Robert G. Byram
Senior Vice President and
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OCT 3 0 2000

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station OP1-17
Washington, DC 20555

SUSQUEHANNA STEAM ELECTRIC STATION
PROPOSED AMENDMENT NO. 235 TO
LICENSE NPF-14 AND PROPOSED AMENDMENT
NO. 200 TO NPF-22: POWER UPRATE
PLA-5212 **Docket No. 50-388**

Reference: *PLA-5213, Submittal Of Supplement Engineering Report-157P, To Topical Report ER-80P In Support Of A Planned Power Uprate Technical Specification Change Request," Dated July 13, 2000.*

The purpose of this letter is to propose changes to the Susquehanna Steam Electric Station Unit 1 and Unit 2 Technical Specifications. This proposed change entails a power uprate justified as a result of a plant modification that will install the Caldon LEFM✓™ feedwater flow element. The LEFM✓™ provides improved feedwater flow measurement accuracy and thus improved operating power level certainty.

Attachment 1 contains a listing of Technical Specification submittals that are currently under NRC review or have been approved by the NRC but not yet implemented by PPL Susquehanna LLC. These are not affected by the changes proposed herein.

Attachment 2 to this letter is the "Safety Assessment" supporting this change.

Consistent with the previous PPL power uprate, the Attachment 3 contains the licensing topical report describing the results of the PPL evaluations of the proposed change. The report follows the NRC-approved generic format and content for BWR Power Uprate licensing reports developed by General Electric and reported in NEDC-31897P-A, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate."

A001

Attachment 4 is the No Significant Hazards Considerations evaluation performed in accordance with the criteria of 10 CFR 50.92 and the Environmental Assessment.

Attachment 5 to this letter contains the applicable pages of the Susquehanna SES Unit 1 and Unit 2 Technical Specifications, marked to show the proposed changes.

Attachment 6 contains "camera ready" versions of the revised Technical Specification pages.

Attachment 7 contains markups reflecting the changes necessary in the PPL Susquehanna LLC Unit 1 and Unit 2 licenses.

No Technical Specification Bases changes are required as a result of this change.

The proposed change has been reviewed by the Susquehanna SES Plant Operations Review Committee and by the Susquehanna Review Committee.

PPL plans to implement the proposed changes by April 30, 2001 on Unit 2 and April 30, 2002 on Unit 1. Therefore, we request NRC complete its review of this change by April 1, 2001 with the changes to be effective upon startup following the Unit 2 10th Refueling and Inspection Outage (Spring 2001) and the Unit 1 12th Refueling and Inspection Outage (Spring 2002).

Any questions regarding this request should be directed to Mr. M. H. Crowthers at (610) 774-7766.

Sincerely,

R. G. Byram

copy: NRC Region I
Mr. R. G. Schaaf, NRC - Sr. Project Manager
Mr. S. L. Hansell, NRC Sr. Resident Inspector
Mr. W. P. Dornsife, PA DEP

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-387

PROPOSED AMENDMENT NO. 235 TO LICENSE NPF-14:
POWER UPRATE
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 1

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-14 dated July 17, 1982.

This amendment contains a revision to the Susquehanna SES Unit 1 Technical Specifications.

PPL Susquehanna, LLC
BY:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 30th day of October, 2000.

Nancy J. Lannen
Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

BEFORE THE
UNITED STATES NUCLEAR REGULATORY COMMISSION

In the Matter of :

PPL Susquehanna, LLC : Docket No. 50-388

PROPOSED AMENDMENT NO. 200 TO LICENSE NPF-22:
POWER UPRATE
SUSQUEHANNA STEAM ELECTRIC STATION
UNIT NO. 2

Licensee, PPL Susquehanna, LLC, hereby files a revision to its Facility Operating License No. NPF-22 dated March 23, 1984.

This amendment contains a revision to the Susquehanna SES Unit 2 Technical Specifications.

PPL Susquehanna, LLC
BY:

R. G. Byram
Sr. Vice-President and Chief Nuclear Officer

Sworn to and subscribed before me
this 30th day of October, 2000.

Nancy J. Lannen
Notary Public

Notarial Seal
Nancy J. Lannen, Notary Public
Allentown, Lehigh County
My Commission Expires June 14, 2004

ATTACHMENT 1

PENDING TECHNICAL SPECIFICATION CHANGES

The following Technical Specification change submittals currently undergoing NRC review have been reviewed by PPL Susquehanna LLC and determined not to be affected by the change proposed here in.

- PLA-5169, Proposed Amendments No. 194 to License NPF-22: MCPR Safety Limits, dated March 20, 2000

- PLA-5148, Proposed Amendment No. 228 to License NPF-14: Revision to the Legend of Technical Specification Figure 3.4.10.1 and Amendment No. 191 to License NPF-22: Revision to the Legend of Technical Specification Figure 3.4.10.1 and Revise a Reference in Technical Specification Section 5.6.5.b, dated January 13, 2000

- PLA-5218, Proposed Amendment No. 232 to License NPF-14 and Proposed Amendment No. 197 To License NPF-22: H_2O_2 Analyzer Penetration dated August 8, 2000.

- PLA-5219, Proposed Amendment No. 231 to License NPF-14 and Proposed Amendment No. 196 to License NPF-22: MSIV Maximum Pathway Leakage dated July 31, 2000.

- PLA-5156, Proposed Amendment No. 230 to License NPF-14: Alternate Reload Analysis Methods and Amendment No. 193 to License NPF-22: Alternate Reload Analysis Methods dated, February 29, 2000.

- PLA-5227, Proposed Amendment No. 233 to License NPF-14: and Proposed Amendment No. 198 to License NPF-22: Relaxation of Surveillance Testing Requirements for Excess Flow Check Valves And Submittal of Pertinent IST Program Relief Requests dated October 4, 2000.

The following NRC issued License Amendments that have not been implemented at PPL Susquehanna LLC have been reviewed and determined to not be affected by the change proposed here in.

- The NRC approved amendments numbered 187 to Facility Operating License No. NPF-12 and Amendment 161 Facility Operating License No. NPF-22 which incorporate changes regarding the Final Response to Generic Letter 94-02; Long Term Stability Solution. These amendments are required to be implemented by November 1, 2001.

ATTACHMENT 2 TO PLA-5212

SAFETY ASSESSMENT

Safety Impact Assessment

Section I – Introduction

The Rated Thermal Power (RTP) of SSES Units 1 and 2 is proposed to be increased by 1.4 percent. The increase in power evaluated and justified herein is obtained by installation of a more accurate feedwater flow measuring system. The Leading Edge Flow Meter (LEFM✓™) supplied by Caldon, Inc., will be installed in both SSES Units.

The increased accuracy of the LEFM✓™ instrument results in an increased accuracy of the calorimetric calculation (< ± 0.6 percent of core thermal power, based on a standard 95% confidence interval evaluation as discussed in Reference 1) versus the currently installed venturi flow instrument (< ± 2. percent of core thermal power). This reduction of uncertainty in the core thermal power calculation allows operation at the proposed increased RTP with no decrease in the confidence level that the actual operating power level is less than the power level required to be assumed in the ECCS accident analyses by 10CFR50, Appendix K.

The improved core thermal power measurement accuracy obviates the need for the full 2 percent power margin required to be assumed in Appendix K analyses, thereby allowing an increase in thermal power available for electrical generation. Concurrently, the LEFM✓™ instrumentation improves the certainty that actual reactor core thermal power remains at or below the value assumed in the Appendix K analyses.

Section II – Change Description

The following two plant modifications are required to implement the proposed changes:

1. The Caldon LEFM✓™ system will be installed in all three feedwater lines on both Unit 1 and Unit 2. The system will be installed by welding a six-foot spool piece into each feedwater line (three in Unit 2 in the spring of 2001 and three in Unit 1 in the spring of 2002). Ultrasonic flow detectors are mounted in each spool piece. The output from each set of ultrasonic detectors is sent to a common signal processing cabinet located near the detectors. The processing cabinet calculates feedwater flow and feedwater temperature. These signals are used directly by the process computer in calculating the operating reactor core thermal power. The LEFM✓™ Topical Report and its supplement (References 1 and 5) contain a detailed discussion of the LEFM✓™ system. Reference 1 was previously approved by the NRC (Reference 2) for use at Comanche Peak. The Topical Report Supplement (ER-157P) discusses an improved measurement system (LEFM CheckPlus™) which supports the additional 0.4% thermal power increase and has not previously been reviewed by the NRC.

2. Due to electrical grid stability concerns, the 3000 amp gang-operated circuit breakers in the SSES 230 KV switchyard will be replaced with 3000 amp independent pole mounted circuit breakers. The purpose for this change is to assure that the electrical grid remains stable at the higher power level for the same series of electrical faults as it is currently evaluated in the SSES FSAR, Table 8.2-1. The replacement of the 3000 amp gang-operated circuit breakers with the 3000 amp independent pole mounted circuit breakers is not required until the increase in rate core thermal power is implemented on Unit 1. The reason is that the increase in rated thermal power on one unit alone is not enough to make the grid unstable under any of the faults evaluated in the FSAR. Further details on the replacement of these circuit breakers can be found in Reference 3, Section 6.1.1.

The following changes to the Unit 1 and Unit 2 Technical Specifications are required to reflect the uprated RTP.

1. The definition of RTP contained in Section 1.1 of the Unit 1 and Unit 2 Technical Specifications is revised to identify a RTP of 3489 MWt.

2. Section 5.6.5 "Core Operating Limits Report (COLR)" of the Unit 1 and Unit2 Technical Specifications is revised to identify that a power level of 3489 MWt (101.4% of current RTP) may be used when the LEFM✓™ system is available to determine the operating core thermal power level. Section 5.6.5 "Core Operating Limits Report (COLR)" is revised to add the LEFM✓™ topical report to the list of analytical methods documents.

The SSES Unit 1 and Unit 2 License paragraphs 2.C.1, "Maximum Power Level" are also revised to identify the revised value of 3489 MWt as the Maximum Core Thermal Power Level.

Mark-up pages of the affected Technical Specifications and License pages are attached for both Units 1 and 2.

In addition, the SSES Units 1 and 2 Technical Requirements Manuals will be revised to include guidance for the case when the LEFM✓™ system becomes unavailable. The guidance will direct the Operations Staff to operate the plant consistent with the accident analyses and the uncertainties associated with the system used to determine the operating RTP.

The SSES FSAR will be revised to reflect both methods of thermal power measurement and the associated instrument uncertainty terms.

Section III – Safety Assessment

Background:

Based on the use of the LEFM✓™ system as described in References 1 and 5, PPL requests a 1.4% increase in rated thermal power (RTP). The 1.4% increase in RTP is justified on the basis of the increased measurement accuracy of the LEFM✓™ system. PPL analysis and Reference 1 conclude that operation at 3489 MWt (101.4% of the current RTP of 3441 MWt) will have approximately the same probability of exceeding the analyzed limit of 3510 MWt as operation at the current RTP using the currently installed venturi flow meter to determine feedwater flow. Therefore, there is no effect on the 10CFR50, Appendix K analysis basis of the SSES units when operating at the increased RTP.

Evaluation

PPL has evaluated the effect on the proposed increase in RTP on safety related components, systems and structures at SSES Units 1 and 2. The effect of the increase was also evaluated with respect to licensed operator performance, training and emergency preparedness.

PPL Licensing Topical Report NE-2000-001 (Reference 3) documents the PPL evaluation of the impact of the 1.4 percent increase in RTP. This report follows the NRC-approved generic format and content of BWR power uprate licensing reports developed by General Electric Company and reported in NEDC-31897-A, "Generic Electric Boiling Water Reactor Power Uprate" (Licensing Topical Report 1, or LTR1).

Detailed evaluations of the reactor and engineered safety features; of power conversion, emergency power and support systems; of environmental issues; of design basis analyses; and of previous licensing evaluations were performed. The PPL Licensing Topical Report demonstrates that both Units 1 and 2 of the Susquehanna Steam Electric Station (SSES) will operate safely with the requested 1.4 percent increase in rated thermal power.

The generating capacity of a BWR such as SSES can be increased by increasing the steam flow through the turbine, up to the limits imposed by the RTP and the turbine and generator designs. The proposed power uprate, discussed herein, increases the licensing power level by decreasing the variability (that is, increasing the accuracy) of the feedwater flow measurement. The power level used for licensing basis and LOCA analyses is not changed. Therefore, any design basis analyses performed using the current licensing basis power level of 3510 MWt is not affected and does not require any further review.

The PPL Licensing Topical Report describes the evaluations performed that show that both Units 1 and 2 of SSES will operate safely with the proposed 1.4 percent increase in licensed thermal power level. Evaluations include those performed on the turbine-generator system to assure that the increased steam flow can be accommodated, and on the balance of plant and support systems to assure that these systems remain within their design basis.

Plant-unique evaluations performed include reviews of design documents, operating data and specific studies prepared for the previous power uprate submittal for the SSES units. These studies and data have been evaluated to confirm that the SSES design is adequate for the 1.4 percent uprated conditions.

Fuel design analyses are based on the use of Siemens Power Corporation (SPC) ATRIUM-10™ fuel bundles in both units. Specifics of the fuel design and analyses methods are delineated in the SSES Technical Specifications. Reference 4 was submitted to propose changes to the Minimum Critical Power Ratio (MCPR) Safety Limits for Unit 2. The changes set forth in Reference 4 are not affected by this proposed license amendment.

The LEFM✓™ Topical Reports (References 1 and 5) detail the applicability of the LEFM✓™ measurement system for boiling water reactors such as SSES. PPL Corporation has reviewed the Topical Report and its Supplement, confirming applicability of these Topical Reports to SSES.

Section IV – References

1. Caldon, Inc., Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997

2. Letter from John N. Hannon, NRC to C. L. Terry, Sr. Vice President and Principle Nuclear Officer, 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 (TAC NOS. MA2298 and MA2299)," March 8, 1999.

3. PPL Licensing Topical Report Number NE-2000-001, "Licensing Topical Report for Power Uprate Resulting From Increased Feedwater Flow Measurement Accuracy," June 2000.

4. PLA-5169, "Proposed Amendment No. 194 to License NPF-22: MCPR Safety Limits," March 20, 2000.

5. Caldon, Inc., Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™ or LEFM CheckPlus™ System,' Revision 0, May 2000.

PROPRIETARY ATTACHMENT 3

HAS BEEN REMOVED PER LICENSEE'S REQUEST

SEE SUPPLEMENTAL SUBMITTAL DATED

02/05/01

PACKAGE ACCESSION NUMBER ML010510373

ATTACHMENT 4 TO PLA-5212

NO SIGNIFICANT HAZARDS CONSIDERATIONS AND ENVIRONMENTAL ASSESSMENT

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

PPL Susquehanna, LLC has evaluated the proposed amendment and determined that it involves no significant hazards consideration. According to 10CFR50.92 (c) a proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility with the propose amendment would not:

- Involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated;
- Create the possibility of a new or different kind of accident from any previously analyzed; or
- Involve a significant reduction in a margin of safety.

PPL Susquehanna, LLC proposes to increase the Rated Thermal Power (RTP) level of Susquehanna Units 1 and 2. The (RTP) increase will be achieved by installation of the Leading Edge Flow Meter (LEFM✓™) supplied by Caldon, Inc. The LEFM✓™ provides improved feedwater flow measurement accuracy and thus improved operation power level certainty.

The determination that the criteria set forth in 10CFR50.92 are met for this amendment as indicated below:

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

   The comprehensive analytical efforts performed to support the proposed change included a review of the NSSS systems and components that could be affected by this change. All systems and components will function as designed and the applicable performance requirements have been evaluated and found to be acceptable.

   The primary loop components (reactor vessel, reactor internals, CRDMs, piping and supports, reactor coolant pumps, etc.) continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the NSSS systems will still perform their intended design functions during normal and accident conditions. The balance of plant systems and components continue to meet their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components. All of the NSSS/BOP interface systems will continue to perform their intended design functions. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level. The current LOCA hydraulic forcing functions are still bounding for the proposed 1.4% increase in power.

Because the integrity of the plant will not be affected by operation at the uprated condition and it can be concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended function, the effects on the remainder of the safety analyses can be assessed. The reduction in the uncertainty allowance provided for the reactor heat balance measurement allows current safety analyses to be used, without change, to support operation at a core power of 3489 MWt. As such, all FSAR Chapter 15 accident analyses continue to demonstrate compliance with the relevant event acceptance criteria. Those analyses performed to assess the effects of mass and energy releases remain valid. The source terms used to assess radiological consequences have been reviewed and determined to bound operation at the 1.4% uprated condition.

Therefore, this proposed amendment does not involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated.

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

   No new accident scenarios, failure mechanisms or single failures are introduced as a result of the proposed change. All systems, structures, and components previously required for the mitigation of a transient remain capable of fulfilling their intended design function. The proposed changes have no adverse effects on any safety-related system or component and do not challenge the performance or integrity of any safety related system.

   Therefore, this proposed amendment does not involve a possibility of a new or different kind of accident from any previously analyzed.

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

   Operation at the uprated power condition does not involve a significant reduction in a margin of safety. Extensive analyses of the primary fission product barriers have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint of compliance of the regulated acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the NRC in Section 5.6.5 of the SSES Technical Specifications, or that are in compliance with all regulatory review guidance and standards.

   Therefore these changes do not involve a significant reduction in margin of safety.

Based upon the above, the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL ASSESSMENT

An environmental assessment is not required for the proposed change because the requested change conforms to the criteria for actions eligible for categorical exclusion as specified in 10 CFR 51.22(c)(9). The requested change will have no impact on the environment. As discussed in the "No Significant Hazards Consideration Evaluation", the proposed change does not involve a significant hazard consideration. The proposed change does not involve a change in the types or increase in the amounts of effluents that may be released off-site. In addition, the proposed change does not involve an increase in the individual or cumulative occupational radiation exposure.

ATTACHMENT 5 TO PLA-5212

TECHNICAL SPECIFICATION MARK-UPs

Note:

The changes shown to Section 5.6.5 for both Unit 1 and Unit 2 are marked up on the current NRC approved affected pages.

UNIT 1 TECHNICAL SPECIFICATION MARK-UPS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3441~~ MWt. 3489

REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME

The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

SHUTDOWN MARGIN (SDM)

SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that:

a. The reactor is xenon free;

b. The moderator temperature is 68°F; and

c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn.

With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

TURBINE BYPASS SYSTEM RESPONSE TIME

The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4. The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. ~~specifically those described in the following documents~~:

Insert 1

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

11. PL-NF-90-001, Supplement 2-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: CASMO-3G Code and ANFB Critical Power Correlation", July 1996.

12. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

13. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

14. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

15. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

16. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

17. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997(P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

Insert 2

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

INSERT 1:

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6 % of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

INSERT 2

40. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report – 80P, March 1997.

41. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0," Engineering Report ER-157P, May 2000.

UNIT 2 TECHNICAL SPECIFICATION MARK-UPS

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3441~~ 3489 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4 The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. ~~specifically those described in the following documents~~:

INSERT 1

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

14. ANF-89-98(P)(A) Revision 1 and Revision 1 Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995.

15. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model," January 1993.

16. XN-NF-80-19(P)(A), Volumes 2, 2A, 2B, and 2C "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," September 1982.

17. XN-NF-80-19(P)(A), Volumes 3 Revision 2 "Exxon Nuclear Methodology for Boiling Water Reactors Thermex: Thermal Limits Methodology Summary Description," January 1987.

18. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

19. EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

INSERT 2

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

INSERT 1:

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6 % of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

INSERT 2

20. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report – 80P, March 1997.

21. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0," Engineering Report ER-157P, May 2000.

ATTACHMENT 6 TO PLA-5212

"CAMERA-READY" TECHNICAL SPECIFICATION PAGES

Note:

The pages provided herein include proposed changes to the Section 5.6.5 pages that NRC has yet to approve. That is, these "camera-ready" pages presume all previously submitted proposed changes that affect these pages are approved by the NRC prior to approval of the changes proposed herein.

UNIT 1 "CAMERA READY"

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during η Surveillance Frequency intervals, where η is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

   2. The Minimum Critical Power Ratio for Specification 3.2.2;

   3. The Linear Heat Generation Rate for Specification 3.2.3;

   4. The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

   5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

   When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6% of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt

(continued)

5.6 Reporting Requirements (continued)

(102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.
2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.
3. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc., September 1986.
4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1 Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.
5. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.
6. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.
7. NEDC-32071P, "SAFER/GESTR-LOCA Loss of Coolant Accident Analysis," GE Nuclear Energy, May 1992.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

17. EMF-1997(P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997(P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

18. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements," September 1999.

19. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

20. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0, "Engineering Report ER-157P, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

UNIT 2 "CAMERA READY"

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3489 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.
TURBINE BYPASS SYSTEM RESPONSE TIME	The TURBINE BYPASS SYSTEM RESPONSE TIME consists of the time from when the turbine bypass control unit generates a turbine bypass valve flow signal

(continued)

5.6 Reporting Requirements (continued)

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. The Average Planar Linear Heat Generation Rate for Specification 3.2.1;

2. The Minimum Critical Power Ratio for Specification 3.2.2;

3. The Linear Heat Generation Rate for Specification 3.2.3;

4 The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and

5. The Shutdown Margin for Specification 3.1.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC.

When an initial assumed power level of 102 percent of rated power is specified in a previously approved method, this refers to the power level associated with the design basis analyses, or 3510 MWt. The power level of 3510 MWt is 100.6% of the rated thermal power level of 3489 MWt. The RTP of 3489 MWt may only be used when feedwater flow measurement (used as input to the reactor thermal power measurement) is provided by the Leading Edge Flow Meter (LEFM✓™) as described in the LEFM✓™ Topical Report and supplement referenced below. When feedwater flow measurements from the LEFM✓™ system are not available, the

(continued)

5.6 Reporting Requirements (continued)

core thermal power level may not exceed the originally approved RTP of 3441 MWt, but the value of 3510 MWt (102% of 3441 MWt) remains the initial power level for the bounding licensing analysis.

Future revisions of approved analytical methods listed in this Technical Specification that are currently referenced to 102% of rated thermal power (3510 MWt) shall include reference that the licensed RTP is actually 3489 MWt. The revisions shall document that the licensing analysis performed at 3510 MWt bounds operation at the RTP of 3489 MWt so long as the LEFM✓™ system is used as the feedwater flow measurement input into the core thermal power calculation.

The approved analytical methods are described in the following documents:

1. PL-NF-90-001-A, "Application of Reactor Analysis Methods for BWR Design and Analysis," July, 1992.

2. XN-NF-80-19(P)(A), Volume 4, Revision 1, "Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads," Exxon Nuclear Company, Inc. June 1986.

3. XN-NF-85-67(P)(A), Revision 1, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel, "Exxon Nuclear Company, Inc., September 1986.

4. XN-NF-80-19(A), Volume 1, and Volume 1 Supplements 1 and 2 (March 1983), and Volume 1, Supplement 3 (November 1990), "Exxon Nuclear Methodology for Boiling Water Reactors: Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc.

5. ANF-524(P)(A), Revision 2 and Supplement 1, Revision 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors", November 1990.

6. ANF-1125(P)(A) and ANF-1125(P)(A), Supplement 1, "ANFB Critical Power Correlation", April 1990.

(continued)

5.6 Reporting Requirements

5.6.5 COLR (continued)

18. XN-NF-79-71(P)(A) Revision 2, Supplements 1, 2, and 3, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," March 1986.

19. EMF-1997 (P)(A) Revision 0, "ANFB-10 Critical Power Correlation," July 1998, and EMF-1997 (P)(A) Supplement 1 Revision 0, "ANFB-10 Critical Power Correlation : High Local Peaking Results," July 1998.

20. PL-NF-90-001, Supplement 3-A, "Application of Reactor Analysis Methods for BWR Design and Analysis: Application Enhancements," September 1999.

21. Caldon, Inc., "TOPICAL REPORT: Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Engineering Report - 80P, March 1997.

22. Caldon, Inc., "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Revision 0, "Engineering Report ER-157P, May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ATTACHMENT 7 TO PLA-5212

PPL SUSQUEHANNA LLC UNIT 1 AND UNIT 2
LICENSE PAGE MARKUPS

UNIT 1 LICENSE PAGE MARKUPS

(3) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

3489

~~PP&L, Inc.~~PPL Susquehanna, LLC ~~(PP&L)~~ is authorized to operate the facility at reactor core power levels not in excess of ~~3441~~ megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational tests, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 185, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. ~~PP&L~~PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment

UNIT 2 LICENSE PAGE MARKUPS

(3) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed neutron sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) ~~PP&L~~PPL Susquehanna, LLC, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

~~PP&L, Inc.~~PPL Susquehanna, LLC (~~PP&L~~) is authorized to operate the facility at reactor core power levels not in excess of ~~3441~~ 3489 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The preoperational test, startup tests and other items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 159, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. ~~PP&L~~PPL Susquehanna, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

For Surveillance Requirements (SRs) that are new in Amendment 151