systems at the uprated power conditions is acceptable.

3.8 Reactor Safety Performance Features

3.8.1 Reactor Transients

3.8.1.1 Regulatory Bases of Anticipated Operational Occurrences

AOOs are abnormal transients which are expected to occur one or more times in the life of a plant and are initiated by a malfunction, a single-failure of equipment, or a personnel error. The applicable acceptance criteria for the AOOs are based on 10 CFR Part 50, Appendix A, GDC 10, 15, and 20. GDC 10 requires that the reactor core and associated control and instrumentation systems be designed with sufficient margin to ensure that the specified acceptable fuel design limits are not exceeded during normal operation and during AOOs. GDC 15 stipulates that sufficient margin be included to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions and AOOs. GDC 20 specifies that a protection system be provided that automatically initiates appropriate systems to ensure that the specified fuel design limits are not exceeded during any normal operating condition and AOOs.

The SRP (Reference 7) provides further guidelines: (1) pressure in the reactor coolant and main steam system should be maintained below 110 percent of the design values according to the ASME Code, Section III, Article NB-7000, "Overpressure Protection"; (2) fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during normal operating conditions and AOOs; (3) an incident of moderate frequency should not generate a more serious plant condition unless other faults occur independently; and (4) an incident of moderate frequency, in combination with any single-active component failure or single operator error, should not result in the loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel cladding perforations are acceptable.

3.8.1.2 Evaluation

Chapter 15 of the SSES FSAR contains the design-basis analyses that evaluate the effects of an AOO resulting from changes in the system parameters such as (1) a decrease in core coolant temperature; (2) a increase in reactor pressure; (3) a decrease in reactor coolant flow rate; (4) reactivity and power distribution anomalies; (5) an increase in reactor coolant inventory;

and (6) a decrease in reactor coolant inventory. The facilities' responses to the most limiting transients are analyzed each reload cycle and corresponding changes in the MCPR are added to the SLMCPR to establish the OLMCPR. A potentially limiting event is an event or an accident that has the potential to affect the core operating and safety limits.

PPL has evaluated the impact of the 1.4 percent power uprate on the AOOs analyzed in the SSES Units 1 and 2 FSAR and stated that the relatively small change in the RTP will not affect the selection of limiting events. The licensee will explicitly analyze the following limiting transients in the cycle-specific reload analysis:

- loss of feedwater heating
- feedwater controller failure (FWCF) - maximum demand
- generator load reject without bypass (GLRWOB)
- turbine trip without bypass
- rod withdrawal error (RWE)
- recirculation flow controller failure, increase (RFCF)
- fuel loading error

The licensee pointed out that the limiting events that establish the OLMCPR are currently GLRWOB, FWCF and RFCF. The licensee stated that these transients are expected to remain the limiting transients, but all seven transients will continue to be analyzed every reload and the RFCF event will be analyzed without taking credit for the flow-biased simulated thermal power trip. The limiting transients will be analyzed at 3510 Mwt (100.6 percent uprated) to establish the OLMCPR according to an NRC-approved methodology.

The NRC staff agrees with the licensee that the plant operating parameters at the uprated conditions will not change significantly. Therefore, a nonlimiting transient is not expected to become limiting or change the applicability of the NRC-approved methodology. Transients are analyzed at offrated conditions, maximum rated conditions, or higher conditions depending on the transient's sensitivity to core parameters such as power, core flow, and inlet subcooling. In the May 22, 2001, draft FSAR supplement, the licensee submitted the results of the transient reload analyses for the uprated cycle for Unit 2. The supplement provided the initial conditions assumed in the limiting transients and the sequence of events and actuation assumed in the analyses. The recirculation pump seizure accident event established the OLMCPR for the Unit 2 uprated cycle, and the licensee also determined the SLMCPR and OLMCPR limits for single-loop operation. Therefore, the licensee has performed the reloaded analysis at the uprated condition for Unit 2 using an NRC-approved methodology and determined that the thermal limits to ensure the fuel cladding integrity will be maintained for operation at the uprated conditions during AOOs and accidents.

3.8.2 Radiological Analysis of DBAs

The licensee stated that the current radiological consequence assessments for the DBAs are based on 105 percent of the current power level (i.e., 3616 MWt). Thus, the licensee concluded that the proposed increase in core thermal power level to 3489 MWt will have no effect on the post-accident source term analyses. The staff reviewed the SSES FSAR, and the licensee's amendment request describing the proposed increase in rated core thermal power. This review of the radiological consequences analyzed for the DBAs in Chapter 15 of the SSES FSAR

confirmed that the current analyzed power level of 3616 MWt bounds the requested power level of 3489 MWt, and that the radiological consequences calculated at a reactor core thermal power level of 3616 MWt meet the relevant dose acceptance criteria. Because the reactor accident source terms and release rates used in the current analyses remain bounding, the current calculated radiological consequences in the SSES FSAR remain bounding.

The NRC staff has reviewed the licensee's amendment request and has concluded that the current design-basis dose analyses, as documented in the SSES FSAR, remain acceptable in that reasonable assurance exists that the radiological consequences, with proposed 1.4 percent reactor core thermal power uprate, will remain the same or bounded by the current values. The NRC staff has determined the proposed changes are acceptable with respect to the radiological consequences of the DBA analyses.

3.8.3 Anticipated Transients Without Scram (ATWS)

3.8.3.1 Regulatory Basis

ATWS is defined as an AOO with failure of the reactor protection system to initiate a reactor scram to terminate the event. The requirements for ATWS are specified in 10 CFR Part 62. The regulation requires BWR facilities to have the following mitigating features for an ATWS event:

- (1) a SLC system with the capability of injecting a borated water solution with reactivity control equivalent to the control obtained by injecting 86 gpm of a 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch-inside diameter reactor vessel,
- (2) an alternate rod injection (ARI) system that is designed to perform its function in a reliable manner and that is independent from sensor output to the final actuation device, and
- (3) equipment to trip the reactor coolant recirculation pumps automatically under conditions indicative of an ATWS.

BWR facilities meet the ATWS acceptance criteria to demonstrate the ability to withstand an ATWS event, that is maintaining the fuel integrity (the core and fuel must maintain a coolable geometry), the primary system integrity (the peak reactor vessel pressure must remain below 1500 psig), and the containment integrity (the containment pressure must not exceed the design limit).

The SLC system provides the alternate means of attaining and maintaining cold shutdown conditions, assuming no control rod movement as required by GDC 26. The ATWS analyses assume the SLC will inject at a specified time to bring the reactor to and maintain it at cold shutdown during an AOO event. For every reload, the licensee must evaluate how plant modifications, reload core designs, changes in fuel design, and other reactor operating changes affect the applicability of the ATWS analysis of record.

3.8.3.2 Evaluation

PPL stated that the reload safety analysis for the first fuel cycle with the increased thermal power will contain the appropriate ATWS analysis. SSES Unit 2 complies with the ATWS requirements, having an automatic recirculation pump trip (ATWS-RPT) to produce negative reactivity and an ARI and SLC system to shut down the reactor.

The May 22, 2001, draft FSAR update discussed the ATWS analysis for Unit 2 Cycle 11. According to the supplement, the ATWS analyses are performed in-house when changes in reactor operating condition or changes in core design warrant reanalysis. In these cases, the licensee analyzes the two limiting ATWS transients of MSIVC and the pressure regulator failure open (PREGO) using the RETRAN code. The ATWS analysis is based on RTP and 87×10^6 lbm/hr core flow along the extended load limit line of the power/flow map. The supplement also contained the MSIVC/ATWS analysis results for Unit 2 Cycle 11. The limiting MSIVC/ATWS transient resulted in a peak vessel pressure of 1341 psig (1500 psig limit) and a peak cladding temperature of 1462 °F (2200 °F limit).

On March 2, 2001, the NRC performed a team inspection of SSES Units 1 and 2. The findings are documented in IR 05000387/20001-004 and 05000388/2001-004. The inspection team determined that in the LOOP/ATWS event analysis, the dome pressure reaches 1200 psig on several occasions during the event. The inspection team stated that considering the number of available SRV lifts, the SRV lift setpoint tolerances, and the dome pressure during the event, the SLC pump discharge relief valve could lift during the LOOP event. The lifting of the SLC pump discharge relief valve could cause the sodium pentaborate solution to be recycled back to the pump suction, thus, reducing the SLC system's ability to inject the flow specified in 10 CFR 50.62. PPL performed a safety assessment of the SLC system's ability to inject the borated solution consistent with the assumptions of the ATWS analyses. In response to the inspection findings, PPL modified the design of the SLC system for Unit 2 by replacing the flanges of the two SLC pumps with higher rated flanges and by increasing the SLC pump discharge relief valve setpoints to 1500 psig. In addition, the licensee committed to perform similar SLC system design changes for Unit 1 during the spring 2002 refueling outage. As noted in Section 3.5.4 of this SE, implementation of this amendment for Unit 1 will be conditioned on the licensee's completion of the SLC system modifications committed to in its May 22, 2001, supplemental letter.

The MSIVC/ATWS event is the limiting transient for peak pressure and heat flux, which occur at the beginning of the event, before SLC is assumed to inject. This event is analyzed for every cycle. The draft FSAR update shows that at the uprated conditions, this ATWS scenario does not threaten the fuel or the vessel integrity. For the LOOP event, the loss of containment instrument gas compressors and the limited SRV actuation lifts from the gas accumulators result in high vessel pressure during the assumed SLC injection time. The inspection team estimated that considering a 1 percent SRV tolerance and a pressure pulsation margin for positive displacement pumps, the maximum SLC pump discharge pressure would be 1437 psig (including the vessel bottom head pressure losses and system piping pressure losses) for the ATWS/LOOP event. Although the 1.4 percent uprate may slightly increase the required SLC discharge pressure during the ATWS/LOOP event, the staff does not expect the impact of the proposed power uprate will require SLC pump discharge pressure of 1500 psig. Therefore, the design modification will ensure that SLC can inject the required flow rate at the uprated conditions. In addition, the fuel and vessel integrity will not be threatened because the peak

pressure, power, and heat flux for the loop event is bounded by the calculated MSIVC event. Therefore, the NRC staff finds that the licensee's assessment that SSES, Units 1 and 2, will meet the ATWS 10 CFR 50.62 analysis, provided the SLC system design modifications are implemented before operation at the uprated conditions.

3.8.4 Station Blackout (SBO)

Under 10 CFR 50.63, the reactor core and the associated coolant, control, and protection systems must have sufficient capacity to cool the core and maintain containment integrity in the event of a SBO of a specified duration. The licensee must analyze the plant's capability to cope with a SBO of specified duration. The licensee evaluated the impact of the 1.4 percent power uprate on SSES Units 1 and 2 capability to cope with a SBO event in accordance with NUMARC 87-00, which has been endorsed by the NRC in RG 1.155, "Station Blackout," August 1988.

SSES, Units 1 and 2, are classified as four-hour SBO duration plants, based on an offsite power design characteristic group of "P1," an emergency AC power configuration group of "D," and a target emergency diesel generator reliability of 0.975. The licensee stated that the proposed power uprate will not change the four hours duration classification. According to PPL, the affected parameters for SBO of longer duration are the water inventory for decay heat removal, the class 1E battery capacity, the compressed air capacity, and the effects of loss of ventilation. The power uprate will increase the decay heat slightly, which may have an effect on the condensate storage tank (CST) inventory necessary for core cooling. The licensee added that the current SBO analysis demonstrates that the CST provides adequate water inventory to meet the additional requirement of increased power operation. In conclusion, the licensee stated that SSES will continue to meet the 10 CFR 50.63 requirements for the proposed power uprate, considering the small increase in decay heat associated with the proposed power uprate. The NRC staff accepted the licensee's analysis of the SBO event in its SE dated November 30, 1993, which analyzed the SBO event at an increased RTP of 3510 MWt, that bounds the proposed increase in RTP (3489 MWt). Therefore, the NRC staff finds that plant operation at the uprated power level is acceptable with regard to the plant's SBO coping capabilities.

3.9 General Issues

3.9.1 Line Break Analyses and Environmental Qualification of Equipment

The licensee determined that the following general issues were not affected by the proposed 1.4 percent power uprate:

- high-energy line break
- moderate energy line break
- equipment qualification

The licensee previously evaluated the plant's high and moderate energy line break analyses at 102 percent of the current licensed power level (3510 MWt) and determined that operation of the plant at 3510 MWt power is acceptable. Because this evaluation bounds the proposed uprated power level, the NRC staff finds that plant operation at the uprated power level is acceptable relative to high and moderate energy line breaks.

In its SE dated November 30, 1993, the NRC staff approved the licensee's evaluation of environmental qualification of mechanical equipment with non-metallic components analyzed at an RTP level of 3510 MWt. The licensee, in the previously approved case, indicated that the qualified life of certain equipment may be reduced, but a revised aging analysis will assure replacement before the equipment exceeds its qualified life. Therefore, the NRC staff finds that the proposed increase in RTP level of 3489 MWt is acceptable with regard to environmental qualification of mechanical equipment, as it is bound by the previous approved analysis.

3.9.2 Selected Generic Issues

The licensee reviewed its GL 89-10, "Safety-Related Motor-Operated Valve Testing And Surveillance," motor-operated valve (MOV) program and indicated that the MOV evaluation at SSES, Units 1 and 2, was performed using the worst-case parameters from the accident analyses and therefore, bounds the proposed 1.4 percent power uprate condition. The licensee evaluated its commitments relating to generic letter (GL) 95-07, "Pressure Locking and Thermal Binding of Safety- Related Power-Operated Gate Valves," associated with the operation of safety-related power-operated gate valves that are required to operate to perform their intended safety function. The licensee found that the existing analysis conditions remain bounding for the 1.4 percent power uprate. The licensee also evaluated its response relating to its GL 96-06, "Assurance of Equipment Operability And Containment Integrity During Design-Basis Accident Conditions," regarding the over-pressurization of isolated piping segments. The licensee indicated that the existing evaluation for GL 96-06 was performed assuming the worst case drywell temperature which is independent of the proposed power uprate. On the basis of the above review, the NRC staff finds acceptable the licensee's conclusions that the power uprate will have no adverse effects on the safety-related valves and that the licensee's conclusions from the GL 95-07, GL 96-06, and GL 89-10 programs remain valid.

3.9.3 Emergency Preparedness and Licensed Operator Performance

3.9.3.1 Changes in Emergency and Abnormal Operating Procedures

The licensee stated that increasing the core thermal operating power results in some modifications to the curves referenced in the Emergency Operating Procedures (EOP's) and the calculations that support the curves. The boron injection initiation time (BIIT) is a weak function of core thermal power and the BIIT curve is part of the ATWS analysis, and will be revised for each operating cycle. Core thermal power level also shows up as a parameter in several other calculations supporting the EOP's and will require small changes, but with training will be transparent to the operators.

The NRC staff finds that the licensee's response is satisfactory because, consistent with RG 1.33, "Quality Assurance Program Requirements (Operation)," PPL has adequately identified the type and scope of plant procedures that will be affected by the uprate, indicated that the procedures will be appropriately revised, operators will be trained on the changes before the procedures are implemented, and adequately described the effect of the procedure changes on operator actions.

3.9.3.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

The Individual Plant Evaluation (IPE) and the evaluation performed to assess the effect of the previous power uprate on the IPE were reviewed by the licensee to determine the effect of increased licensed core thermal power on the conclusions of the IPE. The net effect on the IPE of operation at increased RTP is to remove an amount of time proportional to the operating power increase from operator response and equipment repair times. The previous power uprate of approximately 5 percent thermal power showed that the reduction in time for operator action is approximately equal to the increase in core thermal power and the corresponding increase in decay heat generation. The licensee states that the proposed small increase in RTP will have negligible impact on operator response time and the effect on other analysis documented in the IPE is so small as to be not quantifiable. Therefore, the licensee states that the additional 1.4 percent core thermal power increase does not have any significant impact on IPE results.

The NRC staff finds that the licensee's response is satisfactory because it has adequately addressed the question of operator actions sensitive to the power uprate by describing the effect of the power uprate on operator performance, and adequately justifying the effect/lack of effect on required operator response.

3.9.3.3 Changes to Control Room Controls, Displays and Alarms

The licensee stated that "the LEFMTM feedwater flow measurement system contains a self-checking module, which alarms when the system diagnoses a failure in the flow measurement algorithm." As a result of this alarm, a new alarm response procedure is required to detail required actions for the Operations Staff. The alarm response will be written and the Operations Staff will be trained on the alarm response procedure prior to the implementation of operation at increased RTP.

The NRC staff finds that the licensee's response is acceptable because it has adequately identified the changes that will occur to alarms, displays, and controls as a result of the power uprate, and adequately described how these changes will be accommodated.

3.9.3.4 Changes on the Safety Parameter Display System (SPDS)

The licensee stated that the SPDS is insensitive to normal operating power level, therefore, operation at increased RTP has no effect on the SPDS.

The NRC staff finds that the licensee's response is satisfactory.

3.9.3.5 Changes to the Operator Training Program and the Control Room Simulator

The licensee stated that "the operation staff will be trained on operation at increased RTP prior to actual operation at increased RTP. Since Unit 2 will be the lead unit for the implementation of increased RTP, the operations staff will be fully trained on the differences between the units, as currently occurs in the operator training program. The plant-specific simulator is referenced to Unit 1, however, the simulator can be reprogrammed to operate at the increased RTP. The programmed simulator will accept the increase power level and adjust operating flows, etc., to be consistent with the increased power level."

As discussed above, the major impact on Operations Staff is the slight decrease in timing associated with operator response. The licensee states that the Operations Staff will be trained on this change prior to the implementation of operation at increased RTP.

The NRC staff finds that the licensee's response is acceptable because it has adequately described how the changes to operator actions will be addressed by the simulator and how the simulator will accommodate the changes in accordance with the requirements of American Nuclear Society/American National Standards Institute (ANS/ANSI) Standard 3.5.

3.9.3.6 Human Factors Evaluation Conclusion

The NRC staff concludes that the previously discussed review topics associated with the proposed SSES, Units 1 and 2, power uprate have been, or will be, satisfactorily addressed. The NRC staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

3.10 FOL and TS Changes

The following changes to the SSES Units 1 and 2 FOLs and TSs are required to reflect the new authorized power levels:

- Paragraph 2.C.(1) - FOLs NPF-14 and NPF-22 are revised to authorize operation at reactor core power levels not in excess of 3489 MWt.
- TS 1.1 - The definition of Rated Thermal Power is revised to reflect the increase from 3441 MWt to 3489 MWt.
- TS 5.6.5 - The core operating limits report (COLR) is revised to add references to Topical Reports ER-80P and ER-160P.

These FOL and TS changes reflect the actual proposed changes in the plant and are consistent with the results of the safety analysis. Accordingly, the NRC staff finds these changes to be acceptable.

4.0 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.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the *Federal Register* on June 25, 2001 (66 FR 33716). Accordingly, based upon the Environmental Assessment, the staff has determined that issuance of the amendment will not have a significant effect on the quality of the human environment.

6.0 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.

7.0 REFERENCES

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2. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for Power Uprate with LEFM[✓]™ System," (Engineering Report ER-160P), Revision 0, May 2000.
3. PPL Susquehanna, LLC, "Licensing Topical Report: Susquehanna Steam Electric Station Units 1 and 2, Power Uprate Resulting From Increased Feedwater Flow Measurement Accuracy," NE-2000-001P, Revision 1, February 2001.
4. Pennsylvania Power & Light Company, "Susquehanna Steam Electric Station Units 1 and 2, Licensing Topical report for Power Uprate with Increased Core Flow," Licensing Topical Report NE-92-001, Revision 0, July 1992.
5. Letter from T. E. Murley, U.S. Nuclear Regulatory Commission (NRC), to R. G. Byram, Pennsylvania Power and Light Company (PP&L), "Licensing Topical Report for Power Uprate with Increased Core Flow, Revision 0, Susquehanna Steam Electric Station, Units 1 and 2 (PLA-3788) (TAC Nos. M83426 and M83427)," November 30, 1993.
6. Siemens Power Corporation "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," ANF-91-048 (P)(A), January 1993.
7. NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981.
8. Siemens Power Corporation "Generic Mechanical Design Criteria for BWR Fuel Designs," ANF-89-98 (P)(A), May 1995.
9. Siemens Power Corporation, " Application of ANFB to ATRIUM-10™ for Susquehanna Reloads," EMF-97-010(P), Revision 1.
10. Siemens Power Corporation, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," ANF-524(P)(A), Revision 2 and ANF-524(P)(A) Supplement 1, Revision 2.
11. Siemens Power Corporation, "ANFB-10 Critical Power Correlation," EMF-1997(P)(A), Revision 0, July 1998.

12. Siemens Power Corporation, "ANFB-10 Critical Power Correlation: High Local Peaking Results," EMF-1997 (P)(A) Supplement 1, Revision 0, July 1998.
13. NRC Bulletin Number 88-07, "Power Oscillation in Boiling Water Reactors (BWRs)," June 15, 1988.
14. NRC Bulletin Number 88-07, Supplement 1, "Power Oscillation in Boiling Water Reactors (BWRs)," December 30, 1988.
15. Letter from H. W. Kaiser, PP&L, to C. L. Miller, NRC, "Response to Generic Letter 92-01 (PLA-3804)," dated July 8, 1992.
16. General Electric Nuclear Energy, "Generic Evaluation of Boiling Water Reactor Power Uprate," Volume I, NEDC-31984P, July 1991.
17. Siemens Power Corporation, "LOCA Break Spectrum Analysis for Susquehanna Units 1 and 2 Using 1993 EXEM BWR Evaluation Model," EMF-96-160(P), Revision 0, April 1997.
18. Siemens Power Corporation, "Susquehanna LOCA-ECCS Analysis MAPLHGR Limits for SQB-8 ATRIUM-10™ Fuel Using 1993 EXEM BWR Evaluation Model," EMF-96-161(P), Revision 0, April 1997.
19. Letter from C. Poslusny, NRC, to R. G. Byram, PP&L, "Amendment Nos. 121 and 151 for Technical Specification Change Deleting Reference to the Main Steam Isolation Valve (MSIV) Leakage Control System," August 15, 1995.
20. General Electric Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (non-proprietary), February 1992; and NEDC-31897P-A, Class III (proprietary), May 1992.
21. Letter from J. N. Hannon, NRC, to C. L. Terry, TU Electric, "Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System' (TACS Nos. MA2298 and MA2299), dated March 8, 1999.
22. Letter from R. E. Martin, NRC, to J. A. Scalice, Tennessee Valley Authority, "Watts Bar Nuclear Plant, Unit 1 - Issuance of Amendment Regarding Increase of Reactor Power to 3459 Megawatts Thermal (TAC No. MA9152)," dated January 19, 2001.
23. NRC, "Susquehanna Steam Electric Station, NRC Inspection Report 05000387/2001-004, 05000388/2001-004," May 21, 2001.

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Date: July 6, 2001

UNITED STATES NUCLEAR REGULATORY COMMISSION

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NOS. 50-387 AND 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT NOS. 1 AND 2

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment Nos. 194 and 169 to Facility Operating License Nos. NPF-14 and NPF-22, issued to PPL Susquehanna, LLC (PPL or the licensee), which revised the Facility Operating Licenses and Technical Specifications for operation of the Susquehanna Steam Electric Station (SSES), Unit Nos. 1 and 2, located in Luzerne County, Pennsylvania. The amendments are effective as of the date of issuance.

The amendments modified the Facility Operating Licenses and Technical Specifications for SSES, Units 1 and 2, to increase the licensed power level for each unit from 3441 megawatts thermal (MWt) to 3489 MWt, which is an increase of 1.4 percent of the rated core thermal power for SSES, Units 1 and 2.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for a Hearing in connection with this action was published in the FEDERAL

REGISTER on April 24, 2001 (66 FR 20691). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendments will not have a significant effect on the quality of the human environment (66 FR 33716).

For further details with respect to the action see (1) the application for amendments dated October 30, 2000, and supplemented February 5, May 22, May 31, and June 26, 2001, (2) Amendment No. 194 to License No. NPF-14, and Amendment No. 169 to License No. NPF-22, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland. Publicly available records will be accessible electronically from the Agencywide Documents Access and Management Systems (ADAMS) Public Electronic Reading Room on the internet at the NRC Web site, <http://www.nrc.gov/NRC/ADAMS/index.html>. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the NRC Public Document Room Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr@nrc.gov.

Dated at Rockville, Maryland, this 6th day of July 2001.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Robert G. Schaaf, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Commission public document room and added to the Agencywide Documents Access and Management Systems Publicly Available Records System (ADAMS PARS) Library. The Notice of Issuance will be included in the Commission's Biweekly Federal Register Notice.

Sincerely,

/RA/

Robert G. Schaaf, Project Manager, Section 1
 Project Directorate I
 Division of Licensing Project Management
 Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

- Enclosures: 1. Amendment No. 194 to License No. NPF-14
 2. Amendment No. 169 to License No. NPF-22
 3. Safety Evaluation (nonproprietary)
 4. Safety Evaluation (proprietary)
 5. Notice of Issuance

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TEMPLATE=NRR-058

* SE input provided. No major changes.

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