March 23, 2007

Mr. Britt T. McKinney
Sr. Vice President
and Chief Nuclear Officer
PPL Susquehanna, LLC
769 Salem Blvd., NUCSB3
Berwick, PA 18603-0467

SUBJECT: SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2 - ISSUANCE OF AMENDMENT RE: AVERAGE POWER RANGE MONITOR/ROD BLOCK MONITOR/TECHNICAL SPECIFICATIONS/MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS (ARTS/MELLLA) IMPLEMENTATION (TAC NOS. MC9040 AND MC9041)

Dear Mr. McKinney:

The Commission has issued the enclosed Amendment No. 242 to Facility Operating License No. NPF-14 and Amendment No. 220 to Facility Operating License No. NPF-22 for the Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated November 18, 2005, as supplemented by letters dated November 29, 2006, December 1, 2006, December 15, 2006, January 9, 2007, and March 12, 2007 (PLA-6168 and PLA-6169).

These amendments change the SSES 1 and 2 TSs to implement the ARTS/MELLLA by revising TS 1.1, “Definitions,” TS 5.6.5, “Core Operating Limits Report,” and the surveillance requirement sections of TS 3.3.1.1, “Reactor Protection System Instrumentation,” and TS 3.3.2.1, “Control Rod Block Instrumentation.” The amendments also delete TS 3.2.4, “Average Power Range Monitor Gain and Setpoints,” and its associated references in the TSs. Additionally, the amendments change the method of evaluation for the postulated recirculation line break in the reactor pressure vessel shield annulus region.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular Biweekly *Federal Register* Notice.

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-387 and 50-388

Enclosures:
1. Amendment No. 242 to License No. NPF-14
2. Amendment No. 220 to License No. NPF-22
3. Safety Evaluation

cc w/encls: See next page

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These amendments change the SSES 1 and 2 TSs to implement the ARTS/MELLLA by revising TS 1.1, "Definitions," TS 5.6.5, "Core Operating Limits Report," and the surveillance requirement sections of TS 3.3.1.1, "Reactor Protection System Instrumentation," and TS 3.3.2.1, "Control Rod Block Instrumentation." The amendments also delete TS 3.2.4, "Average Power Range Monitor Gain and Setpoints," and its associated references in the TSs. Additionally, the amendments change the method of evaluation for the postulated recirculation line break in the reactor pressure vessel shield annulus region.

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cc w/encls: See next page

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ADAMS Accession Number: ML070720675 * SE inputs provided by memo. No substantive changes made.

OFFICE	LPLI-1/PM	LPLI-1/LA	SBWB/BC	EICA/BC	ITSB/BC	OGC	LPLI-1/BC(A)
NAME	RGuzman	SLittle	GCranston*	AHowe*	TKobetz*	JMartin	DPickett
DATE	3/14/07	3/15/07	1/22/07 SE DTD	3/20/07 SE DTD	3/19/07 SE DTD	3/22/07	3/22/07

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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. 242
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 November 18, 2005, as supplemented on November 29, 2006, December 1, 2006, December 15, 2006, January 9, 2007, and March 12, 2007 (PLA-6168 and PLA-6169) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 242 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 prior to the startup following the SSES 1 spring 2008 15th refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Douglas V. Pickett, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: March 23, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 242

FACILITY OPERATING LICENSE NO. NPF-14

DOCKET NO. 50-387

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the 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
TS/TOC-1	TS/TOC-1
1.1-5	1.1-5
3.2-7	-
3.2-8	-
3.3-3	TS/3.3-3
3.3-7	TS/3.3-7
3.3-16	TS/3.3-16
3.3-18	TS/3.3-18
3.3-19	TS/3.3-19
3.3-20	TS/3.3-20
TS/5.0-21	TS/5.0-21

PPL SUSQUEHANNA, LLC

ALLEGHENY ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-388

SUSQUEHANNA STEAM ELECTRIC STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
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 PPL Susquehanna, LLC, dated November 18, 2005, as supplemented on November 29, 2006, December 1, 2006, December 15, 2006, January 9, 2007, and March 12, 2007 (PLA-6168 and PLA-6169) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 220 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 prior to the startup following the SSES 2 spring 2007 13th refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Douglas V. Pickett, Acting Chief
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and
Technical Specifications

Date of Issuance: March 23, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. NPF-22

DOCKET NO. 50-388

Replace the following page of the License with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

REMOVE	INSERT
3	3

Replace the following pages of the 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.

TS/TOC-1	TS/TOC-1
1.1-5	1.1-5
3.2-7	-
3.2-8	-
3.3-3	TS/3.3-3
3.3-7	TS/3.3-7
3.3-16	TS/3.3-16
3.3-18	TS/3.3-18
3.3-19	TS/3.3-19
3.3-20	TS/3.3-20
TS/5.0-21	TS/5.0-21

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 242 TO FACILITY OPERATING LICENSE NO. NPF-14

AND AMENDMENT NO. 220 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 388

1.0 INTRODUCTION

By letter dated November 18, 2005, Agencywide Documents Access and Management System (ADAMS) Accession No. ML053420162, as supplemented by letters dated November 29, 2006 (ML063420106), December 1, 2006 (ML063460050), December 15, 2006 (ML063610192), January 9, 2007 (ML070170458), and March 12, 2007 (PLA-6168 and PLA-6169), PPL Susquehanna, LLC (PPL, the licensee), requested changes to the Technical Specifications (TSs) for Susquehanna Steam Electric Station, Units 1 and 2 (SSES 1 and 2). The proposed changes would change the SSES 1 and 2 TSs to implement the Average Power Range Monitor/Rod Block Monitor/Technical Specification/Maximum Load Line Limit Analysis (ARTS/MELLLA) by revising TS 1.1, "Definitions," TS 5.6.5, "Core Operating Limits Report," and the surveillance requirement sections of TS 3.3.1.1, "Reactor Protection System Instrumentation," and TS 3.3.2.1, "Control Rod Block Instrumentation." The amendments would also delete TS 3.2.4, "Average Power Range Monitor Gain and Setpoints," and its associated references in the TSs. Additionally, the amendments would change the method of evaluation for the postulated recirculation line break in the reactor pressure vessel shield annulus region.

This licensing amendment request (LAR) is in support of the SSES Power Range Neutron Monitoring System (PRNMS) installation which, in turn, is part of the power ascension plan for the Extended Power Uprate. The amendment would allow the expanded operating domain resulting from the implementation of ARTS/MELLLA.

The current flow-biased Rod Block Monitor (RBM) is being replaced by a power dependent RBM through the upgraded PRNMS by installation of the General Electric (GE) Nuclear Measurement Analysis and Control (NUMAC) equipment, requested by PPL in an application dated June 27, 2005 (ML051870394) and approved by the staff safety evaluation (SE) in a letter dated March 3, 2006 (ML060540429) for SSES Units 1 and 2. (Herein referred to as the NUMAC PRNMS Upgrade.)

This change, from the flow-biased RBM to the power dependent RBM, will require new trip setpoints. The Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoints would be revised to permit operation in the MELLLA region. The flow-biased APRM scram and rod block trip setdown requirement would be replaced by the more direct power and flow dependent thermal limits to reduce the need for APRM gain adjustments and to allow more direct thermal limits administration during operation. Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated core flow and result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle.

The licensee's proposed TS changes for the SSES PRNMS installation is planned in two phases. Phase 1, the NUMAC PRNMS Upgrade, included a PRNMS installation that retained the previous "non-Average Power Range Monitor/Rod Block Monitor/ Technical Specifications" ["non-ARTS"] version of the RBM. Phase 2 (this amendment) includes minor modification to the PRNMS equipment to incorporate the "ARTS" logic in the RBM and implement associated TS setpoint modifications for RBM and APRM equipment.

For SSES Unit 1, Phase 1 was incorporated during the spring 2006 outage with Phase 2 to follow at a separate time, while, for SSES Unit 2, Phase 1 and Phase 2 are planned to be incorporated during the spring 2007 outage. Note: Phase 1 documentation, the NUMAC PRNMS Upgrade, did not include licensing documentation required for ARTS implementation.

The supplemental letters dated November 29, 2006, December 1, 2006, December 15, 2006, January 9, 2007, and March 12, 2007 (PLA-6168 and PLA-6169), provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on February 14, 2006 (71 FR 7810).

2.0 REGULATORY EVALUATION

The regulatory requirements and guidance which the NRC staff considered in its review of the application are as follows:

1. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 establishes the fundamental regulatory requirements with respect to the domestic licensing of nuclear production and utilization facilities. Specifically, Appendix A, "General Design Criteria for Nuclear Power Plants," provides, in part, the necessary design fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety.

2. General Design Criterion (GDC)-10, "Reactor design," requires that the reactor core and associated coolant, control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

3. GDC-12, "Suppression of reactor power oscillations," states that the reactor core and associated coolant, control and protection systems shall be designed to assure that

power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

4. GDC-20, “Protection system functions,” requires the protection system be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

5. GDC-22, “Protection system independence,” requires that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in the loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

6. GDC-25, “Protection system requirements for reactivity control malfunctions,” requires that the protection system be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

7. 10 CFR 50.36, “Technical specifications,” provides the regulatory requirements for the content required in a licensee’s TS. 10 CFR 50.36 states, in part, that where a limiting safety system setting (LSSS) is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded.

8. Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation,” describes a method acceptable to the NRC staff for complying with the NRC regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

9. Regulatory Issue Summary (RIS) 2006-17, “NRC Staff Position on the Requirements of 10 CFR 50.36 regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,” dated August 24, 2006, provides an approach, acceptable to the NRC staff, for addressing issues that could occur during testing of LSSSs and which, therefore, may have an adverse effect on equipment operability.

3.0 TECHNICAL EVALUATION

3.1 Background - Boiling-Water Reactor Systems

SSES 1 and 2 is a standardized boiling-water reactor/4 (BWR/4) plant which originally included minimum critical heat flux ratio (MCHFR) as the thermal margin criterion. This MCHFR basis included operating, overpower, and safety limit values that, along with a design power peaking factor, translate to the rated power load line, 108% load line, and 120% load line, respectively.

Therefore, these average power range monitor (APRM) flow-biased setpoint values originate with a deterministic overpower. Later, with the change to the minimum critical power ratio (MCPR) thermal margin basis under which SSES 1 and 2 was originally licensed, studies concluded that the MCPR safety limit (MCPRSL) would be met for the design-basis transients with the peaking restrictions being conservative for off-rated transients. The SSES 1 and 2 Final Safety Analysis Report (FSAR) includes the results of rated power transients, which establish the MCPR Operating Limit (MCPROL).

The proposed APRM/Rod Block Monitor/Technical Specification (ARTS) changes replace the power peaking factor restrictions with power and flow dependent limits. However, the flow-biased APRM rod block and scram remain as design features. Other changes that have taken place as part of ARTS implementation for the APRM flow-biased functions include a reduction in the slope, from 0.66 to 0.58 (to improve the ability to reach the rated load line at lower flow), the addition of setpoint uncertainties to the nominal values, and the restoring of margin to the operating load line for the maximum extended load line limit analysis (MELLLA). The original 0.66 flow-biased slope reflected the general relationship between power and flow of a 2 to 3 ratio, but using drive flow was deemed too conservative for low flows, thus the 0.58 slope was justified for the current licensed extended load line limit analysis (ELLLA).

Plants with full ARTS/MELLLA including increased core flow (ICF) implementation are: Hatch Units 1 and 2, Cooper, Pilgrim, Fermi, Monticello, Brunswick Units 1 and 2, Peach Bottom Units 2 and 3, Browns Ferry Units 2 and 3, and Duane Arnold (no ICF). Plants with partial (i.e., excluding the rod block monitor (RBM) modifications) ARTS/MELLLA including ICF implementation are: Dresden Units 2 and 3, Quad Cities Units 1 and 2, Vermont Yankee and Hope Creek.

SSES 1 and 2 have performed 2 power uprates. The first uprate, termed a stretch uprate, increased the licensed thermal power by approximately 4.5%. The second uprate of 1.4% was a result of improved instrumentation allowing a reduction in the uncertainty in thermal power, termed an Appendix K uprate. The key thermal power levels are as follows:

1. Original Licensed Thermal Power (OLTP) is 3293 mega-watt thermal (MWt).
2. Stretch Uprate Licensed Thermal Power is 3441 MWt.
3. Current Licensed Thermal Power (CLTP) and Rated Thermal Power (RTP) is the Appendix K Uprate Power, which is 3489 MWt.
4. Analysis Thermal Power is 1.02 x 3441 MWt or 3510 MWt.

Note that the Appendix K uprate reduced the power uncertainty to 1.006; therefore, the analysis power level remains the same, namely 1.006 x 3489 MWt or 3510 MWt.

Many factors restrict the flexibility of a BWR during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition. Some of the factors that limit plant flexibility in achieving rated power are:

1. the currently licensed allowable operating power/flow map;

2. the APRM flow-biased simulated thermal power-high scram and flow-biased neutron flux-high control rod block setdown requirements; and

3. the RBM flow-referenced rod block trip.

Once rated power is achieved, periodic control rod and core flow adjustments must be made to compensate for reactivity changes due to xenon effects and fuel burnup.

SSES 1 and 2 are currently licensed to operate in the ELLLA region up to approximately the 108% rod line based on current licensed power and the ICF region up to 108% core flow, which results in a core flow window of 87% to 108% at RTP.

A further expansion of the operating domain (MELLLA) and implementation of ARTS would allow rated power to be maintained over a wider core flow range, thereby reducing the frequency of control rod manipulations that require power maneuvers to implement. Expansion of the operating domain beyond the current power-flow map requires changes to the APRM and RBM trip functions discussed below.

The APRM flow-biased trip setpoint varies as a function of reactor recirculation loop flow but is clamped such that it is always less than the APRM neutron flux-high setpoint. The APRM flow-biased neutron flux-high rod block function is designed to avoid conditions that would require reactor protection system (RPS) action if allowed to proceed. The APRM rod block setting is selected to initiate a rod block before the APRM neutron flux-high scram setting is reached.

PPL proposed to replace the flow-biased RBM trips with power-dependent trips. The RBM is designed to prohibit erroneous withdrawal of a control rod during operation at high power levels. This prevents local fuel damage during a Rod Withdrawal Error (RWE) event.

TS Limiting Condition for Operation (LCO) 3.2.4 currently requires the APRM flow-biased scram and rod block trip setpoints to be reduced (setdown) when the Fraction of Rated Thermal Power (FRTP) is less than the Core Maximum Fraction of Limiting Power Density (CMFLPD). The trip setdown requirement ensures that margins to the fuel cladding safety limit are preserved during operation at other than rated conditions. As an alternative to adjusting the APRM setpoints, the APRM gains may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD. According to PPL, the SSES 1 and 2 normal operating practice is to adjust APRM gains when required to meet LCO 3.2.4. Each APRM channel is typically bypassed while the required gain adjustment is made.

The basis for this “APRM trip setdown” requirement originated from the now obsolete Hench-Levy MCHFR thermal limit criterion. Improved methodologies have subsequently been developed and approved by the NRC staff to provide more effective alternatives to the setdown requirement.

The current SSES 1 and 2 reactor recirculation system suction line break (RSLB) blowdown mass and energy release profiles for annulus pressure (AP) loads were calculated based on normal operation at the 100% power/100% core flow point of the power/flow map using the method described in NEDO-24548 (Reference 4). The Reference 4 methodology was

conservative because it used a simple bounding approximation for a complex blowdown process.

For the MELLLA operating conditions, PPL evaluated the mass and energy releases over a range of power/flow conditions since the mass and energy release from the RSLB can be higher due to the lower enthalpy in the downcomer at off-rated conditions. The results determined that the MELLLA minimum pump speed point has the highest mass and energy release profile.

Using the General Electric (GE) LAMB [computer code] computer program, a more realistic blowdown mass and energy release profile for the MELLLA minimum pump speed point was determined. LAMB has been used in several plant amendment applications to calculate the blowdown mass flow rate and energy profile in the event of an RSLB and has been used for amendment applications for power/flow map extension (MELLLA) associated with BWR extended power uprates. In addition, credit was taken for the lower operating steam dome pressure at the lower power level. With these changes, the mass and energy release profile at the MELLLA minimum pump speed point are bounded by the profile in the SSES 1 and 2 AP load design calculation of record.

The NRC staff reviewed the safety analyses and systems response evaluations submitted to justify SSES 1 and 2 operation in the expanded MELLLA region, as discussed in Attachment 3 of Reference 1. The plant-specific, fuel independent evaluations, such as containment response, were performed based on the current hardware design and applicable plant geometry for SSES 1 and 2. The fuel-dependent analyses, such as the limiting anticipated operational occurrences (AOOs), the MCPR calculations, and the reactor vessel overpressure protection analysis, were performed using the SSES 2 Cycle 13 core design with Framatome Advanced Nuclear Power (FANP) ATRIUM-10 fuel. These analyses are to be performed each operating cycle as part of the standard reload design process. The NRC staff's evaluation is discussed further in section 3.3.

3.2 Background - Instrumentation and Controls

The current RBM computed the analog average of all assigned unbypassed Local Power Range Monitors (LPRMs) in much the same manner as the APRM. If the average of the RBM input reading was less than the reference APRM signal, then an automatic RBM gain adjustment occurred such that the average RBM reading was equal to, or greater than the APRM reading (this gain adjustment factor could never be less than one). This comparison and potential RBM gain adjustment occurred whenever a control rod was selected. There was a momentary rod block while the gain adjustment was made. This gain was held until a new control rod was selected. For the RBM to fulfill its intended function, changes in the RBM signal(s) must correlate closely with the thermal margin changes during control rod withdrawal. The current RBM signals do not always correlate well with thermal margin changes during control rod withdrawal.

With the implementation of ARTS/MELLLA, a more direct trip logic than is currently provided would be implemented. The RBM takes input from the LPRMs surrounding the rod that is selected for withdrawal and an average of these readings at the time of rod selection is calculated. A "nulling" operation is then performed which establishes the pre-rod motion value.

This value is normalized to 100. This nulling establishes the fixed reference level. As the rod is pulled, the LPRM readings increase and subsequent average values from the same set of LPRMs are calculated. The value is then divided by the average at the time of nulling and is multiplied by 100 to give the instantaneous RBM reading. If this RBM reading exceeds the trip setpoint, a rod block is issued that protects against rod-withdrawal errors. The reference level is the level the RBM is automatically calibrated to upon control rod selection. The fundamental logic and setpoint changes to implement ARTS and supporting analyses and justifications are delineated in the NRC staff's evaluation below.

To enable the RBM signals to correlate more closely with the thermal margin changes, a new LPRM assignment scheme was studied by GE. A new LPRM assignment scheme, being implemented as part of the ARTS/MELLLA, groups the LPRMs to best achieve: (1) similarity of channel responses; (2) higher response to rod motion; (3) less restrictive MCPR limits; and (4) greater tolerance of LPRM failures.

3.3 Method of Analysis

The NRC-approved or industry-accepted computer codes and calculational techniques are used in the ARTS/MELLLA analyses. A list of the Nuclear Steam Supply System (NSSS) computer codes used in the evaluations is provided in Table 1-1 (Attachment 3 of Reference 1).

The safety analyses and system evaluations performed to justify operation in the MELLLA region consist of a non-fuel-dependent portion and a fuel-dependent portion that is fuel cycle-dependent. In general, the limiting AOO, MCPR calculation, and the reactor vessel overpressure protection analysis are fuel-dependent. These analyses are based on a representative Unit 2 Cycle 13 core design using ATRIUM-10 fuel. Subsequent unit cycle-specific analyses will be performed in conjunction with the reload licensing activities. The non-fuel-dependent evaluations, such as containment response, are based on the current plant design and configuration. The limiting AOOs, identified in the SSES 1 and 2 FSARs, were reviewed for the MELLLA region. PPL states that for the fuel-dependent evaluations of reactor pressurization events, these reviews indicate that there is a small difference in the MCPROL for operation in the MELLLA region and the CLTP condition (100% of CLTP/108% of RCF). The operating limit is calculated on a cycle-specific basis to bound the entire operating domain. The analysis results indicate that performance in the MELLLA region is within allowable design limits for overpressure protection, loss-of-coolant accident (LOCA), containment dynamic loads, flow-induced vibration, and reactor internals structural integrity. The response to the Anticipated Transient Without Scram (ATWS) demonstrates that PPL meets the licensing criteria in the expanded MELLLA operating domain. Therefore, the NRC staff finds PPL's method of analysis for SSES 1 and 2 MELLLA operation acceptable.

3.4 Fuel Thermal Limits

The potentially limiting AOOs and accident analyses were evaluated to support SSES 1 and 2 operation in the MELLLA region with ARTS off-rated limits. The nominal conditions for the power/flow state points chosen for the review of AOOs include the MELLLA region and the current licensed operating domain for SSES 1 and 2.

The core-wide AOOs included in the current SSES 2 Cycle 13 reload licensing analyses and the SSES 1 and 2 FSAR were examined for operation in the ARTS/MELLLA region (including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MELLLA region and were reviewed as part of the ARTS program development:

a. Generator Load Rejection with No Bypass (LRNBP) event;

b. Turbine Trip with No Bypass (TTNBP) event;

c. Feedwater Controller Failure (FWCF) maximum demand event;

d. Loss of Feedwater Heating (LFWH) event;

e. Fuel Loading Error (FLE) event;

f. Inadvertent High-Pressure Coolant Injection (HPCI) Startup event;

g. Recirculation Flow Increase (RFI) event.

The initial ARTS/MELLLA assessment of these events concluded that for plant-specific applications, only the TTNBP, LRNBP, and FWCF events need to be evaluated at both rated and off-rated power and flow conditions. For conservatism, PPL combined the LRNBP and TTNBP events as one event LRNBP/TTNBP.

The analytical method used by FANP for the SSES 1 and 2 evaluations was consistent with the basis used in Reference 5. The results from the SSES 1 and 2 cycle-specific analyses of LFWH, FLE, and HPCI events showed that these events were non-limiting for the following reasons:

- The LFWH evaluation for SSES 2 Cycle 13 considered the flow range for the MELLLA region. The results showed that the LFWH event is not limiting for SSES and the effect

  of MELLLA on the LFWH severity is sufficiently small that the LFWH remains not limiting for MELLLA. The limiting results for SSES 2 Cycle 13, analyses performed at 100% CLTP showed that there is a large margin for MCPROL (1.27 for the LFWH versus 1.36 for the LRNBP/TTNBP and 1.35 for FWCF). The LFWH at off rated conditions is bounded by the FWCF. However, the LFWH event is analyzed on a cycle-specific basis.

- The FLE is a static event that is most limiting at maximum power. Therefore, this event was also not considered in the determination of the off-rated limits.

- The HPCI evaluation for SSES 2 Cycle 13 considered the flow range for the MELLLA region. The limiting result for SSES 2 Cycle 13 analyses performed at 100% CLTP showed a large margin for MCPROL (1.26 for HPCI versus 1.36 for LRNBP/TTNBP and 1.35 for FWCF). The HPCI event tends to be more severe as the initial power decreases (ratio of HPCI flow to initial feedwater flow increases). However, at low initial powers, the subcooling due to FWCF bounds the subcooling due to HPCI. Consequently, the HPCI event was not considered in the determination of the off-rated limits.

- The RFI event is most limiting at reduced flow conditions. The RFI event is protected by the flow-dependent MCPR limits and Linear Heat Generation Rate multipliers (LHGRFAC) which are established to protect a slow flow excursion.

Extensive transient analyses at a variety of power and flow conditions were performed for SSES 2 Cycle 13. These evaluations are applicable for operation in the MELLLA region. The evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between rated power and the power level where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed (P_{Bypass}). The analytical value of P_{Bypass} for SSES 1 and 2 is 30% of CLTP. The second power range is between P_{Bypass} and 25% of CLTP. No thermal limit monitoring is required below 25% of CLTP, per SSES 1 and 2 TS 3.2.

GE Part 21 communication SC04-15, "Turbine Control System Impact in Transient Analysis," identified that certain turbine control systems, (e.g., the Power/Load Unbalance (PLU) feature), may not be relied on to trip the turbine and cause the turbine control valve fast closure and reactor scram down to power levels corresponding to P_{Bypass} (30%). SC04-15 was reviewed by PPL and it was found that the PLU for SSES 1 and 2 is set to actuate down to 40% power. However, it was concluded that generator output breakers do provide a turbine control valve (TCV) fast closure signal which initiates reactor scram, and that this action is independent of reactor power. Also, the review concluded that under the load reject event, the generator output breaker will get a signal to initiate the TCV Fast Closure signal. This action serves as a backup to the PLU TCV Fast Closure signal and will initiate a reactor scram for power levels down to P_{Bypass}.

SSES 1 and 2 cycle-specific evaluations were performed to establish power-dependent MCPR limits and LHGRFAC multipliers for use in the two power ranges (above P_{Bypass} and below P_{Bypass}). Also, SSES 1 and 2 cycle-specific evaluations were performed to establish the flow-dependent MCPR limits and LHGRFAC multipliers.

Since the cycle-specific reload fuel analyses will determine the limits for rated and applicable off-rated conditions, and application of the methodology is demonstrated by the analyses performed for the current operating cycle, this approach is acceptable to the NRC staff.

3.5 Rod Withdrawal Error Analysis

The RWE transient is currently analyzed during the reload fuel licensing analysis for SSES 1 and 2. The FANP RWE methodology which is currently employed for SSES 1 and 2 is discussed in Reference 10. The RWE transient is hypothesized as an inadvertent reactor operator initiated withdrawal of a single control rod from the core. Withdrawal of a single control rod has the effect of increasing local power and core thermal power which lowers the MCPR and increases the Linear Heat Generation Rate (LHGR) in the core limiting fuel rods. The RWE transient is terminated by control rod blocks which are initiated by the RBM system.

The RWE analyses are performed with the NRC-approved FANP MICROBURN-B2 reactor simulator code (Reference 11). The ANFB-10 critical power correlation (Reference 12) is used to calculate the MCPR values for the FANP ATRIUM-10 fuel.

The purpose of the RWE analysis is to determine bounding values for the power-dependent RWE MCPR limit as a function of RBM setpoint. The calculations are performed at representative power and flow conditions to cover the ARTS RBM power ranges with analytical low, intermediate and high power setpoints of 30%, 65%, and 85%. The analyzed reactor conditions are 100% power, 85% power, 65% power, and 40% power.
A non-statistical ARTS RWE analysis was performed using SSES 1 and 2 and fuel related input and methods consistent with the Unit 2 Cycle 13 licensing analysis. Analyzed conditions support plant operation in the MELLLA region of the power and flow operating map. The RWE MCPR results were tabulated as a function of the RBM setpoints for the ATRIUM-10 fuel. The bounding RWE MCPR results are shown in Table 4-3 (Attachment 3 of Reference 1) as a function of percent rated power/flow and the RBM setpoint values. The RWE MCPR values in Table 4-3 are for an MCPRSL value of 1.09. For other MCPRSL values, the RWE MCPR values are adjusted by the difference in the MCPRSL. The bounding RWE MCPR results at all power levels are shown as a function of RBM setpoint in Figure 4-8 (Attachment 3 of Reference 1).

The upgraded performance of the ARTS RBM system significantly reduces the severity of the RWE event when compared to other AOO events for selected RBM setpoints. The Local Power Range Monitor (LPRM) assignments make the ARTS RBM system more sensitive to rod withdrawals. The RWE MCPR operating limits can be compared with the limiting cycle-specific transient ($MCPR_{P)}$ limit and MCPRSL to verify that the RWE is a non-limiting event for a specific set of RBM setpoints. For example, representative SSES 2 Cycle 13 $MCPR_P$ and RWE MCPR limits are shown on Figure 4-9 (Attachment 3 of Reference 1). A comparison of the curves on Figure 4-9 shows that the RWE MCPR limit is bounded by the $MCPR_P$ limit for the RBM setpoints of 108%, 113%, and 118%. At low reactor powers, the RWE event is not limiting as shown on Figure 4-9, but the RBM system is not required to be in service below the RBM low power setpoint. The representative power-dependent RBM analytical setpoints (without filter) of 108, 113, and 118 percent are shown in Figure 4-10 (Attachment 3 of Reference 1).

The SSES 1 and 2 RWE calculations demonstrate that the transient LHGR limits for the ATRIUM-10 fuel are not exceeded in an unblocked RWE event, and therefore, cladding strain induced fuel damage and fuel melting are precluded.

Based on the analyses provided by PPL and given that NRC-approved methodologies were used by PPL, the NRC staff concludes that the SSES 1 and 2 RWE analysis with the proposed Nuclear Measurement Analysis and Control Power Range Neutron Monitor System (NUMAC PRNMS) and ARTS/MELLLA implementation at CLTP conditions are acceptable.

3.6 Vessel Overpressure

The main steam isolation valve (MSIV) closure with a flux scram (MSIVF) event is used to determine the compliance to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. This event was previously analyzed at the 100.6%(power)/108%(flow) state point for the SSES 2 Cycle 13 reload licensing transient analysis. This is a cycle-specific calculation performed at 100.6% of CLTP and the maximum licensed core flow (maximum flow is limiting for this transient for SSES 1 and 2). Since high core flow is limiting and because the implementation of ARTS/MELLLA does not change the maximum core flow, ARTS/MELLLA does not affect the overpressure protection analysis.

3.7 Thermal Hydraulic Stability

Stability criteria are established to demonstrate compliance with the GDC 12 requirements in order to assure that specified acceptable fuel design limits (i.e., MCPRSL) are not exceeded. The analysis and methods used to demonstrate compliance with the stability acceptance criteria are documented in the NRC approved methodology NEDO-31960A (Reference 6).

SSES 1 and 2 is operating with ATRIUM-10 fuel. SSES 1 and 2 implemented the Option III (Reference 7) stability solution beginning in Cycle 12 for SSES 2. This section presents the effect of the MELLLA operating domain expansion on stability for SSES 1 and 2.

Option III is a detect and suppress solution which combines closely spaced LPRM detectors into “cells” to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed Oscillating Power Range Monitor (OPRM) cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have installed new hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. Of these algorithms, only the period based detection algorithm (PBDA) is officially credited in the Option III licensing basis (Reference 6 and 7). This algorithm provides an instrument setpoint designed to trip the reactor before an oscillation can increase to the point where the MCPRSL is exceeded.

The Option III stability reload licensing basis specifies the methodology to calculate the limiting MCPROL required to protect the MCPRSL for instability events. Selection of an appropriate instrument setpoint is based on the MCPROL to provide adequate MCPRSL protection.

The PBDA setpoint calculation requires the use of the regional Delta Critical Power Ratio/Initial Critical Power Ratio versus Oscillation Magnitude (DIVOM) curve, determined for SSES 1 and 2 on a cycle-specific basis. Because the BWR Owners’ Group (BWROG) DIVOM guidelines (Reference 8) specifically provides a suitable DIVOM slope on a cycle-specific basis, the Option III solution is fully capable of supporting SSES 1 and 2 operation in the MELLLA domain with ATRIUM-10 fuel.

The SSES 1 and 2 MELLLA operating domain expansion complies with the current licensing requirements for stability Option III. The Option III solution is fully capable of supporting SSES 1 and 2 operation in the MELLLA domain with ATRIUM-10 fuel because the actual cycle core design is used to produce a suitable DIVOM slope. Should the Option III system be declared inoperable, the interim corrective action (ICA) regions effectiveness evaluation is fully capable of supporting continued operation since the evaluation will be performed on a cycle-specific basis.

Based on the analyses provided by PPL and given that approved methodologies were used, the NRC staff concludes that the SSES 1 and 2 thermal hydraulic stability characteristics with the proposed ARTS/MELLLA implementation at CLTP conditions are acceptable.

3.8 Loss-of-Coolant-Accident Analysis

The Emergency Core Cooling System (ECCS) is designed to provide protection against postulated 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. The maximum average planar linear heat generation rate (MAPLHGR) operating limit is based on the most limiting LOCA and ensures compliance with the ECCS acceptance criteria in 10 CFR 50.46.

The current licensing basis LOCA analysis for SSES 1 and 2 supports operation in the MELLLA domain. The changes associated with ARTS have no impact on the FANP LOCA analyses. The initial assembly planar power is equal to the MAPLHGR limit and the initial assembly average power is set based on a low value for MCPR operating limit.

An evaluation was performed with ATRIUM-10 fuel to determine the ECCS-LOCA analysis effects of SSES 1 and 2 operation in the MELLLA region. The limiting Design-Basis Accident (DBA) was evaluated to show that the estimated Peak Clad Temperature (PCT) remained below the acceptance limits. The maximum local oxidation was less than 2%. The core-wide metal-water reaction was less than 0.2%. The results demonstrate that operation in the MELLLA domain will meet all of the ECCS-LOCA acceptance criteria. Therefore, there are no ECCS-LOCA analysis related plant operating restrictions due to implementation of ARTS/MELLLA.

Since the determination of the sensitivity of the ECCS-LOCA evaluations to operation in the MELLLA domain shows compliance with the acceptance criteria, the NRC staff has determined that no additional operating restrictions would be required for ARTS/MELLLA operation at the CLTP.

3.9 Anticipated Transient without Scram (ATWS)

The basis for the current ATWS requirements is 10 CFR 50.62. This regulation includes requirements for an ATWS Recirculation Pump Trip (RPT), an Alternate Rod Insertion (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate. The purpose of the ATWS analysis is to demonstrate that these systems are adequate for operation in the MELLLA region. This is accomplished by performing a plant-specific analysis in accordance with approved licensing methodology, to demonstrate that ATWS acceptance criteria are met for operation in the MELLLA region.

The NRC staff reviewed the SSES 1 and 2 specific analysis that was performed using the approved licensing methodology (Reference 9) to demonstrate compliance with 10 CFR 50.62 ATWS requirements. The analysis assumed the CLTP with the minimum MELLLA core flow (81.9% of RCF) is the limiting operating condition. The limiting ATWS events, MSIV Closure, and pressure regulator failure open (PRFO), were re-evaluated with ARI assumed to fail, requiring the operator to initiate SLCS injection for shutdown. The adequacy of the margin to the SLCS relief valve lifting, as described in NRC Information Notice 2001-13, “Inadequate Standby Liquid Control System Relief Valve Margin,” was included in this assessment.

The limiting ATWS event for peak vessel pressure, PCT, suppression pool heatup, and containment pressure is the PRFO. The peak vessel bottom pressure for this event is 1288 pounds per square-inch gauge (psig) at end of cycle (EOC), which is below the ATWS vessel overpressure protection criterion of 1500 psig.

The highest calculated peak suppression pool temperature is 207.1 °F at EOC, which is below the ATWS limit of 210 °F. The highest calculated peak containment pressure is 16.5 psig at EOC, which is below the ATWS limit of 53 psig. Thus, the containment criteria for ATWS are met.

Coolable core geometry is ensured by meeting the 2200 °F PCT and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. The highest calculated PCT is 1420 °F, which is less than the ATWS limit. The fuel cladding oxidation is significantly less than the 17% local limit.

The maximum SLCS pump discharge pressure and timing depends primarily on the safety valve mode setpoints for the Crosby Dual Mode Safety/Relief Valves (DS/RVs). The maximum SLCS pump discharge pressure, following SLCS pump start, during the limiting ATWS event is approximately 1389 psig. This value is based on a peak reactor vessel lower plenum pressure of 1217 pounds per square-inch absolute (psia) that occurs during the loss-of-offsite power (LOOP) event at EOC. With a nominal SLCS relief valve setpoint of 1500 psig, there is a margin of approximately 111 psi between the peak SLCS pump discharge pressure and the relief valve nominal setpoint. Therefore, there is adequate margin to prevent the SLCS relief valve from lifting. This issue is also addressed in NRC Information Notice 2001-13.

The NRC staff concludes, based on its review of the above analyses, that PPL meets the ATWS mitigating features stipulated in 10 CFR 50.62, and that the results of the ATWS analyses for MELLLA operation at the CLTP would meet the ATWS acceptance criteria.

3.10 Technical Specification Changes for ARTS/MELLLA

3.10.1 Boiling-Water Reactor Systems

The NRC staff reviewed the proposed changes to the SSES 1 and 2 TS that are identified in PPL's November 18, 2005 submittal. The changes include deletion of the current setdown requirement, and new power and flow-dependent MCPR and MAPLHGR limits. The proposed TS changes include the following:

TS 3.2.4, "APRM Gain and Setpoint," which includes requirements for flow-biased APRM scram and rod block trip setpoint setdown, and the associated TS Bases would be deleted. The following additional changes would be made to reflect the deletion of TS 3.2.4:

a. The TS Table of Contents would be revised.

b. The definition for Maximum Fraction of Limiting Power Density (MFLPD) would be deleted from TS Section 1.1.

c. References to TS 3.2.4 will be deleted from existing SR 3.3.1.1.2, which is proposed to be changed to SR 3.3.1.1.3 by the PRNMS submittal (Reference 3). The associated TS Bases will also be changed.

d. Reference to the APRM Gain and Setpoints for Specification 3.2.4 would be deleted from the 5.6.5 Core Operating Limits Report (COLR).

APRM and RBM allowable values would be revised as follows to permit operation in the MELLLA operating domain:

a. The APRM Simulated Thermal Power-High allowable value in TS Table 3.3.1.1-1, Reactor Protection System Instrumentation, would be changed to:

$$0.62W+64.2\%$$

The equation in footnote (b) to Table 3.3.1.1-1, Reactor Protection System Instrumentation would be changed to:

$$0.62(W-\Delta W) + 64.2\%$$

The sentence in footnote (b) that defines the value of ΔW will be removed to make the footnote consistent with NUREG-1433, Rev. 3 (ΔW is described in Attachment 3, Section 1 of Reference 1).

The APRM high flow clamped setpoints would not be changed.

b. SR 3.3.2.1.4 would be revised to require that the various ARTS based power dependent RBM power ranges are enabled at the appropriate power levels. The associated Intermediate Power Range Setpoint and High Power Range Setpoint would be specified in the COLR since the setpoints must be reconfirmed or modified on a cycle-specific basis.

The surveillance and operability requirements for each RBM power range would be modified from those shown in NUREG-1433 to clarify the requirement for each range. Namely, the applicable limits (i.e., Low Power Range limit, Intermediate Power Range limit, and High Power Range limit) will be effective when the power is at or above the lower power limit for each range (the limit on permitted local power increase becomes more restrictive as the RBM power range increases). PPL's November 18, 2005 submittal states that the original wording in NUREG-1433 implied that the transition from each RBM “range” to the next had to occur at an exact % of RTP whereas the real requirement is that above the lower “threshold” values, the more restrictive limit needs to be in force (i.e., the limit associated with the higher power range). The SR is also written based on APRM Simulated Thermal Power (STP) input, the digital signal that is actually used in the NUMAC RBM.

Consistent with this change, the note stating that neutron detectors are excluded is deleted because the signals used for the SR do not originate from the

detectors. The purpose of this SR is only to confirm the correct setup of the RBM. These additional surveillance and operability requirements clarifications result in no functional changes in the equipment performance or operational limits.

c. TS Table 3.3.2.1-1 Control Rod Block Instrumentation would continue to provide information about the function, modes of operation and surveillance requirements for the Rod Block Monitor.

Table 3.3.2.1-1 would first be modified to change from a flow-biased RBM to a power-dependent RBM consistent with NUREG-1433, “Standard Technical Specifications - General Electric Plants, BWR/4,” Revision 3. Then, the allowable values for the RBM trip for Low Power Range-Upscale, Intermediate Power Range-Upscale and High Power Range-Upscale and the associated Intermediate Power Range Setpoint and High Power Range Setpoint would be specified in the COLR. Also, the MCPR limits applicable to the operability of the RBM would be specified in the COLR. The RBM trip, power range and MCPR values are calculated on a cycle-specific basis. Section 5.6.5 “Core Operating Limits Report (COLR)” ‘Item a’, would also be changed to state what RBM information shall be in the COLR.

In addition to the above changes to Table 3.3.2.1-1 the RBM downscale function would also be deleted. The deletion of the RBM Downscale Function is intended to simplify the TS by deleting a function that has no significant value due to differences between the original analog equipment and the replacement digital system. Further justification is provided in Attachment 4 of Reference 1.

A change will be made in the method of evaluation for the postulated RSLB in the Reactor Pressure Vessel (RPV) shield annulus region. For the RSLB at the MELLLA minimum pump speed point, the mass and energy release profile will be calculated using the LAMB computer program in lieu of the current methodology described in NEDO-24548.

Note that one additional administrative change would be required to change wording in TS Bases Section B 3.2.1 to state that analyses are performed in the MELLLA domain instead of the current ELLLA domain.

The proposed changes are consistent with the guidelines contained in NUREG-1433, Rev. 3. These changes allow the implementation of the thermal limits portion of the ARTS improvement protection program and the MELLLA expanded operating domain. PPL's safety analyses examined the same areas as previous ARTS and MELLLA submittals, which have been reviewed by the NRC staff. The methods used have been previously approved and the results of the analyses fall within accepted limits. The NRC staff concludes that the results submitted by PPL justify the proposed TS changes to SSES 1 and 2 for operation at the CLTP, based on the analyses reviewed and compared with the prior approvals.

3.10.2 Instrumentation and Controls

By safety evaluation (SE) dated September 5, 1995, the NRC staff approved GE Licensing Topical Report (LTR) NEDC-32410P, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function." This LTR addressed the full scope of the modification to replace the power range monitoring portion of an analog neutron monitoring system in GE BWRs with a GE NUMAC PRNMS including an OPRM. In this LTR, the NRC staff approved proposed TS changes for APRM reactor trip and rod-block protective functions. By letter dated August 15, 1997, the NRC staff approved Supplement 1 to NEDC-32410P (herein both referred to as NUMAC PRNMS LTR), which includes TS requirements for an OPRM and clarifies issues related to the APRM. The NUMAC PRNMS LTR describes in detail, the generic NUMAC PRNMS design and several plant-specific variations and plant-specific actions.

As part of ARTS/MELLLA, the current flow-biased RBM is being replaced by a power-dependent RBM. The change from the flow-biased RBM to the power-dependent RBM would require new setpoints for RBM functions.

3.10.2.a Implementation of the ARTS logic into the NUMAC PRNMS

Management by the licensee of the implementation of ARTS logic into the NUMAC PRNMS will be via changes to the previously approved NUMAC PRNMS Upgrade. In these changes, in Attachment 4 to the June 27, 2005, PRNMS application, the licensee provided a description in the form of 7 specific equipment changes to the NUMAC PRNMS. The licensee stated that all software changes necessary for these equipment and firmware revisions will undergo full verification and validation activities fully equivalent to those performed for the Phase 1 installation. In the March 3, 2006, NUMAC PRNMS Upgrade SE, the NRC staff found the software design process, applicable to the Phase 1 installation, acceptable. With the major equipment changes listed and the commitment that the ARTS verification and validation activities will be fully equivalent to the original software design process, the NRC staff finds the implementation of the ARTS logic into the NUMAC PRNMS acceptable.

3.10.2.b Technical Specification Changes for ARTS/MELLLA

TS 3.2.4, “APRM Gain and Setpoints,” is being deleted. The licensee stated that this specification is no longer needed since approved methodologies provide more effective alternatives to the replacement. As a result of this deletion, the licensee provided a list of changes to the TS, in terms of deleting the references, to the deleted TS, and Section 1.1. “Definitions.” The staff finds the deletions to references of TS 3.2.4, and the applicable definition, acceptable because these are administrative changes

TS Table 3.3.1.1-1 for Function 2.b, “Simulated Thermal Power (STP) - High,” is being revised. Specifically, the Allowable Value (AV) for two-loop operation (TLO) and single-loop operation (SLO) is being changed. These AVs include the same conservatisms to account for, among others, testing and calibration errors as the original AV values using the GE setpoint methodology described in NEDC-31336 P-A, approved by the staff and documented in the associated SE and are, therefore, acceptable.

TS Table 3.3.2.1-1, Function 1.a is being revised by replacing both TLO and SLO flow dependent AVs with three power-dependent functions and AVs. The new functions will be 1.a, “Low Power Range - Upscale,” 1.b, “Intermediate Power Range - Upscale,” and 1.c, “High Power Range - Upscale.” The proposed TS changes to the RBM functions are consistent with those in the NUMAC PRNMS LTR.

Related to the addition of the three power-dependent RBM functions, Surveillance Requirement (SR) 3.3.2.1.4 will be revised to require that these RBM power ranges are enabled at the appropriate power levels. Namely, the applicable limits (i.e., Low Power Range limit, Intermediate Power Range limit, and High Power Range limit) will be effective when the power is at or above the lower power limit for each range (the limit on permitted local power increase becomes more restrictive as the RBM power range increases).

Consistent with this change, the note (stating that neutron detectors are excluded) is deleted because the signals used for the SR do not originate from the detectors. In PPL’s supplemental letter dated December 15, 2006, it was identified that the original LAR submittal, dated November 18, 2005, inadvertently failed to delete the applicability of SR 3.3.2.1.4 from the RBM Inop Function in Table 3.3.2.1-1. Additionally, the RBM Inop Function in Table 3.3.2.1-1 is being renumbered from 1.b to 1.d. The NRC staff finds this acceptable since the notes and SR are now consistent with those recommended by the NUMAC PRNMS LTR.

TS Table 3.3.2.1-1, Function 1.c., “Downscale,” is being deleted. PPL states that the RBM Downscale Function was used to detect substantial reductions in the RBM local flux after a "null" is completed (a "null" occurs after a new rod selection). This function, in combination with the RBM Inop Function, was intended in the original system to detect problems with or abnormal conditions in the RBM equipment and system. Unlike other neutron monitoring system downscale Functions (e.g., the APRM downscale), there are no normal operating conditions that are intended to be detected by the Downscale Function. In the original analog RBM, the inclusion of the Downscale Function, in addition to the Inop Function, had some merit in that the analog equipment had some failure modes that could result in a reduction of signal, but not a full failure. PPL further states the Inop Function is enhanced in the NUMAC RBM by the use of automatic self-test and other internal logic to increase the detectability of failures and abnormal conditions that can occur in the digital equipment, and to directly include these in the RBM Inop Function. No credit is taken for the RBM Downscale Function in the establishment of the RBM upscale trip setpoints or AVs. The NRC staff concurs that this function can be removed since this was an intended function inherent to the analog system and provides no added value if included in the new digital system.

3.11 PPL Response to RAIs regarding Limiting Safety System Settings

In its supplemental letter dated December 1, 2006, PPL provided a response to the NRC staff’s Request for Additional Information (RAI) dated October 19, 2006. In the RAI, the NRC staff requested information to support NRC assessment of the acceptability of the LAR setpoint changes, and the issues related to, the NRC letter to the Nuclear Energy Institute (NEI) Setpoints Methods Task Force (SMTF) dated September 7, 2005 (ML052500004).

PPL identified that the new RBM power-dependent functions (Low Power Upscale, Intermediate Power Upscale, and High Power Upscale) are the only TS functions affected by this change that are considered to be LSSSs upon which a safety limit (SL) has been placed. These rod block functions (the three power-dependent AVs) will be credited in the accident analysis with protecting the MCPR Safety Limit specified in the TS 2.1.1.2, for an RWE event. An RWE event is designated as an AOO. AOO events could be mitigated with a highly reliable non-safety system. The RBM is a highly reliable system and is designed to prohibit erroneous withdrawal of a control rod so that local fuel damage does not occur. The RBM consists of two separate channels either of which can provide the required rod block monitor function, although one channel can be bypassed at any time. The RBM is designed to criteria approaching that of a safety system; the RBM meets most of the requirements of the single failure criterion but does not fully meet the seismic and independence criteria of IEEE Std. 279-1971.

Both RBM modules are mounted in the same bay of a seismically qualified panel with physical space between the RBM modules, but without physical barriers. Although the panel and some of the RBM module components are seismically qualified to maintain physical integrity, the RBM function was not verified during the seismic testing. Therefore, the RBM modules are not qualified to operate during or after a seismic event. Because the RBM modules are located in the same bay without physical barriers, the RBM monitors do not fully comply with the independence criteria of IEEE Std. 279-1971.

The RBM was designed, manufactured, and qualified to the same standards as the PRNMS. Procurement and factory acceptance was in accordance with the PRNMS specification for both safety-related and non-safety related equipment.

The RBM is automatically and continuously self-tested to ensure operability during normal operation. Upon a self-test fault, the RBM channel is automatically placed in trip and the operator is alerted via alarm. In the event of a loss of power, the RBM channel would default to a condition that would provide a rod block.

Based on the RBM's robust design as described above, the operability requirements for the RBM in TS 3.3.2.1 and the proposed RBM TS requirements, described in Section 3.12 of this SE, there is reasonable assurance that the RBM will perform its necessary function to mitigate the consequences of an RWE event.

The APRM STP - High Function, being revised, is not SL-related, and it does provide defense-in-depth to the APRM Fixed Neutron Flux - High Function. This function is being retained in the TSs since it is part of the RPS design and the NRC-approved licensing basis. The RBM Downscale Function which is not taken credit for in any safety analysis is being deleted as noted above in 3.2.2.4. The NRC staff agrees that the RBM power-dependent setpoints are the only TS functions removed or altered by this LAR that are considered an SL-related LSSS. The NRC staff also agrees that the remaining TS functions listed above are not SL-related.

PPL stated that the RBM power-dependent functions are established by the SSES 1 and 2 fuel vendor, Framatome Advanced Nuclear Power (FANP) using their methodology on a cycle-specific basis. The associated setpoints (AV, Nominal Trip Setpoint (NTSP), and Process Setpoint) were determined using GE setpoint methodology. The GE setpoint methodology is

described in NEDC-31336 P-A, “General Electric Setpoint Methodology,” September 1996 and has been approved by the NRC as documented in the associated SE. The new SSES 1 and 2 RBM setpoints were calculated using this methodology and the results provided to PPL in GE document "0000-0039-3825 Susq A-M-T506-RBM-Calc-2005,” Revision 0, dated October 2005." This document was attached as an Appendix to the December 1, 2006, RAI response.

Since the RBM power-dependent functions (Table 3.3.2.1-1, Functions 1.a, 1.b, and 1.c) are SL-related LSSSs, PPL is adding two notes: (I) and (j) to the references to SR 3.3.2.1.7 for Functions 1.a, 1.b, and 1.c to implement the setpoint related TSs described in the September 7, 2005, letter.

Note (I) reads, “If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative with respect to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.” Note (I) requires evaluation of channel performance for the condition where the RBM as-found setting for the channel setpoint is not the NTSP, but is conservative with respect to the AV.

Note (j) reads, “The instrument channel setpoint shall be reset to the Nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The NTSP and the methodology used to determine the NTSP is specified in the SSES Final Safety Analysis Report.” Note (j) requires that the as-left setting for the RBM be returned to the NTSP. If the as-left setting cannot be returned to the NTSP, then the channel shall be declared inoperable. Note (j) also requires that the NTSP and the NTSP methodology are to be contained in a document controlled by 10 CFR 50.59.

PPL explained that the Setpoint Control Program at SSES 1 and 2 provides the engineering, procedural and corrective action processes that are used to ensure the channel is capable of performing its specified safety function by implementing these features:

- As-left settings are controlled under the Surveillance and Preventive Maintenance Program.
- As-found settings outside acceptable tolerances are controlled per their 10 CFR Part 50, Appendix B, Criterion XVI corrective action program.
- The Surveillance Testing Program establishes the administrative controls for Surveillance testing .
- Process setpoint changes are controlled by the Engineering Change Process.
- The Setpoint Calculation Methodology provides requirements to ensure compliance to applicable guides and standards per the SSES FSAR.

The NRC staff finds that sufficient measures are in place, through implementation of these controls, to ensure that the associated setpoints are capable of performing their safety functions.

PPL also provided the associated TS Bases that reflect the proposed TS changes as an attachment to its application. The TS Bases changes are consistent with PPL’s proposed plant-specific TS changes. The NRC staff has no objections to the Bases changes presented in PPL's application.

3.12 TS LCO Action Times

The Completion Time for LCO 3.3.2.1, Required Action A.1, when one RBM channel is inoperable is being changed from 5 days to 24 hours. The Completion Time for LCO 3.3.2.1, Required Action B.1, when Required Action and associated Completion Time of Condition A are not met or two RBM channels are inoperable is being changed from 48 hours to 1 hour. Since PPL is taking credit for the RBM to protect the MCPRSL for an RWE event, the completion times for Required Actions A.1 and B.1 are being reduced to the completion times in NUREG-1433, "Standard Technical Specification General Electric Plants, BWR/4, Rev. 3." The 24-hour completion time for Required Action A.1 is based on the low probability of an event occurring coincident with a failure in the remaining operable channel. The 1-hour completion time for Required Action B.1 is intended to allow the operator time to evaluate and repair any discovered inoperabilities, and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels. The NRC staff finds that the completion times are reasonable with respect to protecting the MCPRSL, since the time to evaluate inoperable channels, and either repair or trip inoperable channels has been previously reviewed and analyzed by the NRC.

In the case of the failure of both RBM channels while an operator continues to withdraw a rod, minor localized fuel damage could occur. However, at this stage the event would no longer be considered an AOO, but would be considered an accident. During this phase, minor localized fuel damage is considered acceptable, and the reactor protection system would protect against any major fuel damage.

3.13 Conclusion

The NRC staff has reviewed PPL's LAR application and supplemental information. The review of TS changes in this SE is performed to evaluate the changes that would be required to support the ARTS/MELLLA implementation at SSES 1 and 2. This review covered the ARTS/MELLLA application for the CLTP.

Based on its review, the NRC staff concludes that the proposed TS changes are acceptable because the safety analysis supporting actual operation in the ARTS/MELLLA regimes at the CLTP has been reviewed as acceptable, and the NRC staff concludes that operation will not endanger the public health and safety.

The NRC staff also concludes that for phase 2, the ARTS/ MELLLA logic changes are consistent with the NRC-staff approved guidance in the NUMAC PRNMS LTR, and no exceptions have been taken to the safety bases for the NUMAC PRNMS LTR. The NRC staff also finds the identification and bases of the SL-related LSSSs being removed, altered or added by this LAR meet the requirements of 10 CFR 50.36(c)(1)(ii)(A).

For setpoints that are not SL-related, the NRC staff finds acceptable measures, controls and procedures in place to ensure that the associated instrument channels are capable of performing their specified safety functions in accordance with applicable design requirements and associated analyses.

The NRC staff also finds that the LCO completion times are reasonable with respect to protecting the MCPRSL, the time to evaluate inoperable channels and either repair or trip inoperable channels.

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

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes SRs. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 7810). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

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

1. Letter PLA-5931 from B. T. McKinney (PPL) to NRC, “Susquehanna Steam Electric Station Proposed License Amendment Numbers 279 for Unit 1 Operating License No. NPF-14 and 248 for Unit 2 Operating License No. NPF-22 ARTS/MELLLA Implementation,” dated November 18, 2005.

2. Letter PLA-6136 from B. T. McKinney (PPL) to NRC, “Susquehanna Steam Electric Station Proposed License Amendment No. 279 for Unit 1 Operating License No. NPF-14 & No. 248 for Unit 2 Operating License No. NPF-22, ARTS/MELLLA Implementation Response to Request for Additional Information,” dated November 29, 2006.

3. Letter PLA-5880 from Britt T. McKinney (PPL) to NRC, “Susquehanna Steam Electric Station Proposed License Amendment Numbers 272 for Unit 1 Operating License No. NPF-14 and 241 for Unit 2 Operating License No. NPF-22 Power Range Neutron Monitor System Digital Upgrade,” dated June 27, 2005.

4. GE Nuclear Energy, “Technical Description - Annulus Pressurization Load Adequacy Evaluation,” NEDO-24548, dated January 1979.

5. 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, dated June 1986.

6. NEDO-31960A and NEDO-31960A, Supplement 1, “BWR Owners Group Long Term Stability Solutions Licensing Methodology,” dated November 1995.

7. NEDO-32465-A, “Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications,” dated August 1996.

8. GE-NE-0000-0028-9714-R0, “Plant-Specific Regional Mode DIVOM Procedure Guideline,” dated June 14, 2004.

9. NEDC-24154P-A, “Qualification of the One Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 - Volume 4),” dated February 2000.

10. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis,” Exxon Nuclear Company, dated March 1983.

11. EMF-2158(P)(A) Revision 0, “Seimens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,” Seimens Power Corporation, dated October 1999.

12. EMF-1997(P)(A) Revision 0, “ANFB-10 Critical Power Correlation,” Seimens Power Corporation, dated July 1998.

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Date: March 23, 2007