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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 **Docket Nos. 50-387**
PLA-5931 **and 50-388**

Reference: *PPL Letter PLA-5880, Britt T. McKinney (PPL) to USNRC "Susquehanna Steam Electric Station Proposed License Amendment Numbers 272 for I 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*

Pursuant to 10 CFR 50.90, PPL Susquehanna, LLC (PPL), hereby requests approval of the following amendments to the Susquehanna Steam Electric Station (SSES) Unit 1 and Unit 2 Technical Specifications (TS), as described in the enclosure. The proposal would change Technical Specifications for TOC 3.2.4, TOC 3.3.1.3, "Table of Contents"; 1.1, "Definitions"; 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," SR 3.3.1.1.3, "Surveillance Requirements - Reactor Protection System Instrumentation;" Table 3.3.1.1-1, "Reactor Protection System Instrumentation"; SR 3.3.2.1.4, "Surveillance Requirements - Control Rod Block Instrumentation"; Table 3.3.2.1-1, "Control Rod Block Instrumentation"; and 5.6.5, "Core Operating Limits Requirements."

The proposed changes reflect an expanded operating domain resulting from the implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Analysis (ARTS/MELLLA). 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 current flow-biased Rod Block Monitor (RBM) would also be replaced by a power dependent RBM implemented through the referenced proposed upgrade to a digital Power Range Neutron Monitor System (PRNMS), Reference 1. The change from the flow-biased RBM to the power dependent RBM would also require new trip setpoints. In addition, the flow-biased APRM scram and rod block trip setdown requirement would be replaced by 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 other than rated conditions.

A001

PPL also proposes to make changes to the methods used to evaluate the Annulus Pressurization, (AP), mass blowdown and early release resulting from the postulated Recirculation Suction Line Break (RSLB).

Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated core flow. Operation in the MELLLA region can result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle. Replacement of the APRM scram and rod block trip setdown requirement will improve reliability and provide more direct protection of plant safety.

As demonstrated in the enclosed evaluation, the proposed amendments do not involve a significant hazard consideration.

Precedent licensing submittals have been approved by NRC for other licensees. These precedents are discussed in the Background section of the Licensee Evaluation of proposed changes.

To support the power ascension plan for Extended Power Uprate, PPL requests approval of the proposed ARTS/MELLLA amendments by November 23, 2006. PPL requests that the approved amendment be issued with the Unit 1 and 2 amendments effective upon implementation. This is based on the following implementation plan. ARTS/MELLLA implementation is contingent on NRC approval and PPL implementation of PRNMS. The PRNMS submittal was provided to NRC on June 27, 2005, with a requested approval date by February 1, 2006. PPL plans to implement PRNMS on Unit 1 during the Spring 2006 Outage. After NRC approval, PPL plans to implement ARTS/MELLLA on Unit 1. PPL plans to implement PRNMS and ARTS/MELLLA on Unit 2 during the Spring 2007 Outage.

Attachment 1 is the Technical Specifications mark-up. Attachment 2 is the associated Technical Specification Bases mark-up provided for information. Attachment 3 contains the safety analyses in support of the proposed changes. Attachment 4 is a revision for ARTS Implementation to the Plant-Specific Evaluations, provided in Reference 1, required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A).

There are no regulatory commitments associated with the proposed changes.

The need for the changes has been discussed with the SSES NRC Project Manager.

The proposed changes have been reviewed by the SSES Plant Operations Review Committee and by the Susquehanna Review Committee. In accordance with 10 CFR 50.91(b), PPL Susquehanna, LLC is providing the Commonwealth of Pennsylvania with a copy of this proposed License Amendment request.

If you have any questions or require additional information, please contact Mr. John M. Oddo at (610) 774-7596.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: ______________

B. T. McKinney

Enclosure:
PPL Susquehanna Evaluation of the Proposed Changes

Attachments:
Attachment 1 Proposed Technical Specification Changes (Mark-up)
Attachment 2 Changes to Technical Specifications Bases Pages (Mark-up, Provided for Information)
Attachment 3 Susquehanna Steam Electric Station Units 1 and 2 APRM/RBM/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)
Attachment 4 Revisions to Plant-Specific Evaluations required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A) for ARTS Implementation

Copy: NRC Region I
Mr. B. Bickett, NRC Sr. Resident Inspector
Mr. R. V. Guzman, NRC Project Manager
Mr. R. Janati, DEP/BRP

ENCLOSURE TO PLA-5931

PPL SUSQUEHANNA EVALUATION OF PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS: TOC 3.2.4, TOC 3.3.1.3, "TABLE OF CONTENTS"; 1.1, "DEFINITIONS"; 3.2.4, "AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS"; SR 3.3.1.1.3, "SURVEILLANCE REQUIREMENTS", TABLE 3.3.1.1-1 – "REACTOR PROTECTION SYSTEM INSTRUMENTATION"; SR 3.3.2.1.4 , "SURVEILLANCE REQUIREMENT - CONTROL ROD BLOCK INSTRUMENTATION"; TABLE 3.3.2.1-1, "CONTROL ROD BLOCK INSTRUMENTATION"; 5.6.5, "CORE OPERATING LIMITS REQUIREMENTS" FOR ARTS/MELLLA IMPLEMENTATION

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PPL EVALUATION

SUBJECT: PPL SUSQUEHANNA EVALUATION OF PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS: TOC 3.2.4, TOC 3.3.1.3, "TABLE OF CONTENTS"; 1.1, "DEFINITIONS"; 3.2.4, "AVERAGE POWER RANGE MONITOR (APRM) GAIN AND SETPOINTS"; SR 3.3.1.1.3, "SURVEILLANCE REQUIREMENTS", TABLE 3.3.1.1-1 - "REACTOR PROTECTION SYSTEM INSTRUMENTATION"; SR 3.3.2.1.4 , "SURVEILLANCE REQUIREMENTS", TABLE 3.3.2.1-1, "CONTROL ROD BLOCK INSTRUMENTATION"; 5.6.5, "CORE OPERATING LIMITS REQUIREMENTS" FOR ARTS/MELLLA IMPLEMENTATION

1. DESCRIPTION

The proposal would change the PPL Susquehanna (PPL) Technical Specifications contained in Appendix A to the Operating License to reflect an expanded operating domain resulting from implementation of Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA). The Average Power Range Monitor (APRM) flow-biased scram setpoint and the APRM flow-biased rod block trip setpoints would be revised to permit operation in the MELLLA operating domain. The current flow-biased Rod Block Monitor (RBM) would also be replaced by a power dependent RBM. The change from the flow-biased RBM to the power dependent RBM is possible with the proposed digital upgrade to the Power Range Neutron Monitor System (PRNMS). The power dependent RBM would require new trip setpoints. The flow-biased APRM scram and rod block trip setpoint setdown requirements would be replaced by more direct power and flow dependent thermal limits to reduce the need for APRM setpoint or gain adjustments and to allow more direct thermal limits administration. In addition, the methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) would be changed.

To support the power ascension plan for Extended Power Uprate, PPL requests approval of the proposed ARTS/MELLLA amendments by November 23, 2006. PPL requests that the approved amendment be issued with the Unit 1 and 2 amendments effective upon implementation. This is based on the following implementation plan. ARTS/MELLLA implementation is contingent on NRC approval and PPL implementation of PRNMS. The PRNMS submittal was provided to NRC on June 27, 2005, with a requested approval date by February 1, 2006. PPL plans to implement PRNMS on Unit 1 during the Spring 2006 Outage. After NRC approval, PPL plans to implement ARTS/MELLLA on Unit 1. PPL plans to implement PRNMS and ARTS/MELLLA on Unit 2 during the Spring 2007 Outage.

A separate License Amendment Request has been submitted to NRC for approval of implementation of the Alternative Source Term. Implementation of the Alternative Source Term has no impact on this ARTS/MELLLA submittal or vice versa, because the scope of the accident analyses are mutually exclusive.

2. PROPOSED CHANGE

The marked-up pages for the proposed changes to the Technical Specifications (TS) are included in Attachment 1 of this submittal. A PPL Susquehanna NRC submittal dated June 27, 2005 (Reference 1), currently in NRC review, requested approval for a PRNMS Upgrade. Some of the Technical Specification changes proposed in the PRNMS submittal are affected by this ARTS/MELLLA submittal. One administrative change to the Table of Contents is needed to delete Section 3.3.1.3 for the PRNMS submittal which is currently under NRC review. The markups in Attachment 1 show the current Technical Specifications and the proposed PRNMS changes indicated by Ⓟ, with the effect of the proposed ARTS/MELLLA changes indicated by Ⓐ. This ARTS/MELLLA proposal would change the following Technical Specifications (TS) and analysis methodology:

1. 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 1). 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).

2. 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. ΔW is described in Attachment 3, Section 1.

The APRM high flow clamped setpoints would not be changed.

b. Surveillance Requirement (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). The original wording in NUREG-1433 implied that the transition from each RBM "range" to the next had to occur at an exact % of Rated Thermal Power (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. Technical Specification 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 Rod Block Monitor 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 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 Technical Specification 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.

3. A change will be made in the method of evaluation for the postulated Recirculation Suction Line Break (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 (Reference 2).

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 requirements of NUREG-1433, Standard Technical Specifications and with changes previously approved by the NRC for other licensees as described in Section 3 of this Enclosure. Operation in the MELLLA region will provide improved power ascension capability by extending plant operation at rated power with less than rated core flow. Operation in MELLLA can also result in the need for fewer control rod manipulations to maintain rated power during the fuel cycle, thereby improving operational flexibility. Replacement of the APRM flow-biased trip setdown requirement with power and flow dependent MCPR and LHGR thermal limits will improve reliability and provide more direct protection of plant safety. The proposed changes will reduce the need for APRM gain adjustments and allow more direct thermal limits administration.

3. BACKGROUND

Many factors restrict the flexibility of a Boiling Water Reactor (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:

- the currently licensed allowable operating power/flow map;
- the APRM flow-biased simulated thermal power-high scram and flow-biased neutron flux-high control rod block setdown requirements; and
- 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.

Susquehanna is currently licensed to operate in the Extended Load Line Limit Analysis (ELLLA) region up to approximately the 108% rod line based on Current Licensed Power and the Increased Core Flow (ICF) region up to 108% core flow, which results in a core flow window of 87% to 108% at rated thermal power (References 3, 4, 5 and 6).

A further expansion of the operating domain (MELLLA) and implementation of ARTS would allow for more efficient and reliable power ascensions and 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 described below.

APRM and RBM Trip Setpoints

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.

The flow-biased RBM trips will be replaced by 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 single rod withdrawal error.

APRM Trip Setpoint Setdown Requirement

TS Limiting Condition for Operation (LCO) 3.2.4 currently requires the APRM flow-biased scram and rod block trip setpoints to be reduced 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. PPL 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 setdown requirement originated from the Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion. Improved methodologies have subsequently been developed to provide more effective alternatives to the setdown requirement.

Reactor Recirculation System Suction Line Break (RSLB) Annulus Pressurization (AP) Loads

The current RSLB blowdown mass and energy release profiles for 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 2). The Reference 2 methodology was conservative since it used a simple bounding approximation for a complex blowdown process.

For the MELLLA operating conditions, mass and energy releases were evaluated 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 offrated conditions. The results determined that the MELLLA minimum pump speed point has the highest mass and energy release profile.

Using the GE LAMB 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 licensing applications to calculate the blowdown mass flow rate and energy profile in the event of an RSLB and has been used for licensing 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 SSES AP load design calculation of record.

Industry and PPL Prior Experience

Plants with full ARTS/MELLLA including Increased Core Flow (ICF) implementation are: Hatch Units 1 and 2, Duane Arnold (no ICF), Cooper, Pilgrim, Fermi, Monticello, Brunswick Units 1 and 2, Peach Bottom Units 2 and 3, and Browns Ferry Units 2 and 3. Plants with partial ARTS/MELLLA including ICF implementation are: Dresden Units 2 and 3, Quad Cities Units 1 and 2, and the Vermont Yankee. The Hope Creek Generating Station has a partial ARTS submittal currently under review.

SSES has performed 2 power uprates. The first uprate, termed a Stretch Uprate, increased the licensed thermal power by approximately 4.5%. (References 3, 4, and 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 (References 6). The key thermal power levels are as follows:

- The Original Licensed Thermal Power (OLTP) is 3293 MWt.
- The Stretch Uprate Licensed Thermal Power is 3441 MWt.
- The Current Licensed Thermal Power (CLTP) and Rated Thermal Power (RTP) is the Appendix K Uprate Power, which is 3489 MWt.
- The 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 = 3510 MWt.

Power Range Neutron Monitor System (PRNMS) Digital Upgrade Submittal

Note that the PPL PRNMS Digital Upgrade submittal contained a "Plant Specific Evaluation, required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-3241P-A)," as Enclosure Section 7 of Reference 1. Attachment 4 of the enclosure to this ARTS/MELLLA submittal provides a description of the impact of ARTS/MELLLA on the PRNMS Plant Specific Evaluation.

4. TECHNICAL ANALYSIS

The proposed changes would reflect an expanded operating domain resulting from implementation of Maximum Extended Load Line Limit Analysis. In addition, the flow-biased APRM scram and rod block trip setpoint setdown requirements would be replaced by more direct power and flow dependent thermal limits to reduce the need for manual setpoint adjustments and to allow more direct thermal limits administration.

Safety analyses performed in support of the proposed changes are described in Attachment 3.

Attachment 3, Section 1.0, Introduction, and Section 2.0, Overall Analysis Approach, provide a description and background for the implementation of ARTS/MELLLA at SSES Units 1 and 2. The content of Sections 1.0 and 2.0, relative to fuel dependent evaluations, describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases. However, the assumptions and conclusions described in Section 1.0 and 2.0 for fuel dependent evaluations are based upon a representative core of Framatome ANP (FANP) Inc. ATRIUM™-10[(1)] fuel.

[(1)]ATRIUM™-10 is a trademark of Framatome ANP

The content of Attachment 3, Sections 1.0 and 2.0, relative to non-fuel dependent evaluations, describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases and reflect the SSES configuration. The assumptions and conclusions described in Sections 1.0 and 2.0 relative to non-fuel dependent evaluations are applicable for SSES.

These Safety analyses include evaluations of fuel performance events (Sections 3.0 and 4.0), vessel overpressure protection (Section 5.0), thermal-hydraulic stability (Section 6.0), the loss-of-coolant accident (Section 7.0), containment response (Section 8.0), reactor internals integrity (Section 9.0), an anticipated transient without scram (Section 10.0), steam dryer and separator performance (Section 11.0), high energy line break (Section 12.0), and descriptions of planned testing and training (Sections 13.0 and 14.0).

The technical analysis discussion below summarizes or supplements the information in Attachment 3.

Attachment 3, Sections 3.0, Fuel Thermal Limits, 4.0, Rod Withdrawal Error, and 5.0, Vessel Overpressure Protection describe particular aspects of the implementation of ARTS/MELLLA for SSES Unit 2 Cycle 13. These sections describe fuel dependent evaluations. The content of the sections describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases. The assumptions and conclusions for the fuel dependent evaluations are based upon SSES Unit 2 Cycle 13 with FANP ATRIUM™-10 Fuel.

Attachment 3, Section 6.0, Thermal Hydraulic Stability describes this particular aspect of the implementation of ARTS/MELLLA. The content of this section describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases.

Attachment 3, Section 7.0, Loss-Of-Coolant Accident Analysis describes this particular aspect of the implementation of ARTS/MELLLA. This section describes a fuel dependent evaluation. The content of this section describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases and reflects the SSES plant configuration.

Attachment 3, Section 8.0, Containment Response, describes a non-fuel dependent evaluation. The section describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases and reflects the SSES plant configuration. The assumptions and conclusions described are applicable for SSES.

Attachment 3, Section 9.0, Reactor Internals Integrity, describes non-fuel dependent evaluations with the exception of Section 9.1, Reactor Internal Pressure Differences, which contains some fuel-dependent aspects. The section describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases and reflects the SSES plant configuration. The assumptions and conclusions described are applicable for SSES. Although Section 9.1 has aspects that are fuel dependent, further fuel dependent evaluation is not required. The section describes that the existing SSES ELLLA bases are bounding relative to the MELLLA application and therefore no specific fuel evaluations are required to justify the ARTS/MELLLA bases.

Attachment 3, Section 10.0, Anticipated Transient Without Scram (ATWS), describes an evaluation that can be considered fuel dependent. The ATWS evaluation described in Section 10.0 is a SSES plant specific evaluation; however, the evaluation uses a representative equilibrium core. The content of the section describes the approach PPL is taking to justify and implement the ARTS/MELLLA bases.

Attachment 3, Sections 11.0, Steam Dryer and Separator Performance, and 12.0, High Energy Line Break, describe non-fuel dependent evaluations relative to the effects of the ARTS/MELLLA bases. The sections describe the approach PPL is taking to justify and implement the ARTS/MELLLA bases and reflect the SSES plant configuration. In addition, the assumptions and conclusions described are applicable for SSES.

Attachment 3, Sections 13.0 and 14.0 describe the planned pre-operational testing and operations training, which will be performed in support of ARTS/MELLLA implementation.

ARTS/MELLLA Implementation

The expanded operating domain includes the operating domain changes for ARTS/MELLLA consistent with approved operating domain improvements for other BWRs. The current licensed ELLLA power-flow upper boundary is modified to include the operating region bounded by the rod line which passes through the 100% of Current Licensed Thermal Power (CLTP) at 81.9% of Rated Core Flow (RCF). The power-flow region that is above the current licensed ELLLA boundary is referred to as the MELLLA region.

The current power range monitoring system will be replaced with the proposed (Reference 1) upgrade to a digital PRNMS. As part of ARTS/MELLLA, the current flow-biased RBM would also be replaced by a power dependent RBM. The change to the power dependent RBM is possible with the proposed upgrade to a digital PRNMS. The change from the flow-biased RBM to the power dependent RBM would also require new trip setpoints.

The ARTS/MELLLA application is evaluated on a plant-specific basis via a safety and impact analysis for meeting thermal and reactivity margins for BWR plants. When compared to the existing power/flow operating domain, operation in the MELLLA region results in plant operation along a higher rod line, which at off-rated operation allows for higher core power at a given core flow. This increases the fluid subcooling in the downcomer region of the reactor vessel and alters the power distribution in the core in a manner that can potentially affect steady-state operating thermal limit and transient/accident performances. The effect of this operating mode relative to fuel dependent analyses will be evaluated to confirm compliance with the required fuel thermal margins during plant operation. For subsequent reload cycles, PPL will

include the ARTS/MELLLA operating condition in the reload licensing basis. Attachment 3 presents the results of the safety analyses and system response evaluations for the non-fuel dependent tasks and the assumptions and conclusions that will be validated or updated for the fuel dependent tasks performed for operation of SSES in the region above the current licensed ELLLA and up to the MELLLA boundary line.

With the proposed power/flow map expansion to include the MELLLA region, the upper boundary of the licensed operating domain would be extended to approximately the 121% rod line. To accommodate this expanded operating domain, the APRM flow-biased scram and rod block trip setpoints would be revised. The clamped setpoints for the APRM will be unchanged.

Although they are part of the SSES design configuration and Technical Specifications, the APRM flow-biased scram and rod block lines are not credited in any SSES safety licensing analyses. The proposed setpoint changes would permit plant operation in the MELLLA region for operational flexibility purposes.

Representative results of the Rod Withdrawal Error (RWE) event (with the ARTS based power dependent RBM hardware) demonstrate the safety limit MCPR (SLMCPR) and fuel thermal-mechanical design limits are not exceeded, when appropriate power dependent trip setpoints are used in the RBM.

One objective of the ARTS/MELLLA APRM improvements is to justify removal of the APRM trip setdown requirement (TS 3.2.4, APRM Gain and Setpoints) using the following criteria:

- MCPR safety limit shall not be violated as a result of any Anticipated Operational Occurrence (AOO).
- All fuel thermal-mechanical design bases shall remain within the licensing limits.
- Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

Power and flow dependent MCPR adjustments to the MCPR and LHGR thermal limits will be determined using NRC approved analytical methods in TS 5.6.5. These adjustments will ensure that the above three criteria are met during operation at other than rated conditions without the APRM trip setdown.

The following additional changes would be made to reflect the deletion of TS 3.2.4:

a. References to TS 3.2.4 would be deleted from TS SR 3.3.1.1.3 and from TS 5.6.5 COLR.

b. The definition of Maximum Fraction of Limiting Power Density (MFLPD) would be deleted from Technical Specification 1.1 Definitions.

c. Section B3.2.4 and references to Technical Specification 3.2.4 and LCO 3.2.4 would be removed from the Technical Specification Bases. Reference to LCO 3.2.4 in SR 3.3.1.1.3 in the Technical Specification Bases would be removed.

Reactor Recirculation Suction Line Break (RSLB) Annulus Pressurization (AP) Loads

Attachment 3, Section 8.5 describes the evaluation of reactor asymmetric loads at the bounding condition (MELLLA minimum pump speed). A more realistic blowdown mass and energy release profile was determined using the GE LAMB computer program in lieu of the Reference 2 methodology. The LAMB analysis considers the pipe break separation time history, but ignores the fluid inertia effect, and thus, still provides conservative mass and energy release results. LAMB has been used in several plant licensing applications to calculate the blowdown mass flow rate and energy release profile in the event of an RSLB and has been used for licensing applications for power/flow map extension (MELLLA) associated with BWR extended power uprates (Reference 7). The LAMB methodology has been used to calculate the mass and energy releases for short-term post-LOCA containment response analysis for several applications. The LAMB results at the minimum MELLLA pump speed are bounded by the SSES design calculation of record.

The AP loads, the jet reaction loads/jet impingement loads, and the pipe whip loads would occur during the time periods following the double ended guillotine break of the recirculation suction line, and are combined for the evaluation of the structural integrity of the Reactor Pressure Vessel (RPV), reactor internals, the biological shield wall, control rod drive mechanism and the piping systems that are connected to the RPV and penetrate the biological shield wall. Since the mass and energy releases at off-rated conditions associated with ARTS/MELLLA have been shown to be bounded by the current analysis, these analyses are not performed.

Anticipated Transients Without Scram (ATWS)

Attachment 3, Section 10.0 describes the plant-specific analyses performed to demonstrate that the ATWS acceptance criteria are met for operation in the MELLLA region. All ATWS acceptance criteria were met. The peak vessel pressure and suppression pool temperature reported in Attachment 3 are for an assumed initial power level of 3489 MWt.

The peak vessel bottom pressure for this event is 1288 psig, which is below the ATWS overpressurization protection criterion of 1500 psig. The highest calculated suppression pool temperature is 207.1° F, which is below the SSES ATWS limit of 210° F. The highest calculated pellet clad temperature is 1420° F, which is significantly less than the ATWS limit. It was also determined that the containment pressure response following the ATWS satisfied the ATWS acceptance criteria. Consequently, the overall containment response following the limiting scenario satisfies ATWS acceptance criteria.

High Energy Line Break (HELB)

Attachment 3, Section 12.0 states that the mass and energy release profiles, assumed in the current design basis analysis for the Reactor Water Cleanup (RWCU) HELB analysis, are not bounding for MELLLA conditions. PPL evaluated the effects of the higher mass and energy release profiles and concluded the resulting subcompartment pressures, temperatures and humidity levels are acceptable with respect to existing design criteria.

Conclusion:

The proposed changes will increase operating flexibility in power ascension and operation at rated power. Replacement of the APRM trip setdown requirement with more direct power and flow dependent thermal limits will reduce the need for manual setpoint or gain adjustments and allow more direct thermal limits administration. This will improve the human/machine interface, update thermal limits administration, increase reliability, and provide more direct protection of plant safety.

5. REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

PPL Susquehanna has evaluated whether or not a significant hazards consideration is involved with the proposed change, by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," for each of the three proposed categories of changes described in Section 2 "Proposed Change," as discussed below:

Proposed Change No. 1:

The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. The APRM flow-biased scram and rod block trip setpoint setdown requirements is replaced by more direct power and flow dependent thermal limits to reduce the need for APRM setpoint or gain adjustments and to allow more direct thermal limits administration. Thermal limits are determined using NRC approved analytical methods specified by Technical Specification 5.6.5 b and will be specified in the Core Operating Limits Report. The APRM flow-biased scram setpoint and the APRM flow-biased rod block trip setpoints are reset to permit operation in the MELLLA operating domain.

Proposed Change No. 2:

The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS).

Proposed Change No. 3:

The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but

still conservative, methods of analysis to determine AP mass and energy release profile for AP loads resulting from the postulated RSLB.

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

Response: No

Proposed Change No. 1:

The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Thermal limits will be determined using NRC approved analytical methods. The proposed change will have no effect upon any accident initiating mechanism. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any Anticipated Operational Occurrence (AOO), and that the fuel thermal and mechanical design bases will be maintained. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change No. 2:

The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). The APRM and RBM are not involved in the initiation of any accident; and the APRM flow-biased scram and rod block functions are not credited in any PPL safety licensing analyses.

The analysis of the instrument line break event resulted in an insignificant change in the radiological consequences. The change for the instrument line break was an insignificant increase of 0.1 Rem.

Since the proposed changes will not affect any accident initiator, or introduce any initial conditions that would result in NRC approved criteria being exceeded, and since the APRM and RBM will remain capable of performing their design functions, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Proposed Change No. 3:

The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release

profile for AP loads resulting from the postulated RSLB. The releases resulting from the RSLB at off-rated conditions have been demonstrated to be bounded by the current design basis loads. Since the proposed changes do not affect any accident initiator and since the RSLB AP releases remain bounded by the current design basis, the proposed changes do not involve a significant increase in the probability or radiological consequences of an accident previously evaluated. Therefore the proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

Response: No

Proposed Change No. 1:

The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Because the thermal limits will continue to be met, no analyzed transient event will escalate into a new or different type of accident due to the initial starting conditions permitted by the adjusted thermal limits. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change No. 2:

The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). Changing the formulation for the APRM flow-biased scram and rod block trip setpoints and from a flow-biased RBM to a power dependent RBM does not change their respective functions and manner of operation. The change does not introduce a sequence of events or introduce a new failure mode that would create a new or different type of accident. The APRM flow-biased rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM flow-biased scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continues beyond the APRM flow-biased rod block setpoint. The power dependent RBM will prevent rod withdrawal when the power dependent RBM rod block setpoint is reached. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. In addition, operating within the expanded power flow map will not require any systems, structures or components to function differently than previously evaluated and will not create initial conditions that would result in a new or different accident. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Proposed Change No. 3:

The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release profile for AP loads resulting from the postulated RSLB. The proposed changes to the methods of analysis to determine AP mass and energy releases resulting from the postulated RSLB do not change the design function or operation of any plant equipment. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed changes. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No

Proposed Change No. 1:

The proposed change eliminates the Average Power Range Monitor (APRM) flow-biased scram and rod block trip setpoint setdown requirements and substitutes power and flow dependent adjustments to the Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR) thermal limits. Replacement of the APRM setpoint setdown requirement with power and flow dependent adjustments to the MCPR and LHGR thermal limits will ensure that margins to the fuel cladding Safety Limit are preserved during operation at other than rated conditions. Thermal limits will be determined using NRC approved analytical methods. The power and flow dependent adjustments will ensure that the MCPR safety limit will not be violated as a result of any Anticipated Operational Occurrence (AOO), and that the fuel thermal and mechanical design bases will be maintained. The 10 CFR 50.46 acceptance criteria for the performance of the Emergency Core Cooling System (ECCS) following postulated Loss-Of-Coolant Accidents (LOCAs) will continue to be met. Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Proposed Change No. 2:

The proposed change expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased scram and rod block scram trip setpoints and the replacement of the current flow-biased RBM with a new power dependent RBM, which will be implemented using a digital Power Range Neutron Monitoring System (PRNMS). The APRM flow-biased rod block trip setpoint will continue to block control rod withdrawal when core power significantly exceeds normal limits and approaches the scram level. The APRM flow-biased scram trip setpoint will continue to initiate a scram if the increasing power/flow condition continues beyond the APRM flow-biased rod block setpoint. The RBM will continue to prevent rod withdrawal when the power dependent RBM

rod block setpoint is reached. The MCPR and LHGR thermal limits will be developed to ensure that fuel thermal mechanical design bases shall remain within the licensing limits during a rod withdrawal error event and to ensure that the MCPR safety limit will not be violated as a result of a rod withdrawal error event. Operation in the expanded operating domain will not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. Anticipated operational occurrences and postulated accidents within the expanded operating domain will be evaluated using NRC approved methods. Therefore, the proposed change will not involve a significant reduction in the margin of safety.

Proposed Change No. 3:

The methods used to evaluate Annulus Pressurization (AP) and mass blowdown and energy releases resulting from the postulated Recirculation Suction Line Break (RSLB) at the MELLLA conditions are changed to use more realistic, but still conservative, methods of analysis to determine an AP mass and energy release profile for AP loads resulting from the postulated RSLB. Mass and energy releases for AP loads resulting from the postulated RSLB remain bounded by the current design basis releases. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, PPL concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

5.2.1 Analysis

10 CFR 50, Appendix A, General Design Criterion (GDC) 10 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. The assumptions and conclusions relative to fuel dependent evaluations will be validated on a cycle specific basis to ensure the requirements of GDC 10 continue to be met.

10 CFR 50, Appendix A, GDC 12 requires 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. The assumptions and conclusions relative to fuel dependent evaluations will be validated on a cycle specific basis to ensure the requirements of GDC 12 continue to be met.

10 CFR 50, Appendix A, GDC 50 requires that the reactor containment structure be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. Evaluations described in Attachment 3, Section 8.0 demonstrate that all containment parameters stay within their design limits.

10 CFR 50.46 sets forth acceptance criteria for the performance of the ECCS following postulated LOCAs. 10 CFR 50 Appendix K describes required and acceptable features of the evaluation models used to calculate ECCS performance. The plant specific LOCA analysis in Section 7.0 of Attachment 3 demonstrates that the requirements of 10 CFR 50.46 continue to be met.

10 CFR 50.49 establishes requirements for environmental qualification of electric equipment important to safety for nuclear power plants. Evaluations described in Attachment 3, Section 12.0 demonstrate acceptable results for steam line breaks and feedwater line break. For the RWCU HELB, the resulting subcompartment pressures, temperatures and humidity levels are acceptable with respect to existing design criteria.

10 CFR 50.62, in part, specifies the equivalent flow rate, level of boron concentration and boron-10 isotope enrichment required for BWR standby liquid control systems. The analyses described in Attachment 3, Section 10, confirm that key performance parameters (reactor vessel pressure, peak cladding temperature, suppression pool temperature, and containment pressure) remain within acceptable limits.

5.2.2 Conclusion

Based on the analyses provided in Section 4.0 Technical Analysis, the proposed change is consistent with applicable regulatory requirements and criteria. In conclusion, there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, such activities will be conducted in compliance with the Commission's regulations, and the approval of the proposed change will not be inimical to the common defense and security or to the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c)(9) identifies certain licensing and regulatory actions that are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operation of the facility in accordance with the proposed amendment would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a

significant increase in individual or cumulative occupational radiation exposure. PPL Susquehanna has evaluated the proposed change and has determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with issuance of the amendment. The basis for this determination, using the above criteria, follows:

1. As demonstrated in the No Significant Hazards Consideration Evaluation, the proposed amendment does not involve a significant hazards consideration.

2. There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite. The Technical Specification changes do not increase the consequences of any previously evaluated accident.

3. There is no significant increase in individual or cumulative occupational radiation exposure, because the Technical Specification changes do not result in a new mode of operation that would cause additional occupational exposure.

7. REFERENCES

1. PPL Letter PLA-5880, Britt T. McKinney (PPL) to USNRC "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.

2. GE Nuclear Energy, "Technical Description – Annulus Pressurization Load Adequacy Evaluation," NEDO-24548, January 1979.

3. PPL Letter PLA-3788, H. W. Keiser (PPL) to C. L. Miller (NRC) "Susquehanna Steam Electric Station Submittal of Licensing Topical Report On Power Uprate With Increased Flow," dated June 15, 1992.

4. PPL Letter PLA-4055, George T. Jones (PPL) to C. L. Miller (NRC) "Susquehanna Steam Electric Station Proposed Amendment No. 117 to License No. NPF-22: Power Uprate With Increased Flow," dated November 24, 1993.

5. PPL Letter PLA-4173, Robert G. Byram (PPL) to C. L. Miller (NRC) "Susquehanna Steam Electric Station Proposed Amendment No. 168 to License No. NPF-14: Power Uprate With Increased Flow," dated July 27, 1994.

6. PPL Letter PLA-5212, Robert G. Byram (PPL) to USNRC, "Susquehanna Steam Electric Station Proposed License Amendment No 235 to License NPF-14 and Proposed Amendment No. 200 to NPF-22: Power Uprate," October 30, 2000.

7. GE Nuclear Energy, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate," Licensing Topical Report NEDC- 32424P-A, February 1999.

Attachment 1 to PLA-5931

Changes To Technical Specifications

Unit 1

Technical Specification Mark-ups

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1.1 Definitions (continued)

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MAXIMUM FRACTION OF LIMITING POWER DENSITY(MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR for the APRM setpoint limit for that bundle type.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

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3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD when MFLPD is greater than Fraction of RTP.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

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SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	---------------------------NOTE--------------------------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. -- Verify MFLPD is within limits.	Once within 24 hours after ≥ 25% RTP <u>AND</u> 24 hours thereafter <u>AND</u> Prior to exceeding 50% RTP
SR 3.2.4.2	---------------------------NOTE--------------------------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. -- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

(A)

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SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
INSERT 3A (P) → SR 3.3.1.1.~~2~~ 3 (P)	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints"~~ while operating at ≥ 25% RTP. (DELETE) (A)	7 days
~~SR 3.3.1.1.3~~	~~Adjust the channel to conform to a calibrated flow signal.~~	~~7 days~~
SR 3.3.1.1.4	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days

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INSERT 3A:

SR 3.3.1.1.2	Perform CHANNEL CHECK	24 hours

P

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5[a]	3	H	SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—~~High~~ (Setdown)	2	~~2~~ 3[c]	G	SR 3.3.1.1.~~1~~ 2 ~~SR 3.3.1.1.4~~ SR 3.3.1.1.7 SR 3.3.1.1.8 ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.15~~ SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. ~~Flow Biased~~ Simulated Thermal Power—High	1	~~2~~ 3[c]	F	~~SR 3.3.1.1.1~~ SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~ ~~SR 3.3.1.1.15~~ ~~SR 3.3.1.1.17~~ SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	~~≤ 0.58 W + 62% RTP[b]~~ (DELETE) ≤ 0.62 W + 64.2% RTP[b] (A INSERT) and ≤ 115.5% RTP

(P)

(continued)

(a) ~~With any control rod withdrawn from a core cell containing one or more fuel assemblies.~~

(b) ~~0.58 W + 57% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."~~ insert 7

(c) insert 7

(A) ≤ 0.62 W + 64.2 % RTP[b] INSERT

P A 0.62 64.2

INSERT 7:

(b) 0.58(W-ΔW) + 62% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." ~~For single loop operation the value of ΔW = 5%/0.58. For two loop operation, the value of ΔW = 0.~~ A

(c) Each APRM channel provides inputs to both trip systems.

SURVEILLANCE REQUIREMENTS

NOTES

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days (184) (P)
SR 3.3.2.1.2	NOTE Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	NOTE Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	NOTE Neutron detectors are excluded. Verify the RBM Trip Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	DELETE (A) 24 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months

(continued)

INSERT ARTS-1 (A)

INSERT ARTS -1:

Verify the RBM:

a. Low Power Range - Upscale Function OR Intermediate Power Range - Upscale Function OR High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 28% RTP.

b. Intermediate Power Range - Upscale Function OR High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ Intermediate Power Range Setpoint specified in the COLR.

c. High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ High Power Range Setpoint specified in the COLR.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

INSERT ARTS-2

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range-Upscale	1[a]	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ 0.58W+55%[b]
~~b.~~ d. Inop	~~1[a]~~ (d)(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	NA
c. Downscale	1[a]	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≥ 3/125 divisions of full scale
2. Rod Worth Minimizer	1[c], 2[c] (g)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch—Shutdown Position	(d) (h)	2	SR 3.3.2.1.6	NA

DELETE (item 1.a)
DELETE (item 1.c)

(a) When THERMAL POWER is ≥ 30% RPT

(b) ~~≤ 0.58W + 50% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating"~~

DELETE (notes (a) and (b))

P — INSERT 10

(g) ~~(c)~~ With THERMAL POWER ≤ 10% RTP.

(h) ~~(d)~~ Reactor mode switch in the shutdown position.

A — INSERT ARTS-3

INSERT 10: — (P)

(b) 0.58(W-ΔW) + 55% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." For single loop operation the value of ΔW = 5%/0.58. For two loop operation, the value of ΔW = 0.

DELETE

(A)

INSERT ARTS-2: (A)

a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)
c. High Power Range - Upscale	(c), (d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)

INSERT ARTS-3: (A)

a. THERMAL POWER is ≥ 28% RTP and MCPR is less than the limit specified in the COLR except not required to be OPERABLE if the Intermediate Power Range - Upscale Function or High Power Range - Upscale Function is OPERABLE.

b. THERMAL POWER is ≥ Intermediate Power Range Setpoint specified in the COLR and MCPR is less than the limit specified in the COLR except not required to be OPERABLE if the High Power Range - Upscale Function is OPERABLE.

c. THERMAL POWER is ≥ High Power Range Setpoint specified in the COLR and < 90% RTP and MCPR is less than the limit specified in the COLR.

d. THERMAL POWER is ≥ 90% RTP and MCPR is less than the limit specified in the COLR.

e. THERMAL POWER is ≥ 28% RTP and < 90% RTP and MCPR is less than the limit specified in the COLR.

f. Allowable Value specified in the COLR.

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; ~~and~~ DELETE
4. ~~The Average Power Range Monitor (APRM) Gain and Setpoints for Specification 3.2.4; and~~
4 ~~5~~. The Shutdown Margin for Specification 3.1.1 ; and
~~6. The OPRM setpoints for Specification 3.3.1.3.~~

INSERT 11 (A) (P)

INSERT ARTS-4 (A)

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)

INSERT 11: (P)

5 ~~6.~~ Oscillation Power Range Monitor (OPRM) Trip setpoints, for Specification 3.3.1.1~~;~~ and (A)

INSERT ARTS-4: (A)

6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

Unit 2

Technical Specification Mark-ups

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3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation	TS/3.3-37
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3.3.7.1	Control Room Emergency Outside Air Supply (CREOAS) System Instrumentation	3.3-67

A DELETE A DELETE

(continued)

1.1 Definitions (continued)

DELETE

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR for the APRM setpoint limit for that bundle type.

(A)

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE—OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

LCO 3.2.4

a. MFLPD shall be less than or equal to Fraction of RTP; or

b. Each required APRM setpoint specified in the COLR shall be made applicable; or

c. Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD when MFLPD is greater than Fraction of RTP.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

DELETE

A

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.2.4.1	----------NOTE---------- Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4 Item b or c requirements. -------------------- Verify MFLPD is within limits.	Once within 24 hours after ≥ 25% RTP **AND** 24 hours thereafter **AND** Prior to exceeding 50% RTP
SR 3.2.4.2	----------NOTE---------- Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4 Item a requirements. -------------------- Verify APRM setpoints or gains are adjusted for the calculated MFLPD.	12 hours

DELETE

A

SURVEILLANCE REQUIREMENTS

--NOTES--

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2 3 (P) INSERT 3A (P)	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power is ≤ 2% RTP plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Setpoints" while operating at ≥ 25% RTP.	7 days e DELETE (A)
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days
SR 3.3.1.1.4	------------------------------NOTE------------------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

INSERT 3A:

SR 3.3.1.1.2.	Perform CHANNEL CHECK	24 hours

(P)

Table 3.3.1.1-1 (page 1 of 3)

Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux—High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	H	SR 3.3.1.1.5 SR 3.3.2.2.15	NA
2. Average Power Range Monitors					
a. Neutron Flux—High, (Setdown)	2	~~2~~ 3(c)	G	SR 3.3.1.1.~~1~~2 ~~SR 3.3.1.1.4~~ SR 3.3.1.1.7 SR 3.3.1.1.8 ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.15~~ SR 3.3.1.1.12 SR 3.3.1.1.18	≤ 20% RTP
b. ~~Flow Biased~~ Simulated Thermal Power—High	1	~~2~~ 3(c)	F	~~SR 3.3.1.1.1~~ SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~ ~~SR 3.3.1.1.15~~ ~~SR 3.3.1.1.17~~ SR 3.3.1.1.12 SR 3.3.1.1.18 SR 3.3.1.1.20	~~≤ 0.58 W + 62% RTP(b)~~ (DELETE; INSERT ≤ 0.62 W + 64.2% RTP(b)) and ≤ 115.5% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) ~~0.58 W + 57% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."~~ INSERT 7

(c) Insert 7

P P P P A

P

A

0.62 64.2

INSERT 7:

(b) 0.58(W-ΔW) + 62% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." ~~For single loop operation the value of ΔW = 5%/0.58. For two loop operation, the value of ΔW = 0.~~

A

(c) Each APRM channel provides inputs to both trip systems.

SURVEILLANCE REQUIREMENTS

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days 184 (P)
SR 3.3.2.1.2	NOTE Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	NOTE Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	NOTE Neutron detectors are excluded. Verify the RBM trip functions are not bypassed when THERMAL POWER is ≥ 30%.	DELETE (A) 24 months
SR 3.3.2.1.5	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months

INSERT ARTS-1 (A)

INSERT ARTS -1:

Verify the RBM:

a. Low Power Range - Upscale Function OR Intermediate Power Range - Upscale Function OR High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ 28% RTP.

b. Intermediate Power Range - Upscale Function OR High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ Intermediate Power Range Setpoint specified in the COLR.

c. High Power Range - Upscale Function is enabled (not bypassed) when APRM Simulated Thermal Power is ≥ High Power Range Setpoint specified in the COLR.

Table 3.3.2.1-1 (page 1 of 1)

Control Rod Block Instrumentation

INSERT ARTS-2

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range-Upscale	1(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ 0.58W+55%(b)
~~b.~~ d. Inop	~~1(a)~~ (d)(e)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	NA
c. Downscale	1(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≥ 3/125 divisions of full scale
2. Rod Worth Minimizer	1(c), 2(c) (g)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch—Shutdown Position	(d) (h)	2	SR 3.3.2.1.6	NA

DELETE

DELETE

(a) When THERMAL POWER is ≥ 30% RPT

~~(b) ≤ 0.58W + 50% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating"~~

(g) ~~(c)~~ With THERMAL POWER ≤ 10% RTP.

(h) ~~(d)~~ Reactor mode switch in the shutdown position.

DELETE

DELETE

INSERT 10

INSERT ARTS-3

(A)

(P)

(A)

INSERT 10: (P)

(b) 0.58(W-ΔW) + 55% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating." For single loop operation the value of ΔW = 5%/0.58. For two loop operation, the value of ΔW = 0.

(A) DELETE

INSERT ARTS-2: (A)

a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)
c. High Power Range - Upscale	(c), (d)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	(f)

INSERT ARTS-3: (A)

a. THERMAL POWER is ≥ 28% RTP and MCPR is less than the limit specified in the COLR except not required to be OPERABLE if the Intermediate Power Range - Upscale Function or High Power Range - Upscale Function is OPERABLE.

b. THERMAL POWER is ≥ Intermediate Power Range Setpoint specified in the COLR and MCPR is less than the limit specified in the COLR except not required to be OPERABLE if the High Power Range - Upscale Function is OPERABLE.

c. THERMAL POWER is ≥ High Power Range Setpoint specified in the COLR and < 90% RTP and MCPR is less than the limit specified in the COLR.

d. THERMAL POWER is ≥ 90% RTP and MCPR is less than the limit specified in the COLR.

e. THERMAL POWER is ≥ 28% RTP and < 90% RTP and MCPR is less than the limit specified in the COLR.

f. Allowable Value specified in the COLR.

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.
6. ~~The OPRM setpoints for Specification 3.3.1.3.~~

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)

INSERT 11:

5 ~~6.~~ Oscillation Power Range Monitor (OPRM) Trip setpoints, for Specification 3.3.1.1; and

A

INSERT ARTS-4:

6. The Allowable Values and power range setpoints for Rod Block Monitor Upscale Functions for Specification 3.3.2.1, Table 3.3.2.1-1.

Attachment 2 to PLA-5931

Changes To Technical Specification Bases For Information

Unit 1

Technical Specification Bases Mark-ups

For Information

B2.0	SAFETY LIMITS (SLs)	B2.0-1
B2.1.1	Reactor Core SLs	B2.0-1
B2.1.2	Reactor Coolant System (RCS) Pressure SL	B2.0-7
B3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	B3.0-1
B3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	TS/B3.0-10
B3.1	REACTIVITY CONTROL SYSTEMS	B3.1-1
B3.1.1	Shutdown Margin (SDM)	B3.1-1
B3.1.2	Reactivity Anomalies	B3.1-8
B3.1.3	Control Rod OPERABILITY	B3.1-13
B3.1.4	Control Rod Scram Times	B3.1-22
B3.1.5	Control Rod Scram Accumulators	B3.1-29
B3.1.6	Rod Pattern Control	TS/B3.1-34
B3.1.7	Standby Liquid Control (SLC) System	B3.1-39
B3.1.8	Scram Discharge Volume (SDV) Vent and Drain Valves	B3.1-47
B3.2	POWER DISTRIBUTION LIMITS	TS/B3.2-1
B3.2.1	Average Planar Linear Heat Generation Rate (APLHGR)	TS/B3.2-1
B3.2.2	Minimum Critical Power Ratio (MCPR)	TS/B3.2-5
B3.2.3	Linear Heat Generation Rate (LHGR)	B3.2-10
B3.2.4	Average Power Range Monitor (APRM) Gain and Setpoints	B3.2-14
B3.3	INSTRUMENTATION	TS/B3.3-1
B3.3.1.1	Reactor Protection System (RPS) Instrumentation	TS/B3.3-1
B3.3.1.2	Source Range Monitor (SRM) Instrumentation	TS/B3.3-35
B3.3.1.3	Oscillation Power Range Monitor (OPRM)	TS/B3.3-43a
B3.3.2.1	Control Rod Block Instrumentation	TS/B3.3-44
B3.3.2.2	Feedwater – Main Turbine High Water Level Trip Instrumentation	B3.3-55
B3.3.3.1	Post Accident Monitoring (PAM) Instrumentation	TS/B3.3-64
B3.3.3.2	Remote Shutdown System	B3.3-76
B3.3.4.1	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	B3.3-81
B3.3.4.2	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	B3.3-92
B3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation	B3.3-101
B3.3.5.2	Reactor Core Isolation Cooling (RCIC) System Instrumentation	B3.3-135
B3.3.6.1	Primary Containment Isolation Instrumentation	B3.3-147
B3.3.6.2	Secondary Containment Isolation Instrumentation	TS/B3.3-180
B3.3.7.1	Control Room Emergency Outside Air Supply (CREOAS) System Instrumentation	B3.3-192

A

DELETE

DELETE

A

(continued)

B. 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

A INSERT ARTS B5A

APLHGR limits are developed as a function of fuel type and exposure. The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Extended Load Line Limit Analysis (ELLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the 100% rod line and the APRM rod block line (i.e., the ELLA region). The ELLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.

(continued)

TECH SPEC BASES MARKUP

INSERT ARTS B5A —(A)

The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Maximum Extended Load Line Limit Analysis (MELLLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the rod line that runs through 100% RTP and maximum core flow and the upper boundary of the MELLLA region. The MELLLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required.

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel transient mechanical design limit (i.e., Protection Against Power Transient (PAPT) limit).

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP to ensure the PAPT limits are not violated under steady state or transient conditions.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow,

(continued)

BASES

BACKGROUND (continued)

the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected APRM setpoints are supported by the analyses that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, LHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also conditioned to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins be maintained to the fuel transient mechanical design limit (PAPT).

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, LHGR and MCPR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO

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(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

(MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the LHGR or the MCPR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, the MCPR margin degradation at reduced power and flow is factored into the power and flow dependent MCPR limits (LCO 3.2.2). For LHGR (Ref. 4), either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is reduced by the ratio of FRTP to the core limiting MFLPD. The adjustment in the APRM gain can be performed provided it is during power ascension up to 90% of RATED THERMAL POWER, that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. 5).

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BASES (continued)

LCO

Meeting any one of the following conditions ensures acceptable operating margin to the transient mechanical design limit (PAPT) for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

MFLPD is the ratio of the limiting LHGR to the LHGR limit for APRM setpoints for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel transient mechanical design limit (PAPT) is not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3,

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(continued)

BASES

APPLICABILITY (continued)

sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

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ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel transient mechanical design limit (PAPT) may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

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INSERT B1

~~The APRM setpoints include the APRM Rod Block Flow Bias Neutron Flux Upscale Setpoint which is controlled in Technical Requirement Manual (TRM) 3.1.3 "Control Rod Block Instrumentation."~~

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

The MFLPD is required to be calculated and compared to FRTP or APRM gain or setpoints to ensure that the reactor

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(continued)

INSERT B1: — (P)

The APRM setpoints include the APRM Simulated Thermal Power – High RPS scram setpoint, LCO 3.3.1.1 "RPS Instrumentation," Function 2.b, and APRM Simulated Thermal Power – High rod block setpoint, Technical Requirements Manual (TRM) TRO 3.1.3 "Control Rod Block Instrumentation", Function 1.b.

(A)

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2 (continued)

is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry functions. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the LHGR (LCO 3.2.3). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MFLPD must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. When MFLPD is greater than FRTP, SR 3.2.4.2 must be performed. The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification when MFLPD is greater than the fraction of rated thermal power (FRTP) because more rapid changes in power distribution are typically expected.

(P)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Section 4.
3. FSAR, Section 15.
4. 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.
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

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(A)

and SR 3.3.1.1.2

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit, and does not necessarily indicate the channel is inoperable.

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INSERT B8A

The Frequency is based upon operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the channels required by the LCO.

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SR 3.3.1.1.2 3

P

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD times 100. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

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A

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in

(continued)

INSERT B8A: — (P)

The Frequency of once every 12 hours for SR 3.3.1.1.1 is based upon operating experience that demonstrates that channel failure is rare. The Frequency of once every 24 hours for SR 3.3.1.1.2 is based upon operating experience that demonstrates that channel failure is rare and the evaluation in References 15 and 16.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch—Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to enable the RBM. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 2).

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence

(continued)

INSERT B16: (P)

An APRM flux signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and an APRM flux signal from another of the APRM channels supplies the reference signal to the second RBM channel.

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(A)

INSERT ARTS B1: (A)

A simulated thermal power signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a simulated thermal power signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM simulated thermal power is

BASES

BACKGROUND (continued)

based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 1). The RWM is a single channel system that provides input into RMCS rod block channel 2.

The function of the individual rod sequence steps (banking steps) is to minimize the potential reactivity increase from postulated CRDA at low power levels. However, if the possibility for a control rod to drop can be eliminated, then banking steps at low power levels are not needed to ensure the applicable event limits can not be exceeded. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn control rods are coupled.

To eliminate the possibility of a CRDA, administrative controls require that any partially inserted control rods, which have not been confirmed to be coupled since their last withdrawal, be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. If a control rod has been checked for coupling at notch 48 and the rod has not been moved inward, this rod is in contact with it's drive and is not required to be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. However, if it cannot be confirmed that the control rod has been moved inward, then that rod shall be fully inserted prior to reaching the THERMAL POWER of ≤10% RTP. The remaining control rods may then be inserted without the need to stop at intermediate positions since the possibility of a CRDA has been eliminated.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

(A) INSERT ARTS B1A

1. Rod Block Monitor

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The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The RBM was originally designed to

(continued)

Insert ARTS B1A —— (A)

The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 14. The fuel thermal performance as a function of RBM Allowable Value is determined from the analysis. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

For Information Only

BASES

DELETE

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

prevent fuel damage during a Rod Withdrawal Error (RWE) event while operating in the power range in a normal mode of operation. FSAR 15.4.2 (Ref. 10) (Rod Withdrawal Error - At Power) originally took credit for the RBM automatically actuating to stop control rod motion and preventing fuel damage during an RWE event at power. However, current reload analyses do not take credit for the RBM system. The Allowable Values are chosen as a function of power level to not exceed the APRM scram setpoints.

A

The RBM function satisfies Criterion 4 of the NRC Policy Statement (Ref. 7).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block for this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

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INSERT ARTS B2

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

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INSERT ARTS B3

The RBM will function when operating greater than ~~30~~% RTP. Below this power level, the RBM is not required to be OPERABLE.

28 A

2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated.

(continued)

INSERT ARTS B2: (A)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter, the calculated RBM flux (RBM channel signal). When the RBM flux value exceeds the applicable setpoint, the RBM provides a trip output. The analytic limits are derived from the limiting values of the process parameters. The Allowable Values are determined from the analytic limits corrected for calibration, process, and some instrument errors. The trip setpoints are then determined, based on the Allowable Values, by accounting for calibration-based errors. These calibration based instrument errors are limited to instrument drift, errors associated with measurement and test equipment, and calibration tolerance of LPRM input processing in the average power range monitor (APRM) equipment. The RBM performs only digital calculations on digitized LPRM signals received from the APRM equipment. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environment errors are accounted for and appropriately applied for the instrumentation.

TECH SPEC BASES MARKUP

INSERT ARTS B3:

The RBM selects one of three different RBM flux trip setpoints to be applied based on the current value of THERMAL POWER. THERMAL POWER is indicated to each RBM channel by a simulated thermal power (STP) reference signal input from an associated reference APRM channel. The OPERABLE range is divided into three "power ranges," a "low power range," an "intermediate power range," and a "high power range." The RBM flux trip setpoint applied within each of these three power ranges is, respectively, the "low trip setpoint," the "intermediate trip setpoint," and the "high trip setpoint" (Allowable Values for which are defined in the COLR). To determine the current power range, each RBM channel compares its current STP input value to three power setpoints, the "low power setpoint",(28%), the "intermediate power setpoint" (current value defined in the COLR), and the "high power setpoint" (current value defined in the COLR), which define, respectively, the lower limit of the low power range, the lower limit of the intermediate power range, and the lower limit of the high power range. The trip setpoint applicable for each power range is more restrictive than the corresponding setpoint for the lower power range(s). When STP is below the low power setpoint, the RBM flux trip outputs are automatically bypassed but the low trip setpoint continues to be applied to indicate the RBM flux setpoint on the NUMAC RBM displays.

The calculated (required) setpoints and applicable power ranges are bounding values. In the equipment implementation, it is necessary to apply a "deadband" to each setpoint. The deadband is applied to the RBM trip setpoint selection logic and the RBM trip automatic bypass logic such that the setpoint being applied is always equal to or more conservative than the required setpoint. Since the RBM flux trip setpoint applicable to the higher power ranges are more conservative than the corresponding trip setpoints for lower power ranges, the trip setpoint applicable to the higher power range (high power range or intermediate power range) continues to be applied when STP decreases below the lower limit of that range until STP is below the power range setpoint by a value exceeding the deadband. Similarly, when STP decreases below the low power setpoint, the automatic bypass of RBM flux trip outputs will not be applied until STP decreases below the trip setpoint a value exceeding the deadband.

The RBM channel uses THERMAL POWER, as represented by the STP input value from its reference APRM channel, to automatically enable RBM flux trip outputs (remove the automatic bypass) and to select the RBM flux trip setpoint to be applied. However, the RBM Upscale function is only required to be OPERABLE when the MCPR values are less than the values defined in the COLR, depending on the THERMAL POWER level. Therefore, even though the RBM Upscale Function is implemented in each RBM channel as a single trip function with a selected trip setpoint, it is characterized in Table 3.3.2.1-1 as three Functions, the Low Power Range – Upscale Function, the Intermediate Power Range – Upscale Function, and the High Power Range – Upscale Function, to facilitate correct definition of the OPERABILITY requirements for the Functions. Each Function corresponds to one of the RBM power ranges. Due to the deadband effects on the determination of the current power range, the transition between these three Functions will occur at slightly different THERMAL POWER levels for increasing power versus decreasing power. Since the RBM flux trip setpoints applied for the higher power ranges are more conservative, the OPERABILITY requirement for the Low Power Range – Upscale Function is satisfied if the Intermediate Power Range – Upscale Function or the High Power Range – Upscale Function is OPERABLE. Similarly, the OPERABILITY requirement for the Intermediate Power Range – Upscale Function is satisfied if the High Power Range – Upscale Function is OPERABLE.

BASES

SURVEILLANCE REQUIREMENTS (continued)

assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. The Frequency of 92 days is based on reliability analyses (Ref. 8).

P 184 (Refs. 8, 12, and 13) P

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs and by verifying proper indication of the selection error of at least one out-of-sequence control rod. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

INSERT ARTS B4

SR 3.3.2.1.4

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The RBM trips are automatically bypassed when power is below a specified value and a peripheral control rod is not selected. The power Allowable Value must be verified periodically to not be bypassed when ≥ 30% RTP. This is performed by a Functional check. If any RBM bypass setpoint is non-conservative, then the affected RBM channel is

(continued)

INSERT ARTS B4: —— (A)

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, one corresponding to each specific power range. The purpose of this SR is to assure that for each RBM power range, the RBM flux trip rod block outputs are enabled (not bypassed) and that the RBM flux trip setpoint being applied is equal to or more conservative than the specified Allowable Values in the COLR. If any power range setpoint is non-conservative, then the affected RBM channel is considered inoperable.

The Low Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for low trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed). The Intermediate Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for intermediate trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed). The High Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for high trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed).

The SR is performed by varying the APRM Simulated Thermal Power input to the RBM from the reference APRM channel, and confirming that the criteria in the SR is met for both increasing and decreasing values of Simulated Thermal Power.

SR 3.3.2.1.4, item a is satisfied if, for an APRM Simulated Thermal Power level ≥ 28%, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the low trip setpoint Allowable Value defined in the COLR. (Note that the intermediate trip setpoint and the high trip setpoint Allowable Values are less than the low trip setpoint Allowable Value.)

SR 3.3.2.1.4, item b is satisfied if, for an APRM Simulated Thermal Power level ≥ the intermediate power level setpoint Allowable Value defined in the COLR, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the intermediate trip setpoint Allowable Value defined in the COLR. (Note that the high trip setpoint Allowable Value is less than the intermediate trip setpoint Allowable Value.)

SR 3.3.2.1.4, item c is satisfied if, for an APRM Simulated Thermal Power level ≥ the high power level setpoint Allowable Value defined in the COLR, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the high trip setpoint Allowable Value defined in the COLR.

This SR is performed using APRM STP, which is received digitally from the reference APRM channel. All logic in the RBM is digital. Therefore, consistent with the calibration frequency justified in Reference 12 and the APRM STP calibration SR 3.3.1.1.18 frequency, a frequency of 24 months is selected for this SR.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.4 (continued)

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(A)

considered inoperable. Alternatively, the RBM channel can be placed in the conservative condition (i.e., enabling the RBM trip). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal.

Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 24 month Frequency is based on the need to perform the surveillance during a plant start-up.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signals. The automatic bypass setpoint must be verified periodically to be not bypassed ≤ 10% RTP. This is performed by a Functional check. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the need to perform the Surveillance during a plant start-up.

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch—Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch—Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 7.7.1.2.8.

2. FSAR, Section 7.6.1.a.5.7

3. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.

4. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

5. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

6. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

7. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 32193)

8. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation, Technical Specifications," February 1991.

10. FSAR, Section 15.4.2.

11. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.

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INSERT ARTS B4A

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12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

13. NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November 1997.

Insert ARTS B4A (A)

14. 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, June 1986.

Unit 2

Technical Specification Bases Mark-ups

For Information

B2.0	SAFETY LIMITS (SLs)	TS/B2.0-1
B2.1.1	Reactor Core SLs	TS/B2.0-1
B2.1.2	Reactor Coolant System (RCS) Pressure SL	B2.0-6
B3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	B3.0-1
B3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	B3.0-10
B3.1	REACTIVITY CONTROL SYSTEMS	B3.1-1
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B3.1.2	Reactivity Anomalies	B3.1-8
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B3.2	POWER DISTRIBUTION LIMITS	TS/B3.2-1
B3.2.1	Average Planar Linear Heat Generation Rate (APLHGR)	TS/B3.2-1
B3.2.2	Minimum Critical Power Ratio (MCPR)	TS/B3.2-5
B3.2.3	Linear Heat Generation Rate (LHGR)	TS/B3.2-10
B3.2.4	Average Power Range Monitor (APRM) Gain and Setpoints	TS/B3.2-14
B3.3	INSTRUMENTATION	TS/B3.3-1
B3.3.1.1	Reactor Protection System (RPS) Instrumentation	TS/B3.3-1
B3.3.1.2	Source Range Monitor (SRM) Instrumentation	TS/B3.3-35
B3.3.1.3	Oscillation Power Range Monitor	TS/B3.3-43a
B3.3.2.1	Control Rod Block Instrumentation	TS/B3.3-44
B3.3.2.2	Feedwater – Main Turbine High Water Level Trip Instrumentation	B3.3-55
B3.3.3.1	Post Accident Monitoring (PAM) Instrumentation	TS/B3.3-64
B3.3.3.2	Remote Shutdown System	B3.3-76
B3.3.4.1	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	B3.3-81
B3.3.4.2	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	B3.3-92
B3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation	TS/B3.3-101
B3.3.5.2	Reactor Core Isolation Cooling (RCIC) System Instrumentation	B3.3-135
B3.3.6.1	Primary Containment Isolation Instrumentation	B3.3-147
B3.3.6.2	Secondary Containment Isolation Instrumentation	TS/B3.3-180
B3.3.7.1	Control Room Emergency Outside Air Supply (CREOAS) System Instrumentation	B3.3-192

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B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that limits specified in 10 CFR 50.46 are not exceeded during the postulated design basis loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

SPC performed LOCA calculations for the SPC ATRIUM™-10 fuel design. The analytical methods and assumptions used in evaluating the fuel design limits from 10 CFR 50.46 are presented in References 3, 4, 5, and 6 for the SPC analysis. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) that determine the APLHGR Limits are presented in References 3 through 9.

LOCA analyses are performed to ensure that the APLHGR limits are adequate to meet the Peak Cladding Temperature (PCT), maximum cladding oxidation, and maximum hydrogen generation limits of 10 CFR 50.46. The analyses are performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. A complete discussion of the analysis codes are provided in References 3, 4, 5, and 6 for the SPC analysis. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within the assembly.

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APLHGR limits are developed as a function of fuel type and exposure. The SPC analysis is valid for full cores of ATRIUM™-10 fuel. The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g. Extended Load Line Limit Analysis (ELLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the 100% rod line and the APRM rod block line (i.e., the ELLA region). The ELLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required. The results of the analysis demonstrate the PCTs are within the 10 CFR 50.46 limit, and that APLHGR multipliers as a function of core flow are not required.

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TECH SPEC BASES MARKUP

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The SPC LOCA analyses also consider several alternate operating modes in the development of the APLHGR limits (e.g., Maximum Extended Load Line Limit Analysis (MELLLA), Suppression Pool Cooling Mode, and Single Loop Operation (SLO)). LOCA analyses were performed for the regions of the power/flow map bounded by the rod line that runs through 100% RTP and maximum core flow and the upper boundary of the MELLLA region. The MELLLA region is analyzed to determine whether an APLHGR multiplier as a function of core flow is required.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoints

BASES

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BACKGROUND

The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design," GDC 13, "Instrumentation and Control," GDC 20, "Protection System Functions," and GDC 23, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel transient mechanical design limit (i.e., Protection Against Power Transient (PAPT) limit).

The condition of excessive power peaking is determined by the ratio of the actual power peaking to the limiting power peaking at RTP. This ratio is equal to the ratio of the core limiting MFLPD to the Fraction of RTP (FRTP), where FRTP is the measured THERMAL POWER divided by the RTP. Excessive power peaking exists when:

$$\frac{\text{MFLPD}}{\text{FRTP}} > 1,$$

indicating that MFLPD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is compensated by a gain adjustment on the APRMs or adjustment of the APRM setpoints. Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP to ensure the PAPT limits are not violated under steady state or transient conditions.

The normally selected APRM setpoints position the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The setpoints are flow biased with a slope that approximates the upper flow control line, such that an approximately constant margin is maintained between the flow biased trip level and the upper operating boundary for core flows in excess of about 45% of rated core flow. In the range of infrequent operations below 45% of rated core flow, the margin to scram is reduced because of the nonlinear core flow versus drive flow relationship. The normally selected

(continued)

BASES

BACKGROUND (continued)

APRM setpoints are supported by the analyses that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, LHGR and MCPR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the flow biased APRM scram setpoints may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.

The APRM neutron flux signal is also conditioned to more closely follow the fuel cladding heat flux during power transients. The APRM neutron flux signal is a measure of the core thermal power during steady state operation. During power transients, the APRM signal leads the actual core thermal power response because of the fuel thermal time constant. Therefore, on power increase transients, the APRM signal provides a conservatively high measure of core thermal power. By passing the APRM signal through an electronic filter with a time constant approximately equal to, that of the fuel thermal time constant, an APRM transient response that more closely follows actual fuel cladding heat flux is obtained. The delayed response of the filtered APRM signal allows the flow biased APRM scram levels to be positioned closer to the upper bound of the normal power and flow range, without unnecessarily causing reactor scrams during short duration neutron flux spikes. These spikes can be caused by insignificant transients such as performance of main steam line valve surveillances or momentary flow increases of only several percent.

APPLICABLE SAFETY ANALYSES

The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins be maintained to the fuel transient mechanical design limit (PAPT).

FSAR safety analyses (Refs. 2 and 3) concentrate on the rated power condition for which the minimum expected margin to the operating limits (APLHGR, LHGR and MCPR) occurs. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO(MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the LHGR or the MCPR during a transient can be greater than at the rated condition

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, the MCPR margin degradation at reduced power and flow is factored into the power and flow dependent MCPR limits (LCO 3.2.2). For LHGR (Ref. ~~4 and 5~~), either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the flow biased APRM scram level is reduced by the ratio of FRTP to the core limiting MFLPD. The adjustment in the APRM gain can be performed provided it is during power ascension up to 90% of RATED THERMAL POWER, that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER, the required gain adjustment increment does not exceed 10% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the flow biased APRM scram setpoints, dependent on the increased peaking that may be encountered.

The APRM gain and setpoints satisfy Criteria 2 and 3 of the NRC Policy Statement (Ref. ~~5~~5).

LOC

Meeting any one of the following conditions ensures acceptable operating margin to the transient mechanical design limit (PAPT) for events described above:

a. Limiting excess power peaking;

b. Reducing the APRM flow biased neutron flux upscale scram setpoints by multiplying the APRM setpoints by the ratio of FRTP and the core limiting value of MFLPD; or

c. Increasing APRM gains to cause the APRM to read greater than 100 times MFLPD (in %). This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

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BASES (continued)

LCO (continued)

MFLPD is the ratio of the limiting LHGR to the LHGR limit for APRM setpoints for the specific bundle type. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM flow biased scram setpoints are reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM flow biased scram setpoints. Adjusting APRM gain or setpoints is equivalent to MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM setpoint adjustment) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," are required to be adjusted. In addition, each APRM may be allowed to have its gain or setpoints adjusted independently of other APRMs that are having their gain or setpoints adjusted.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, and APRM flow biased scram and associated setdowns are provided to ensure that the fuel transient mechanical design limit (PAPT) is not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the reactor is operating at ≥ 25% RTP.

ACTIONS

A.1

If the APRM gain or setpoints are not within limits while the MFLPD has exceeded FRTP, the margin to the fuel transient mechanical design limit (PAPT) may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.

The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or setpoints to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.

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BASES

ACTIONS

A.1 (continued)

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The APRM setpoints include the APRM Rod Block Flow Bias Neutron Flux Upscale Setpoint which is controlled in the Technical Requirements Manual 3.1.3 "Control Rod Block Instrumentation."

P

B.1

If MFLPD cannot be restored to within its required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER is reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 and SR 3.2.4.2

P

functions

The MFLPD is required to be calculated and compared to FRTP or APRM gain or setpoints to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are only required to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate gain or setpoint, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or flow biased neutron flux scram circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with the determination of other thermal limits, specifically those for the APLHGR (LCO 3.2.1). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 24 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels and because the MFLPD must be calculated prior to exceeding 50% RTP unless performed in the previous 24 hours. When MFLPD is greater than FRTP, SR 3.2.4.2 must be performed. The 12 hour Frequency of SR 3.2.4.2 requires a more frequent verification when MFLPD is greater than the fraction of rated thermal power (FRTP) because more rapid changes in power distribution are typically expected.

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TECH SPEC BASES MARKUP

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The APRM setpoints include the APRM Simulated Thermal Power – High RPS scram setpoint, LCO 3.3.1.1 "RPS Instrumentation," Function 2.b, and APRM Simulated Thermal Power – High rod block setpoint, Technical Requirements Manual (TRM) TRO 3.1.3 "Control Rod Block Instrumentation", Function 1.b.

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BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 23.
2. FSAR, Section 4.
3. FSAR, Section 15.
4. 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.
5. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 39132).

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and SR 3.3.1.1.2 (P)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

Agreement criteria which are determined by the plant staff based on an investigation of a combination of the channel instrument uncertainties, may be used to support this parameter comparison and include indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit, and does not necessarily indicate the channel is inoperable.

(P) Insert B8A

~~The Frequency is based upon operating experience that demonstrates channel failure is rare.~~ The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.1.~~2~~ 3 (P)

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. (A) LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD times 100. (DELETE) The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at ≥ 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large, inherent margin to thermal limits (MCPR, LHGR and APLHGR). At ≥ 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days, in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating

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TECH SPEC BASES MARKUP

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The Frequency of once every 12 hours for SR 3.3.1.1.1 is based upon operating experience that demonstrates that channel failure is rare. The Frequency of once every 24 hours for SR 3.3.1.1.2 is based upon operating experience that demonstrates that channel failure is rare and the evaluation in References 15 and 16.

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch—Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each Reactor Protection System (RPS) trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to enable the RBM. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 2).

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and

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An APRM flux signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and an APRM flux signal from another of the APRM channels supplies the reference signal to the second RBM channel.

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TECH SPEC BASES MARKUP

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A simulated thermal power signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a simulated thermal power signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM simulated thermal power is

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

INSERT ARTS B1A

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The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The RBM was originally designed to prevent fuel damage during a Rod Withdrawal Error (RWE) event while operating in the power range in a normal mode of operation. FSAR 15.4.2 (Ref. 10) (Rod Withdrawal Error - At Power) originally took credit for the RBM automatically actuating to stop control rod motion and preventing fuel damage during an RWE event at power. However, current reload analyses do not take credit for the RBM system. The Allowable Values are chosen as a function of power level to not exceed the APRM scram setpoints.

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The RBM function satisfies Criterion 4 of the NRC Policy Statement (Ref. 7).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block for this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

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¶ INSERT ARTS B2

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

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The RBM is designed to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 14. The fuel thermal performance as a function of RBM Allowable Value is determined from the analysis. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

INSERT ARTS B2: (A)

Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter, the calculated RBM flux (RBM channel signal). When the RBM flux value exceeds the applicable setpoint, the RBM provides a trip output. The analytic limits are derived from the limiting values of the process parameters. The Allowable Values are determined from the analytic limits corrected for calibration, process, and some instrument errors. The trip setpoints are then determined, based on the Allowable Values, by accounting for calibration-based errors. These calibration based instrument errors are limited to instrument drift, errors associated with measurement and test equipment, and calibration tolerance of LPRM input processing in the average power range monitor (APRM) equipment. The RBM performs only digital calculations on digitized LPRM signals received from the APRM equipment. The trip setpoints and Allowable Values determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environment errors are accounted for and appropriately applied for the instrumentation.

BASES

28 — A

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The RBM will function when operating greater than ~~30~~% RTP. Below this power level, the RBM is not required to be OPERABLE.

INSERT ARTS B3

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2. Rod Worth Minimizer

The RWM enforces the banked position withdrawal sequence (BPWS) to ensure that the initial conditions of the CRDA analysis are not violated. The analytical methods and assumptions used in evaluating the CRDA are summarized in References 2, 3, 4, and 5. The BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions. Requirements that the control rod sequence is in compliance with the BPWS are specified in LCO 3.1.6, "Rod Pattern Control."

When performing a shutdown of the plant, an optional BPWS control rod sequence (Ref. 7) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference 11 control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the improved BPWS control rod insertion, or may be bypassed and the improved BPWS shutdown sequence implemented under the controls in Condition D.

The RWM Function satisfies Criterion 3 of the NRC Policy Statement. (Ref. 7)

Since the RWM is designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 6). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with inoperable control rods, or to allow correction of a control rod pattern not in compliance with the BPWS. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

(continued)

INSERT ARTS B3:

The RBM selects one of three different RBM flux trip setpoints to be applied based on the current value of THERMAL POWER. THERMAL POWER is indicated to each RBM channel by a simulated thermal power (STP) reference signal input from an associated reference APRM channel. The OPERABLE range is divided into three "power ranges," a "low power range," an "intermediate power range," and a "high power range." The RBM flux trip setpoint applied within each of these three power ranges is, respectively, the "low trip setpoint," the "intermediate trip setpoint," and the "high trip setpoint" (Allowable Values for which are defined in the COLR). To determine the current power range, each RBM channel compares its current STP input value to three power setpoints, the "low power setpoint",(28%), the "intermediate power setpoint" (current value defined in the COLR), and the "high power setpoint" (current value defined in the COLR), which define, respectively, the lower limit of the low power range, the lower limit of the intermediate power range, and the lower limit of the high power range. The trip setpoint applicable for each power range is more restrictive than the corresponding setpoint for the lower power range(s). When STP is below the low power setpoint, the RBM flux trip outputs are automatically bypassed but the low trip setpoint continues to be applied to indicate the RBM flux setpoint on the NUMAC RBM displays.

The calculated (required) setpoints and applicable power ranges are bounding values. In the equipment implementation, it is necessary to apply a "deadband" to each setpoint. The deadband is applied to the RBM trip setpoint selection logic and the RBM trip automatic bypass logic such that the setpoint being applied is always equal to or more conservative than the required setpoint. Since the RBM flux trip setpoint applicable to the higher power ranges are more conservative than the corresponding trip setpoints for lower power ranges, the trip setpoint applicable to the higher power range (high power range or intermediate power range) continues to be applied when STP decreases below the lower limit of that range until STP is below the power range setpoint by a value exceeding the deadband. Similarly, when STP decreases below the low power setpoint, the automatic bypass of RBM flux trip outputs will not be applied until STP decreases below the trip setpoint a value exceeding the deadband.

The RBM channel uses THERMAL POWER, as represented by the STP input value from its reference APRM channel, to automatically enable RBM flux trip outputs (remove the automatic bypass) and to select the RBM flux trip setpoint to be applied. However, the RBM Upscale function is only required to be OPERABLE when the MCPR values are less than the values defined in the COLR, depending on the THERMAL POWER level. Therefore, even though the RBM Upscale Function is implemented in each RBM channel as a single trip function with a selected trip setpoint, it is characterized in Table 3.3.2.1-1 as three Functions, the Low Power Range – Upscale Function, the Intermediate Power Range – Upscale Function, and the High Power Range – Upscale Function, to facilitate correct definition of the OPERABILITY requirements for the Functions. Each Function corresponds to one of the RBM power ranges. Due to the deadband effects on the determination of the current power range, the transition between these three Functions will occur at slightly different THERMAL POWER levels for increasing power versus decreasing power. Since the RBM flux trip setpoints applied for the higher power ranges are more conservative, the OPERABILITY requirement for the Low Power Range – Upscale Function is satisfied if the Intermediate Power Range – Upscale Function or the High Power Range – Upscale Function is OPERABLE. Similarly, the OPERABILITY requirement for the Intermediate Power Range – Upscale Function is satisfied if the High Power Range – Upscale Function is OPERABLE.

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 9) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. The Frequency of 92 days is based on reliability analyses (Ref. 8).

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs and by verifying proper indication of the selection error of at least one out-of-sequence control rod. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. As noted, SR 3.3.2.1.3 is not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. This allows entry into MODE 2 for SR 3.3.2.1.2, and entry into MODE 1 when THERMAL POWER is ≤ 10% RTP for SR 3.3.2.1.3, to perform the required Surveillance if the 92 day Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. The Frequencies are based on reliability analysis (Ref. 8).

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

INSERT ARTS B4

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SR 3.3.2.1.4

The RBM trips are automatically bypassed when power is below a specified value and a peripheral control rod is not selected. The power Allowable Value must be verified periodically to not be bypassed when ≥ 30% RTP. This is performed by a Functional check. If any RBM bypass setpoint is non-conservative, then the affected RBM channel is considered inoperable. Alternatively, the RBM channel can be placed in the conservative condition (i.e., enabling the RBM trip). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal.

Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.8. The 24 month Frequency is based on the need to perform the Surveillance during a plant start-up.

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from steam flow signals. The automatic bypass setpoint must be verified periodically to be not bypassed ≤ 10% RTP. This is performed by a Functional check. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on the need to perform the surveillance during a plant start-up.

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch—Shutdown Position Function to ensure that the entire channel will perform the intended function. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch—Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

(continued)

INSERT ARTS B4: —(A)

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, one corresponding to each specific power range. The purpose of this SR is to assure that for each RBM power range, the RBM flux trip rod block outputs are enabled (not bypassed) and that the RBM flux trip setpoint being applied is equal to or more conservative than the specified Allowable Values in the COLR. If any power range setpoint is non-conservative, then the affected RBM channel is considered inoperable.

The Low Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for low trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed). The Intermediate Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for intermediate trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed). The High Power Range – Upscale Function is enabled when the RBM flux trip setpoint being applied is equal to or less than the Allowable Value for high trip setpoint defined in the COLR, and the RBM flux trip rod block outputs are enabled (not bypassed).

The SR is performed by varying the APRM Simulated Thermal Power input to the RBM from the reference APRM channel, and confirming that the criteria in the SR is met for both increasing and decreasing values of Simulated Thermal Power.

SR 3.3.2.1.4, item a is satisfied if, for an APRM Simulated Thermal Power level ≥ 28%, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the low trip setpoint Allowable Value defined in the COLR. (Note that the intermediate trip setpoint and the high trip setpoint Allowable Values are less than the low trip setpoint Allowable Value.)

SR 3.3.2.1.4, item b is satisfied if, for an APRM Simulated Thermal Power level ≥ the intermediate power level setpoint Allowable Value defined in the COLR, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the intermediate trip setpoint Allowable Value defined in the COLR. (Note that the high trip setpoint Allowable Value is less than the intermediate trip setpoint Allowable Value.)

SR 3.3.2.1.4, item c is satisfied if, for an APRM Simulated Thermal Power level ≥ the high power level setpoint Allowable Value defined in the COLR, the RBM flux trip rod block outputs are not bypassed and the RBM flux trip setpoint being applied is less than or equal to the high trip setpoint Allowable Value defined in the COLR.

This SR is performed using APRM STP, which is received digitally from the reference APRM channel. All logic in the RBM is digital. Therefore, consistent with the calibration frequency justified in Reference 12 and the APRM STP calibration SR 3.3.1.1.18 frequency, a frequency of 24 months is selected for this SR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

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1. FSAR, Section 7.7.1.2.8.

2. FSAR, Section 7.6.1.a.5.7

3. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.

4. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

5. NEDO-21231, "Banked Position Withdrawal Sequence," January 1977.

6. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

7. Final Policy Statement on Technical Specifications Improvements, July 22, 1993 (58 FR 32193)

8. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

9. GENE-770-06-1, "Addendum to Bases for changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation, Technical Specifications," February 1991.

10. FSAR, Section 15.4.2.

11. NEDO 33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," April 2003.

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TECH SPEC BASES MARKUP

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12. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October 1995.

13. NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November 1997.

Insert ARTS B4A (A)

14. 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, June 1986.

Attachment 3 to PLA-5931

APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)

Susquehanna Steam Electric Station Units 1 and 2
APRM/RBM/Technical Specifications/
Maximum Extended Load Line Limit Analysis
(ARTS/MELLLA)

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ACRONYMS

Term	Definition
ΔW	Difference between two loop and single loop effective drive flow at the same core flow
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ARTS	APRM/RBM/Technical Specifications
ATWS	Anticipated Transient Without Scram
BOC	Beginning-of-Cycle
BT	Boiling Transition
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CLTP	Current Licensed Thermal Power
COLR	Core Operating Limits Report
CPR	Critical Power Ratio
CRGT	Control Rod Guide Tube
DAR	Design Assessment Report
DBA	Design Basis Accident
DEG	Double Ended Guillotine Break
DIVOM	Delta CPR over Initial MCPR Versus the Oscillation Magnitude
DTPF	Design Total Peaking Factor
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
EOC	End-of-Cycle
FCL	Flow Control Line
FCTR	Flow Control Trip Reference
FIV	Flow-Induced Vibration
FSAR	Final Safety Analysis Report
FWCF	Feedwater Controller Failure
GE	General Electric
HCOM	Hot Channel Oscillation Magnitude
ICA	Interim Corrective Action

Term	Definition
ICF	Increased Core Flow
ICGT	Incore Guide Tube
IORV	Inadvertent Opening of a Relief Valve
IRLS	Idle Recirculation Loop Start-up
JPSL	Jet Pump Sensing Line
LCO	Limiting Condition for Operation
LCRP	Limiting Control Rod Pattern
LFWH	Loss of Feedwater Heating
LHGR	Linear Heat Generation Rate
LHGRFAC	LHGR Multiplier
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPRM	Local Power Range Monitor
LRNBP	Load Rejection with No Bypass
MAPFAC	MAPLHGR multiplier
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MOP	Mechanical Over-Power
MSIV	Main Steam Line Isolation Valve
MSIVC	Main Steam Line Isolation Valve Closure
MSIVF	Main Steamline Isolation Valve Closure with a Flux Scram
MTPF	Maximum Total Peaking Factor
NBP	No Bypass
NFI	New Fuel Introduction
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMAC	Nuclear Measurement Analysis and Control
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor

Term	Definition
PAPT	Protection Against Power Transient
PCT	Peak Cladding Temperature
PRNM	Power Range Neutron Monitor
PRNMS	Power Range Neutron Monitoring System
PRFO	Pressure Regulator Failure Open
RBM	Rod Block Monitor
RCF	Rated Core Flow
RFI	Recirculation Flow Increase
RIPD	Reactor Internal Pressure Difference
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
RWE	Rod Withdrawal Error
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SR	Surveillance Requirement
SRV	Safety-Relief Valve
SRVOOS	Safety-Relief Valve Out of Service
SSES	Susquehanna Steam Electric Station Units 1 and 2
SSV	Spring Safety Valve
STP	Simulated Thermal Power
TLO	Two Loop Operation
TOP	Thermal Over-Power
TPO	Thermal Power Optimization
TRM	Technical Requirements Manual
TS	Technical Specification
TTNBP	Turbine Trip with No Bypass
VPF	Vane Passing Frequency
W_c	% Rated Core Flow
W	% Recirculation Drive Flow

1.0 INTRODUCTION

Many factors restrict the flexibility of a Boiling Water Reactor (BWR) during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition. Once rated power is achieved, periodic adjustments must also be made to compensate for reactivity changes due to xenon effects and fuel burnup. Some of the factors currently existing at the Susquehanna Steam Electric Station (SSES) Units 1 and 2 that restrict plant flexibility in quickly achieving rated power are:

1. The currently licensed allowable operating power/flow map,

2. The Average Power Range Monitor (APRM) flow-biased flux scram and flow-biased control rod block setdown requirements, and

3. The Rod Block Monitor (RBM) flow-referenced rod block trip.

The current licensed Extended Load Line Limit Analysis (ELLLA) power-flow region is replaced by the operating region bounded by the rod line which passes through the 100% of current licensed thermal power (CLTP) / 81.9% of Rated Core Flow (RCF) point, the rated thermal power (RTP) line, and the rated load line, which passes through 100% RCF. The power-flow region that is above the current licensed ELLLA boundary is referred to as the Maximum Extended Load Line Limit Analysis (MELLLA) region. The MELLLA expansion of the power-flow map permits improved power ascension capability by allowing operation at RTP with less than RCF. Figure 1-1 shows a power-flow map with the MELLLA domain.

The operating restrictions resulting from the existing APRM and RBM systems can be significantly relaxed or eliminated by the implementation of several APRM/RBM/Technical Specifications (ARTS) improvements. These improvements increase plant-operating efficiency by improving the thermal limits administration. The operating flexibility associated with the ARTS improvements complement the expansion of the operating domain to the MELLLA boundary. In addition, the NUMAC PRNMS is planned to be installed (Reference 39); it upgrades the electronic components of the APRM and RBM. The improvements associated with ARTS, along with the objectives attained by each improvement, are as follows:

1. A power-dependent Minimum Critical Power Ratio (MCPR) thermal limit, similar to that used by BWR6 plants, is implemented as an update to reactor thermal limits administration.

2. The APRM trip setdown and Design Total Peaking Factor (DTPF) are replaced by more direct power-dependent and flow-dependent thermal limits to reduce the need for manual setpoint adjustments and to provide more direct thermal limits administration. This improves human/machine interface, improves thermal limits administration, increases reliability, and provides more direct protection of plant limits.

3. The flow-biased RBM trips are replaced by power-dependent trips. The RBM inputs are reassigned to: improve the response characteristics of the system, improve the response predictability, and reduce the frequency of nonessential alarms.

4. The Rod Withdrawal Error (RWE) analysis is performed in a manner that more accurately reflects actual plant operating conditions, and is consistent with the system changes.

5. Operability requirements are redefined to be consistent with the modified configuration and supporting analyses.

This report presents the results of the safety analyses and system response evaluations performed for operation of SSES for the permissible operational regions of the P/F map for Unit 2 Cycle 13 implementation with Framatome ATRIUM™-10[1] fuel.

[1]ATRIUM™-10 is a trademark of Framatome ANP

1.1 Background

SSES is a standardized 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 (thus, the APRM flow-biased rod block and scram protection functions). Therefore, these APRM flow-biased setpoint values originate with a deterministic overpower. Later, with the change to the MCPR thermal margin basis under which SSES was originally licensed, studies concluded that the Safety Limit MCPR (SLMCPR) would be met for the design basis transients with the peaking restrictions being conservative for off-rated transients. The SSES Final Safety Analysis Report (FSAR) includes the results of rated power transients, which establish the Operating Limit MCPR (OLMCPR).

The 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 over the years 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 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 ELLLA (Reference 1).

Plants with full ARTS/MELLLA including ICF implementation are: Hatch Units 1 and 2, Cooper, Pilgrim, Fermi, Monticello, Brunswick Units 1 and 2, Peach Bottom Units 2 and 3, and Browns Ferry Units 2 and 3, Duane Arnold (no ICF). Plants with partial (i.e., excluding the RBM modifications) ARTS/MELLLA including ICF implementation are: Dresden Units 2 and 3, Quad Cities Units 1 and 2, and Vermont Yankee. The Hope Creek Generating Station has a partial ARTS submittal currently under review.

SSES has performed 2 power uprates. The first uprate, termed a Stretch Uprate, increased the licensed thermal power by approximately 4.5% (References 1, 2, and 3). 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 (Reference 4). The key thermal power levels are as follows:

- The Original Licensed Thermal Power (OLTP) is 3293 MWt.
- The Stretch Uprate Licensed Thermal Power is 3441 MWt.
- The CLTP and RTP is the Appendix K Uprate Power, which is 3489 MWt.
- The 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.

1.2 ARTS/MELLLA Bases

1.2.1 Analytical Bases

The power/flow operating map (Figure 1-1) includes the operating domain changes for ARTS/MELLLA consistent with approved operating domain improvements for other BWRs. This performance improvement application expands the operating domain along the approximate 114% rod line to 100% of RTP at 81.9% of RCF. The 114% rod line is defined relative to a 100% rod line that intercepts 100% of RTP at 100% of RCF. The 114% rod line was determined by adjusting the typical 120.8% MELLLA boundary line by the ratio of SSES Original Licensed Thermal Power (3293 MWt) to Current Licensed Thermal Power (3489 MWt). This operating domain is defined by the following boundary:

- The MELLLA boundary line, extended up to the existing maximum RTP of 3489 MWt. The MELLLA boundary is defined as the line that passes through the 100% of RTP / 81.9% of RCF state point.
- The currently analyzed Increased Core Flow (ICF) condition of 108.0% of RCF.

The MELLLA boundary line defines an increase in the current operating domain above the current boundary. The current boundary is the uprated ELLLA boundary, corresponding to the 108% APRM Rod Block setpoint, and allows operation to approximately 108% of the rod line that intercepts 100% of RTP at 100% of RCF.

The currently analyzed power level used for Single Loop Operation (SLO) is 2652 MWt.

When compared to the current power/flow operating domain, the MELLLA region allows a higher core power at core flows below 87 Mlb/hr. This increases the fluid subcooling in the reactor vessel downcomer and changes the power distribution in the core, which can potentially affect the steady-state operating thermal limit and transient/accident analyses results. The effect of the MELLLA operating domain has been evaluated to support compliance with the Technical Specification fuel thermal margins during plant operation. This report presents the results of the safety analyses and system response evaluations performed for operation of SSES in the region above the uprated ELLLA and up to the MELLLA boundary line. The scope of the analyses performed covers the initial application for SSES operation with ARTS/MELLLA. For subsequent reload cycles, SSES will include the ARTS/MELLLA operating condition in the reload licensing basis.

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 Anticipated Operational Occurrences (AOOs) MCPR calculation and the reactor vessel overpressure protection analysis are fuel dependent. These analyses, discussed in this report, 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 Reference 5 were reviewed for the MELLLA region. For the fuel-dependent evaluations of reactor pressurization events, these reviews indicate that there is a small difference in the operating limit minimum critical power ratio (OLMCPR) 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 SSES meets the licensing criteria in the expanded MELLLA operating domain.

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.

1.2.2 APRM Flow-Biased Simulated Thermal Power Scram and Rod Block Design Bases

The APRM Flow-Biased Simulated Thermal Power (STP) scram line is conservatively not credited in any SSES licensing analyses. In addition, the APRM Flow-Biased STP rod block line is conservatively not credited in any SSES safety licensing analyses, although it is part of the SSES design configuration. This section discusses the limit changes for these systems for operational flexibility purposes and provides inputs to the SSES Technical Specifications changes.

For the current power/flow map, the APRM Flow-Biased STP scram line Analytical Limit (AL) for TLO is defined as: 0.58 W_d + 65%, and for SLO, 0.58 W_d + 60%, of RTP. The APRM Flow-Biased STP Scram clamp is at 118%. W_d is defined as the recirculation drive flow for two loop operation (TLO) in percent of rated, where 100% drive flow is that required to achieve 100% core power and flow. The APRM Flow-Biased rod block line analytical limit (AL) is currently set at: for TLO 0.58 W_d + 56%, and for SLO 0.58 W_d + 51%, with the APRM Flow-Biased rod block clamp at 113.5%. The current clamp values are unchanged by the implementation of ARTS/MELLLA.

The form of the flow-biased equations has been changed to be consistent with the PRNMS input requirements. The current W_d term will be replaced by the term $W_d - \Delta W$. Using this form of the equation allows for an equivalent change in setpoints between TLO and SLO operation by changing the value of ΔW instead of changing the intercept term of the equation.

With the current power/flow map, the operational margin between the APRM Flow-Biased STP rod block line and the uprated ELLLA Boundary line is significantly reduced, in comparison to the operational margin originally available with respect to the 100% rod line. With the proposed MELLLA power/flow map expansion, the upper boundary of the licensed operating domain is extended to approximately the 120.8% rod line. To accommodate this expanded operating domain and to restore the original margin between the MELLLA boundary line and the APRM Flow-Biased STP rod block line, the following analytical limits are redefined:

Analytical Limit		TLO	SLO
APRM Flow-Biased STP High Scram	Flow-Biased Equation*	$0.62(W_d - \Delta W) + 67.0\%$ $= 0.62\ W_d + 67.0\%$	$0.62(W_d - \Delta W) + 67.0\%$ $= 0.62\ W_d + 61.6\%$
APRM Flow-Biased STP Rod Block	Flow-Biased Equation*	$0.62(W_d - \Delta W) + 62.5\%$ $= 0.62\ W_d + 62.5\%$	$0.62(W_d - \Delta W) + 62.5\%$ $= 0.62\ W_d + 57.1\%$

* ΔW is the difference in the percent flow between the TLO and SLO Recirculation drive flow at the same core flow. The TLO ΔW is 0% and the SLO ΔW is 8.62%.

The above ALs were determined using standard GE methodology.

The following changed Allowable Values (AVs) are determined from the above ALs.

Allowable Value		TLO	SLO
APRM Flow-Biased STP High Scram	Flow-Biased Equation	$0.62(W_d - \Delta W) + 64.2\%$ $= 0.62\ W_d + 64.2\%$	$0.62(W_d - \Delta W) + 64.2\%$ $= 0.62\ W_d + 58.8\%$
APRM Flow-Biased STP Rod Block	Flow-Biased Equation	$0.62(W_d - \Delta W) + 59.7\%$ $= 0.62\ W_d + 59.7\%$	$0.62(W_d - \Delta W) + 59.7\%$ $= 0.62\ W_d + 54.3\%$

The Rod Block Monitor (RBM) Upscale Flow-Biased rod block line limits are currently set at: the TS AVs are: TL0, $0.58W_d + 55\%$ and SLO, $0.58W_d + 50\%$; AL values are; TL0, $0.58W_d + 58\%$ and SLO, $0.58W_d + 53\%$. The ARTS implementation changes the form of the RBM from a flow-biased to a power dependent function. In Section 4.3, Rod Withdrawal Error Analysis, the evaluation of the RWE event was performed taking credit for the mitigating effect of the power-dependent RBM. The power-dependent RBM ALs and AVs are presented in Table 4-6. The RBM ALs were determined using Framatome methodology. The RBM AV revisions were performed using GE instrument setpoint methodology (Reference 41). This methodology complies with ISA Standard S67.04 and with Method 2 of ISA-RP67.04, Part II.

1.3 APRM Improvements

The functions of the APRM are integrated within the NUMAC PRNMS (Reference 6) and are effectively the same as in the previous system, but are performed in an improved digital system. The APRM design functions:

1. Generate trip signals to automatically scram the reactor during core-wide neutron flux transients before the neutron flux level exceeds the safety analysis design bases. This prevents exceeding design bases and licensing criteria from single operator errors or equipment malfunctions.
2. Block control rod withdrawal before core power approaches the scram level when operation occurs in excess of set limits in the power/flow map.
3. Provide an indication of the core average power level of the reactor in the power range.

The NUMAC PRNMS APRM calculates a reactor power level from the Local Power Range Monitor (LPRM) chamber signal such that the APRM signal is proportional to the core average neutron flux and can be calibrated as a means of measuring core thermal power. The APRM signals are used to calculate the STP that closely approximates reactor thermal power during a transient. The STP signals are compared to a recirculation drive flow-referenced scram and a recirculation drive flow-referenced control rod withdrawal block.

The plant currently operates such that the core Maximum Total Peaking Factor (MTPF) does not exceed the Design Total Peaking Factor (DTPF), which limits the maximum local power at lower core power and flows to a fraction of that allowed at rated power and flow. Maintaining the MTPF less than the DTPF is accomplished through the use of a "T-Factor" which is defined in the SSES COLR as the Fraction of Rated Thermal Power (FRTP) divided by the Maximum Fraction of Limiting Power Density (MFLPD). If the T-Factor is calculated to be less than one, the flow-referenced APRM trips must be lowered (setdown) to limit the maximum power that the plant can achieve. The basis for this "APRM trip setdown" requirement originated under the original BWR design Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion (Reference 42) and provides conservative restrictions with respect to current fuel thermal limits.

The change to a critical power correlation, with its emphasis on bundle critical power rather than local critical heat flux allows for a more direct determination of fuel thermal limits.

The SSES ARTS/MELLLA application utilizes the results of the AOO analyses to define initial condition operating thermal limits, which conservatively ensure that all licensing criteria are satisfied without DTPF and setdown of the flow-referenced APRM scram and rod block trips.

Two licensing areas that can be affected by the elimination of the APRM trip setdown and DTPF requirement are: (1) fuel thermal-mechanical integrity, and (2) LOCA analysis.

The following criteria ensure satisfaction of the applicable licensing requirements for the elimination of the APRM trip setdown requirement:

1. The SLMCPR shall not be violated as a result of any AOO.
2. All fuel thermal-mechanical design bases shall remain within the licensing limits described in the Framatome ANP, Inc. (FANP) generic fuel licensing report (Reference 31).
3. Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The safety analyses used to evaluate the OLMCPR are documented in Section 3.0 of this report. These analyses ensure that the SLMCPR and the fuel thermal-mechanical design bases are satisfied. These analyses also establish the cycle-specific power-dependent and flow-dependent MCPR limits and LHGRFAC multipliers for SSES. The effect on the LOCA response due to the ARTS program implementation is discussed in Section 7.0 of this report.

1.4 RBM Improvements

The function of the RBM system is to assist the operator in safe plant operation by:

1. Initiating a rod block to prevent violation of the fuel safety limit MCPR during withdrawal of a single control rod.
2. Providing a signal to permit operator evaluation of the change in local relative power during the movement of a single control rod.

The ARTS improvement makes several changes to the RBM system. A discussion of the current RBM system configuration and the ARTS modification is included in Section 4.0.

The modifications to the RBM system are supported by new RWE analyses to determine the MCPR requirements and the corresponding RBM setpoints, (Section 4.0).

Table 1-1 Computer Codes Used for ARTS/MELLLA Analyses

Task	Computer Code	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y (1)	NEDE-24011-P Rev. 0 SER
Reactor Core and Fuel Performance	CASMO-4 MICROBURN-B2	UNOV0 3 UAUG0 4	Y Y	EMF-2158(P)(A) Rev. 0 EMF-2158(P)(A) Rev. 0
Transient Analysis	MICROBURN-B2 XCOBRA COTRANSA2 XCOBRA-T RODEX2	UAUG0 4 UOCT03 UJUL04 UOCT03 UAPR02	Y Y (7) Y (8) Y (8) Y	EMF-2158(P)(A) Rev. 0 XN-NF-80-19(P)(A) Vol. 3 Rev. 2 ANF-913(P)(A) Vol. 1 Rev. 1 XN-NF-84-105(P)(A) Vol. 1 XN-NF-81-58(P)(A) Rev. 2
Stability Analysis	STAIF RAMONA5-FA	UOCT04 USEP04	Y (9)	EMF-CC-074(P)(A) Vol. 4 Rev. 0 BAW-10255 (P) Revision 0
ECCS-LOCA	RELAX HUXY RODEX2	UAUG0 2 UJAN01 UAPR02	Y Y Y	EMF-2361(P)(A) Rev. 0 XN-CC-33(P)(A) Rev. 1 XN-NF-81-58(P)(A) Rev. 2
Containment System Response	M3CPT LAMB PICSM ISCOR	05 08 01 09	Y (2) Y Y (1)	NUREG-0661 and NUREG-0661, Supplement 1 NEDE-20566P-A NUREG-0487 NEDE-24011-P Rev. 0 SER
Annulus Pressurization-Mass and Energy Releases	LAMB ISCOR	08 09	(2) Y (1)	NEDE-20566P-A NEDE-24011-P Rev. 0 SER
Reactor Internal Pressure Differences	TRACG ISCOR	02 09	(3) Y (1)	NEDE-32176P, Rev. 2 NEDC-32177P, Rev. 2 NRC TAC No. M90270 NEDE-24011-P Rev. 0 SER
Anticipated Transient Without Scram	PANAC ODYN STEMP TASC ISCOR	11 10 (5) 04 03 09	Y (4) Y (6) Y Y (1)	NEDC-24154P-A, Vol 4, Sup 1 NEDC-32084P-A Rev. 2 NEDE-24011-P Rev. 0 SER

Notes:

(1) The ISCOR code is not approved by name. However, the Safety Evaluation Report (SER) supporting approval of NEDE-24011-P Rev. 0 by the May 12, 1978 letter from D. G. Eisenhut (NRC) to R. Gridley (GE) finds the models and methods acceptable, and mentions the use of a digital computer code. The referenced digital computer code is ISCOR. The use of ISCOR to provide core thermal-hydraulic information in reactor internal pressure differences, Transient, ATWS, Stability, and LOCA applications is consistent with the approved models and methods.

(2) The LAMB code is approved for use in Emergency Core Cooling System (ECCS) LOCA applications (NEDE-20566P-A). While there is no approving SER for the use of LAMB in the evaluation of containment system response (see Section 8.2), or Annulus Pressurization – Mass and Energy Releases (see Section 8.5), the use of LAMB for these applications is consistent with the model description of NEDE-20566P-A, "General Electric Model for LOCA Analysis in Accordance with 10CFR50 Appendix K," September 1986.

(3) NRC has reviewed and accepted the TRACG application for the flow-induced loads on the core shroud as stated in NRC SER TAC No. M90270.

(4) The physics code PANACEA provides inputs to the transient code ODYN. The use of PANAC Version 11 in this application was initiated following MFN-035-99, S. Richards (NRC) to G. Watford (GE), Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, "GESTAR II" – Implementing Improved GE Steady State Methods (TAC No. MA6481), November 10, 1999.

(5) Version 10 of ODYN is applicable to plants that use Recirculation Pump speed control for recirculation flow control.

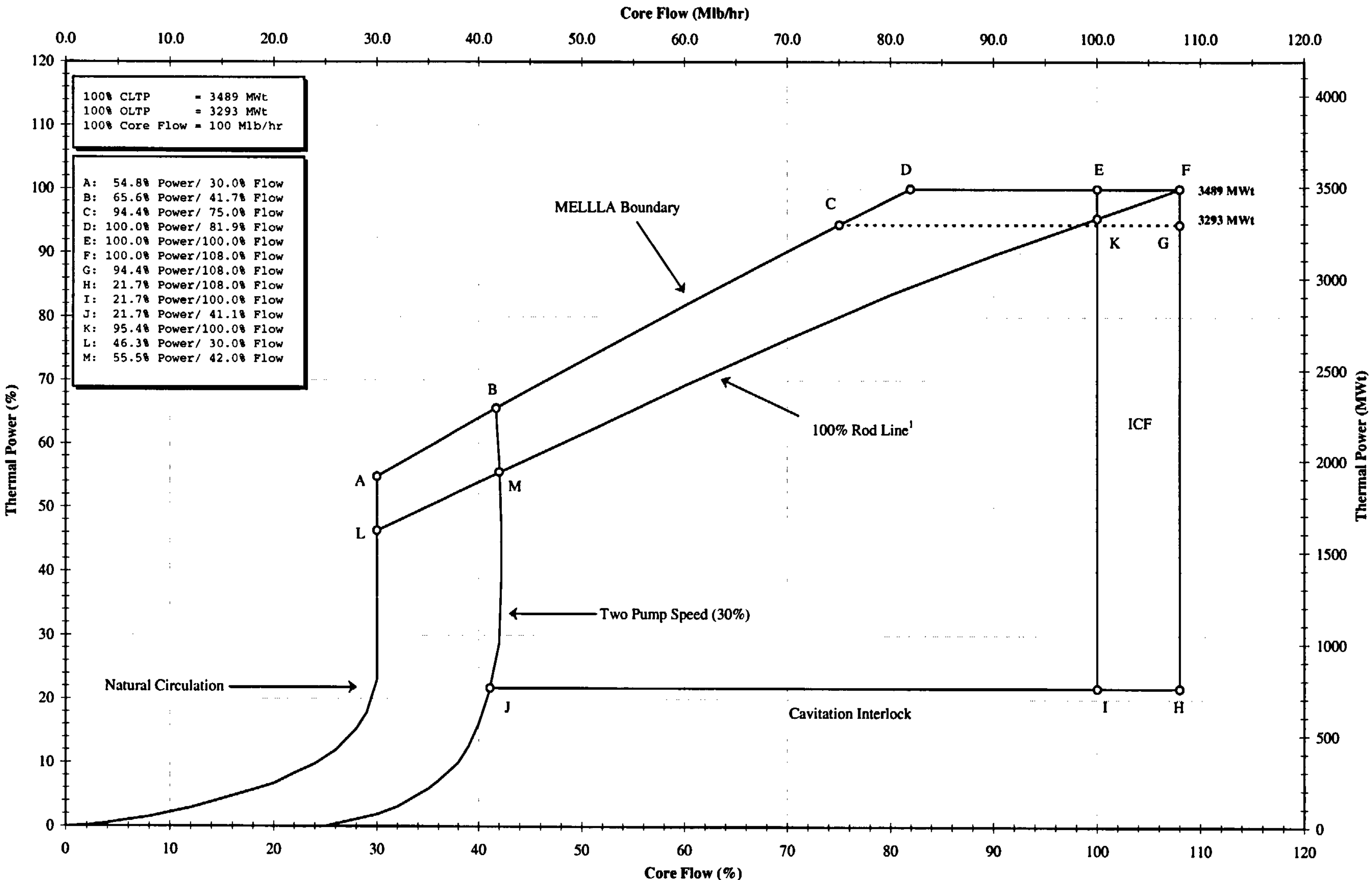
(6) The STEMP code uses fundamental mass and energy conservation laws to calculate the suppression pool heatup. The use of STEMP was noted in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume I & II (NUREG-0460 Alternate No. 3) December 1, 1979." The code has been used in ATWS applications since that time. There is no formal NRC review and approval of STEMP or the ATWS topical report.

(7) The approval of XCOBRA is included in the approval of the THERMEX methodology in Reference 20.

(8) The list of events for which COTRANSA2 and XCOBRA-T can be used was expanded in the clarification acceptance in Reference 28.

(9) RAMONA5-FA is currently being used for determining OPRM setpoints at each reload.

Figure 1-1 ARTS/MELLLA Power/Flow Map



[1] The 100% Rod Line intersects maximum power at maximum core flow.

2.0 OVERALL ANALYSIS APPROACH

This section identifies the analyses that may be affected by the proposed MELLLA region. The analyses performed in the following sections are based on the current plant operating parameters. For the transient and stability tasks, the SSES Unit 2 Cycle 13 core design was utilized. These tasks will be revalidated as part of the subsequent unit-cycle-specific reload licensing analyses. The remainder of the ARTS/MELLLA scope of work is applicable to SSES, unless there is a plant configuration change that affects the analysis.

Table 2-1 identifies the safety and regulatory concerns that are potentially affected as a result of ARTS/MELLLA. Each applicable safety and regulatory concern implied in the listed items was reviewed to determine the acceptability of changing the power/flow map to include the MELLLA range. In addition, the characteristics of each analysis, whether generic or plant-specific, and cycle-dependent or cycle-independent, are identified in Table 2-2.

Table 2-1 Analyses Presented In This Report

Section	Item	Result
3.0	Fuel Thermal Limits	Acceptable – Based on Limits Presented in Section 3.0
4.0	Rod Withdrawal Error Analysis	Acceptable for Unit 2 Cycle 13 Core
5.0	Vessel Overpressure Protection	Acceptable - Below ASME Limit
6.0	Thermal-Hydraulic Stability	Acceptable – New Region for ARTS/MELLLA
7.0	LOCA Analysis	Acceptable for ATRIUM™-10 fuel
8.0	Containment Response	Acceptable - Bounded by Current Results or Design Criteria
9.0	Reactor Internals Integrity	Acceptable - Bounded by Current Results or Design Criteria
1.0	ATWS	Acceptable - Bounded by Current Results or Design Criteria
11.0	Steam Dryer and Separator Performance	Acceptable - Bounded by Design Criteria
12.0	High Energy Line Break	Acceptable - Bounded by Current Results or Design Criteria
13.0	Testing	Acceptable with the performance of the identified tests

Table 2-2 Applicability of Analyses

Task Description	Generic or Plant-Specific	Cycle-Independent or Cycle-Dependent
Power-dependent MCPR limits and LHGRFAC multipliers	Plant-specific	Cycle-dependent
Flow-dependent MCPR limits and LHGRFAC multipliers	Plant-specific	Cycle-dependent
RBM power-dependent setpoints	Plant-specific	Cycle-dependent
ECCS-LOCA	Plant-specific	Cycle-independent unless change in plant configuration from licensing analysis basis

3.0 FUEL THERMAL LIMITS

The potentially limiting AOOs and accident analyses were evaluated to support SSES 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 are presented in Table 3-1 and Table 3-2. These state points include the MELLLA region and the current licensed operating domain for SSES. The AOO evaluations are discussed in Sections 3.1 through 3.2. Section 3.3 discusses the governing MCPR limits and LHGRFAC multipliers. Section 4.0 includes consideration of the RWE analyses and the LOCA analyses are presented in Section 7.0.

3.1 Limiting Core-Wide AOO Analyses

The core-wide AOOs included in the current Unit 2 Cycle 13 reload licensing analyses (Reference 29) and the SSES Final Safety Analysis Report (FSAR) (Reference 5) 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:

(1) Generator Load Rejection with No Bypass (LRNBP) event;

(2) Turbine Trip with No Bypass (TTNBP) event;

(3) Feedwater Controller Failure (FWCF) maximum demand event;

(4) Loss of Feedwater Heating (LFWH) event;

(5) Fuel Loading Error (FLE) event;

(6) Inadvertent High Pressure Coolant Injection (HPCI) Startup event;

(7) 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. The LRNBP and TTNBP events were conservatively combined as one event LRNBP/TTNBP.

The analytical method used by FANP for the SSES evaluations was consistent with the basis used in Reference 32. The results from the SSES 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 Unit 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 Unit 2 Cycle 13, analyses performed at 100% CLTP showed that there is a large margin for OLMCPR (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 Unit 2 Cycle 13 considered the flow range for the MELLLA region. The limiting result for SSES Unit 2 Cycle 13 analyses performed at 100% CLTP showed a large margin for OLMCPR (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 LHGRFAC multipliers established to protect a slow flow excursion, Sections 3.3.3 and 3.3.4.

3.1.1 Elimination of APRM Trip Setdown and Design Total Peaking Factor (DTPF) Requirement

Extensive transient analyses at a variety of power and flow conditions were performed for SSES Unit 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 (P_{Bypass}) where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. The analytical value of P_{Bypass} for SSES 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 Technical Specification 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 the SSES Units is set to actuate down to 40% power. However, it was concluded that Generator Output Breakers do provide a 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 cycle-specific evaluations were performed to establish power-dependent MCPR limits and LHGR multipliers (LHGRFAC) for use in the two power ranges (above P_{Bypass} and below P_{Bypass}).

SSES cycle-specific evaluations were performed to establish the flow-dependent MCPR limits and LHGRFAC multipliers.

3.2 Input Assumptions

The maximum power/flow state condition for the operating region analysis is the rated power and maximum flow point (100%P / 108%F). Figure 1-1 shows the power/flow map used in the AOO analyses. Plant heat balance, core coolant hydraulics, and nuclear dynamic parameters corresponding to the rated and off-rated conditions were used for the analysis and reflect the SSES Unit 2 Cycle 13 core configuration (Reference 29). The initial conditions for the AOO analyses at rated and off-rated conditions are presented in Tables 3-1 and 3-2.

AOO analyses were performed for SSES Unit 2 Cycle 13 with the approved reload licensing methodology (Table 1-1). The following assumptions and initial conditions were used in the AOO analyses to bound the MELLLA operational flow/power conditions:

Analytical Assumptions	Bases/Justifications
Initial core flow range of 81% to 108% flow for thermal limits transients at 100% of CLTP	Bounding power/flow state points for MELLLA
Conservative end-of-Cycle 13 nuclear dynamic parameters	Consistent with SSES Unit 2 current licensing bases
The lowest opening setpoint safety-relief valve (SRV) declared Out-of-Service (OOS)	Consistent with SSES current licensing bases
SLMCPR = 1.09	Consistent with SSES Unit 2 current licensing bases
The LFWH, and HPCI events are not limiting at off-rated conditions.	Consistent with SSES Unit 2 current licensing bases

3.3 Analyses Results

The operating limits associated with operation in the MELLLA region are presented in Table 3-3. The operating limits are set to bound the results from the analyses. The MELLLA region will be incorporated into subsequent unit-cycle-specific reload licensing analyses.

3.3.1 Power-Dependent MCPR Limit

The power-dependent MCPR ($MCPR_p$) limits protect against exceeding the SLMCPR during anticipated operational occurrences from full and partial power conditions. The $MCPR_p$ limits are set to be equal to or greater than the sum of the ΔCPR for the limiting event and the calculated SLMCPR.

The $MCPR_p$ limits for the ATRIUM™-10 fuel which protect an SLMCPR of 1.09 are presented in Figure 3-1. The $MCPR_p$ limits presented in Figure 3-1 are based on the results of the Susquehanna Unit 2 Cycle 13 analyses.

In the high power range (between rated power and P_{Bypass}), the trend for the power-dependent MCPR responses for the FWCF event is more severe than other fast pressurization transient severity trends. As power is reduced from the rated condition in this power range, the LRNBP and TTNBP events become relatively less severe because the reduced steam flow rate at low

power results in milder reactor pressurization. However, for the FWCF event, the power decrease results in greater mismatch between runout and initial feedwater flow, resulting in an increase in reactor subcooling and more severe changes in thermal limits during the event. The SSES Unit 2 Cycle 13 specific analyses results used to establish $MCPR_p$ limits at power levels above P_{Bypass} are summarized in Table 3-4.

Below P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on the closure of the turbine stop valve and turbine control valve. Consequently, the scram signal is delayed until the vessel pressure reaches the high pressure scram setpoint. FANP transient analyses show a significant sensitivity to the initial core flow for transients initiated below P_{Bypass}. The SSES Unit 2 Cycle 13 specific analyses results used to establish $MCPR_p$ limits at power levels below P_{Bypass} are summarized in Table 3-5.

The power/flow map for SSES Unit 2 Cycle 13 shows the maximum core flow is 108 Mlbm/hr when core power ≥ 40% rated core power and 60 Mlbm/hr for core power < 40% rated core power. Therefore, core flow rates > 60 Mlbm/hr were not evaluated for power level ≤ P_{Bypass}. Therefore, the $MCPR_p$ and LHGRFACp limits do not need to protect the results for core power less than 40% and core flow greater than 60 Mlbm/hr. This area on the flow/power map for the SSES units is currently a restricted region since it is not analyzed. However, in the future if this region were included in future AOO analyses and the results were acceptable the restriction may be removed.

3.3.2 Power-Dependent LHGRFAC Multipliers

In the absence of the APRM trip setdown requirement, power-dependent LHGR limits, expressed in terms of $LHGRFAC_p$ multipliers, are substituted to ensure adherence to the fuel thermal-mechanical design bases. A power-dependent LHGRFAC multiplier is applied to the LHGR thermal limits when the plant is operating at less than 100% power. The $LHGRFAC_p$ multipliers protect against both fuel melting and cladding strain during anticipated system transients from partial power conditions. The $LHGRFAC_p$ multipliers assure that the PAPT, (Protection Against Power Transient), LHGR limits are not exceeded for ATRIUM™-10 fuel.

The $LHGRFAC_p$ multipliers for ATRIUM™-10 fuel are presented in Figure 3-2. The $LHGRFAC_p$ multipliers presented in Figure 3-2 are based on results from the SSES Unit 2 Cycle 13 analyses. The SSES Unit 2 Cycle 13 specific analyses result used to establish $LHGRFAC_p$ lists are presented in Table 3-6 for power levels above P_{Bypass} and Table 3-7 for power levels below P_{Bypass}.

3.3.3 Flow-Dependent MCPR Limit

Flow-dependent MCPR ($MCPR_f$) limits are established to ensure that the SLMCPR is not exceeded during a slow flow excursion from reduced core flow to a specified maximum core flow. $MCPR_f$ limits were established for ATRIUM™-10 fuel. Table 3-8 shows the initial power/flow state points from which SSES Unit 2 Cycle 13 flow excursions were evaluated. The $MCPR_f$ limits for ATRIUM™-10 fuel which protect an SLMCPR of 1.09 are presented in Figure 3-3.

3.3.4 Flow-Dependent LHGRFAC Multipliers

A flow-dependent multiplier ($LHGRFAC_f$) is applied to the LHGR thermal limits when the plant is operating at less than 100% core flow. $LHGRFAC_f$ multipliers are established to protect against both fuel melting and cladding strain during a slow flow excursion event. The $LHGRFAC_f$ multipliers assure that the PAPT LHGR limits are not exceeded for ATRIUM™-10 fuel. The $LHGRFAC_f$ multipliers are presented in Figure 3-4.

3.3.5 Safety Limit MCPR Adjustment Procedure

The power-dependent and flow-dependent MCPR limits are established on a cycle-specific basis. These limits are established to protect the SLMCPR which is also established on a cycle-specific basis.

3.3.6 Single Loop Operation Adjustment Procedure

Separate limits are established for single-loop operation when more restrictive limits are needed to protect the applicable criteria when the reactor is operating with one of the recirculation loops out-of-service. Separate limits for single-loop operation include MCPR safety limit and power-dependent MCPR limits. Separate flow-dependent MCPR limits and LHGRFAC multipliers are not needed for single-loop operation because the maximum core flow that can be achieved is significantly reduced during single-loop operation; the limits and multipliers established for two-loop operation remain applicable during single-loop operation.

3.4 Conclusion

The SLMCPR, power-dependent and flow dependent MCPR limits and LHGRFAC multipliers will be determined on a cycle-specific basis. At any power/flow state, within the allowed operating domain, all applicable off-rated limits are determined. The most limiting MCPR (maximum of $MCPR_p$ and $MCPR_f$), and the most limiting LHGR (determined from the minimum of $LHGRFAC_p$ and $LHGRFAC_f$) will be the governing limits.

Table 3-1 Base Conditions for ARTS/MELLLA Rated Transient Analyses

	Normal	81%F MELLLA	108%F ICF
Power (MWt / % of CLTP)	3489 / 100	3489 / 100	3489 / 100
Flow (Mlb/hr / % rated)	100 / 100	81 / 81	108 / 108
Steam Flow (Mlb/hr)	14.42	14.42	14.42
FW Temperature (°F)	391	391	391
Core Inlet Enthalpy (Btu/lb)	525	518	527
Dome Pressure (psig)	1034	1034	1034

Table 3-2 Base Conditions for ARTS/MELLLA Off-rated Transient Analyses

	80%P/108%F	60%P/108%F	40%P/108%F	30%P/60%F	25%P/60%F
Power (MWt)	2791	2093	1396	1047	872
Flow (Mlb/hr)	108	108	108	60	60
Steam Flow (Mlb/hr)	11.27	8.22	5.26	3.85	3.17
FW Temperature (°F)	374	352	318	297	285
Core Inlet Enthalpy (Btu/lb)	527	528	530	523	525
Dome Pressure (psig)	1009	988	972	967	964

Table 3-3 MELLLA Transient Analysis Results at CLTP Conditions, Unit 2 Cycle 13

Initial Power / Flow (% Rated) Transient	Peak Neutron Flux (% Initial)	Peak Heat Flux (% Initial)	ΔCPR ATRIUM™-10	MCPR[(a)] ATRIUM™-10	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
100 / 108-EOC						
LRNBP/TTNBP	254	116	0.26	1.35	1285	1277
FWCF	210	117	0.26	1.35	1234	1252
100 / 81-EOC						
LRNBP/TTNBP	205	112	0.25	1.34	1284	1275
FWCF	165	111	0.22	1.31	1234	1251

Notes:
(a) MCPR is the sum of the ΔCPR for this transient and the SLMCPR.

Table 3-4 ARTS Transient Analysis Results – Above P-Bypass

Initial Power / Flow (%Rated)	Transient	ΔCPR ATRIUM™-10 EOC	MCPR[(a)] ATRIUM™-10 EOC	$MCPR_p$ Limit
100 / 108	LRNBP/TTNBP	0.26	1.35	1.36
	FWCF	0.26	1.35	1.36
80 / 108	LRNBP/TTNBP	0.27	1.36	1.45
	FWCF	0.32	1.41	1.45
60 / 108	LRNBP/TTNBP	0.25	1.34	1.58
	FWCF	0.36	1.45	1.58
40 / 108	LRNBP/TTNBP	0.22	1.31	1.71
	FWCF	0.45	1.54	1.71

Notes:
(a) MCPR is the sum of the ΔCPR for this transient and the SLMCPR.

Table 3-5 ARTS Transient Analysis Results – Below P-Bypass

Initial Power / Flow (%Rated)	Transient[a]	ΔCPR ATRIUM™-10 EOC	MCPR[a] ATRIUM™-10 EOC	$MCPR_p$ Limit
30 / 60	LRNBP/TTNBP	0.94	2.03	2.57
	FWCF	1.23	2.32	2.57
25 / 60	LRNBP/TTNBP	0.88	1.97	2.81
	FWCF	1.17	2.26	2.81

Notes:

(a) MCPR is the sum of the ΔCPR for this transient and the SLMCPR.

Table 3-6 ARTS Transient Analysis Results – $LHGRFAC_p$ Above P-Bypass

Initial Power / Flow (%Rated)	Transient	LHGRFAC	Limiting $LHGRFAC_p$
100 / 108	LRNBP/TTNBP	1.00	1.00
	FWCF	1.00	1.00
80 / 108	LRNBP/TTNBP	1.00	0.91
	FWCF	1.00	0.91
60 / 108	LRNBP/TTNBP	1.00	0.82
	FWCF	1.00	0.82
40 / 108	LRNBP/TTNBP	1.00	0.73
	FWCF	0.99	0.73

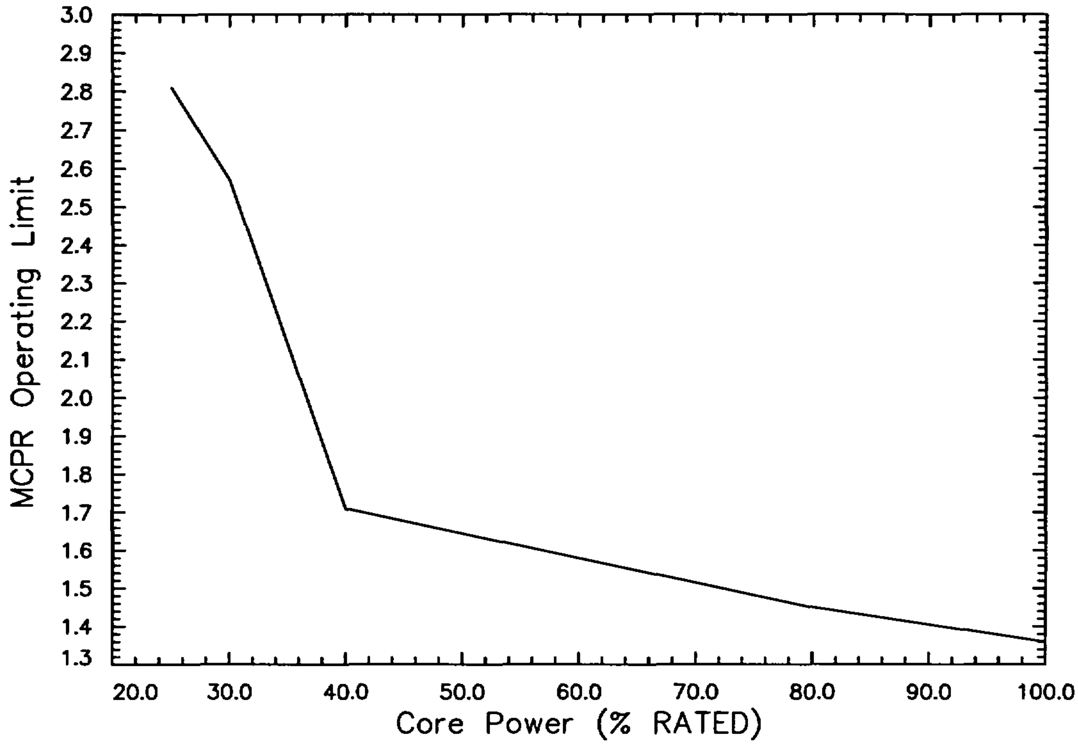
Table 3-7 ARTS Transient Analysis Results – $LHGRFAC_p$ Below P-Bypass

Initial Power / Flow (%Rated)	Transient	LHGRFAC	Limiting $LHGRFAC_p$
30 / 60	LRNBP/TTNBP	0.87	0.52
	FWCF	0.71	0.52
25 / 60	LRNBP/TTNPB	0.83	0.47
	FWCF	0.68	0.47

Table 3-8 ARTS Rod Line for $MCPR_f$ and $LHGRFAC_f$ Determination

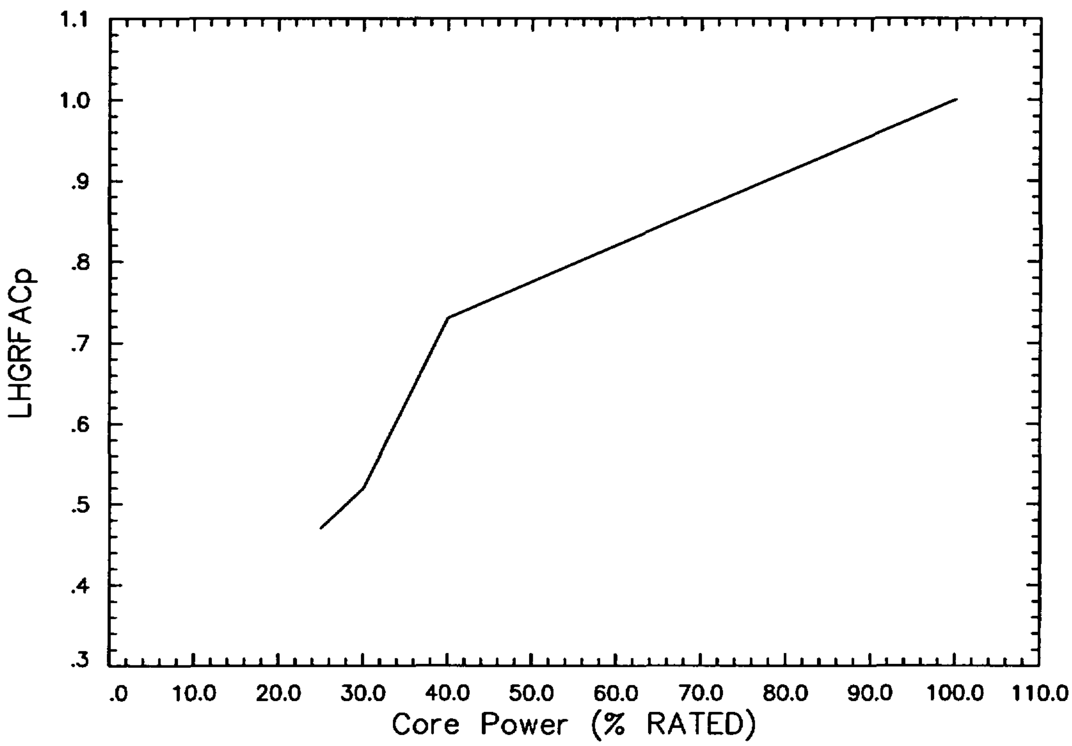
Low Power / Flows		100% Xe Rod Line		MELLLA Line	
Power (% rated)	Flow (Mlbm/hr)	Power (% rated)	Flow (Mlbm/hr)	Power (% rated)	Flow (Mlbm/hr)
36	30	60	48	82	60
48	40	86	84	91	71
		100	108	100	82

Figure 3-1 Power-Dependent MCPR Limits, $MCPR_p$



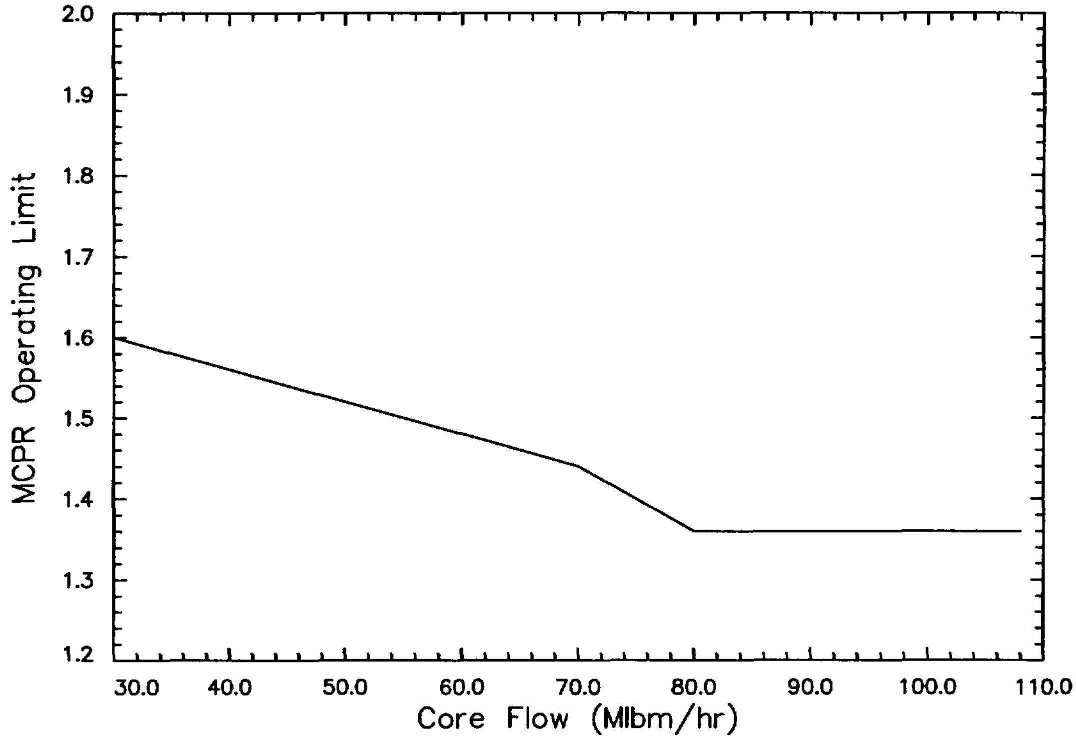
Core Power (% Rated)	$MCPR_p$	Core Flow (Mlbm/hr)
100.0	1.36	All Flows
80.0	1.45	
40.0	1.71	
30.0	2.57	≤ 60 Mlbm/hr
25.0	2.81	

Figure 3-2 Power-Dependent LHGRFAC Multiplier



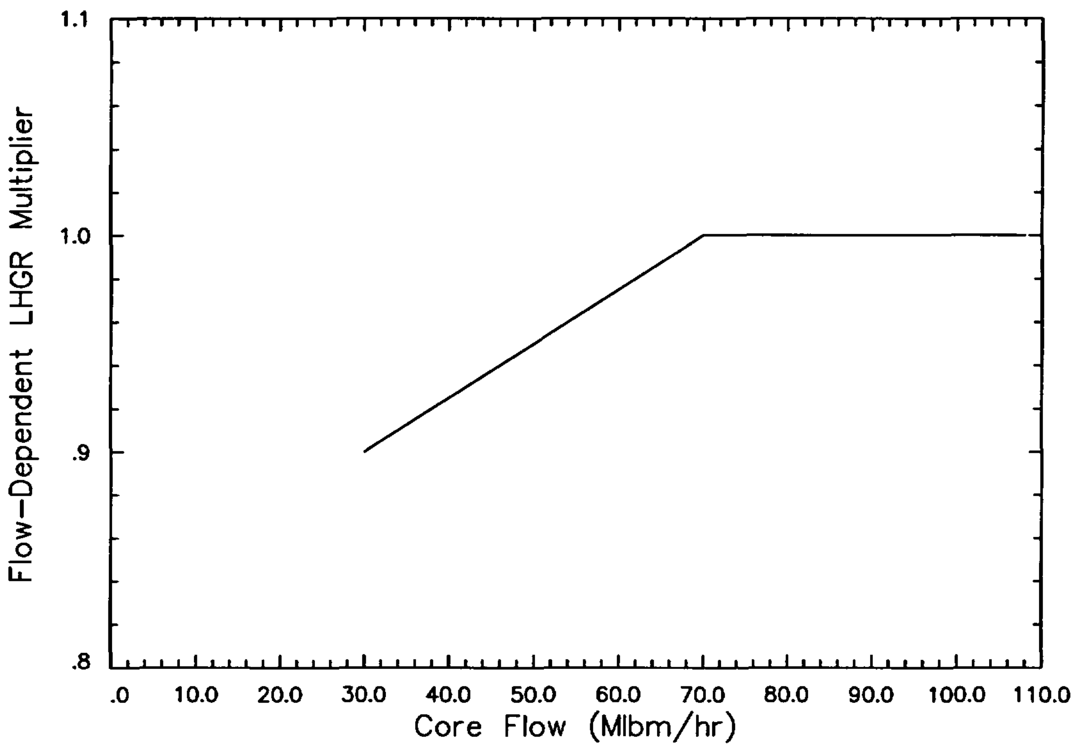
Core Power (% Rated)	LHGRFACp	Core Flow (Mlbm/hr)
100.0	1.00	All Flows
40.0	0.73	
30.0	0.52	≤ 60 Mlbm/hr
25.0	0.47	

Figure 3-3 Flow-Dependent MCPR Limits, $MCPR_f$



Core Flow (% Rated)	$MCPR_f$
108.0	1.36
80.0	1.36
70.0	1.44
30.0	1.60

Figure 3-4 Flow-Dependent LHGRFAC Multiplier



Core Flow (Mlbm/hr)	$LHGRFAC_f$
108.0	1.00
70.0	1.00
30.0	0.90

4.0 ROD BLOCK MONITOR SYSTEM DESCRIPTION AND RWE ANALYSIS

The function of the Rod Block Monitor (RBM) System is to assist the operator in safe plant operation in the power range by:

- initiating a rod block to prevent violation of the fuel integrity safety criteria during withdrawal of a single control rod, and
- providing a signal to permit operator evaluation of the change in local relative power during control rod movement.

This section provides a discussion of the RBM System evaluation and features provided by the ARTS improvement, including the Rod Withdrawal Error (RWE) analysis based on the improved RBM system.

4.1 Current RBM System

4.1.1 Current System Description

To provide the measure of local power change, the RBM System uses the set of Local Power Range Monitors (LPRMs) that is displayed to the reactor operator on the four-rod display. There are two RBM circuits (designated Channel A and Channel B); one uses the LPRM readings from the A&C level detectors and the other uses the B&D level detectors. The RBM has between four and eight LPRM inputs, depending on whether it is operating on an interior or peripheral rod.

The RBM computes the analog average of all assigned unbypassed LPRMs in much the same manner as the APRM. If the average of the RBM input reading is less than the reference APRM signal, then an automatic RBM gain adjustment occurs such that the average RBM reading is equal to, or greater than the APRM reading (this gain adjustment factor can never be less than one). This comparison and potential RBM gain adjustment occurs whenever a control rod is selected. There is a momentary rod block while the gain adjustment is made. This gain is held until a new control rod is selected.

The RBM automatically limits the local thermal power changes by allowing the local average neutron flux indications to increase by a controlled amount. If the change is too large, the rod withdrawal permissive is removed. Only one of the two RBM channels is required to trip to prevent rod motion.

The RBM has three drive flow-biased trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. Typical BWR4 settings are 110%, 102%, and 94% at 100% flow. Each trip level is automatically varied with recirculation system flow to protect against fuel overpower at lower flows. The operator may encounter any number (up to three) of the trip points, depending on the starting power of a given control rod withdrawal. The lower two points may be successively bypassed (acknowledged) by manual operation of a pushbutton. The reset permissive is actuated (and indicated by a light) when the RBM indicates a power 2% less than the trip point. The operator should then assess the local power and either acknowledge or select a new rod. The highest trip point cannot be bypassed.

A count of the active LPRMs is made automatically and the RBM declared inoperative if too few detectors are available for use. The rod withdrawal permissive is removed if the RBM is inoperative and not bypassed. Only one RBM channel may be manually bypassed at any time. If the reference APRM is indicating less than 30% power, the RBM is bypassed automatically. The RBM also is bypassed if the control rod has one or more adjacent fuel bundles located in the outer boundary of the reactor core. In this case, the high neutron leakage prevents overpower conditions. An RBM reading downscale and not automatically bypassed by the APRM low power feature is considered to have failed and the rod withdrawal permissive is not given. The RBM has outputs to recorders located on the reactor operator's console, local meters, trip units, LPRM flux meters, and the on-line computer.

The signal conditioning electronics for the RBM form the average of the LPRM chambers as described above. The detectors are assigned upon selection of a control rod by a selection matrix. The matrix receives a voltage signal corresponding to the selected rod group. The selection of the rod routes the proper LPRM signals to the meter displays and to the assigned RBM.

The power for the RBM is supplied from low voltage power supplies located in the same cabinet as the RBM. Although the RBM has no reactor protection outputs, each RBM channel is assigned to a separate trip system, and the ac power for the RBM low voltage power supply is supplied from independent sources.

The trip unit utilizes the output voltage from a flow converter to drive the linear variation of the trip setpoints with flow. The slope of the rod block trip (analytical value) is variable between 0.52 and 0.78 with a setting of 0.58 for the SSES units, as required for the current ELLLA power/flow map.

One RBM channel may be manually bypassed by operator action. Automatic bypass occurs if the APRM level is below a prescribed value or one of the reactor core outer boundary control rods is selected. All trips are bypassed if the reactor mode switch is in any position other than "RUN".

An illustration of the current SSES RBM System is presented in Figure 4-1.

4.1.2 Limitations of Current RBM System

The SSES RBM System was designed in the middle 1960s. Since that time, there have been significant technological advances in the fields of two-phase heat transfer and electronics. More advanced critical power ratio correlations are now used in place of older critical heat flux ratio correlations. Therefore, the optimum evaluation of fuel thermal margins is not as effective when performed solely on a local basis, but requires information about the entire fuel bundle. 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, and the system performs its function at the expense of significant operational penalties due to the conservatism required by the current system limitations.

The current selection of LPRM inputs that form the RBM signals (Figure 4-2) is not optimum for monitoring fuel integrity criteria because the two RBM channels have significantly different responses to the same control rod movement. For determination of RWE event consequences and the trip setpoints, the most responsive channel is assumed to be bypassed and the setpoints are determined by the operating (least responsive) channel. It is also assumed that some of the LPRMs assigned to the operating channel have failed. This further diminishes the response of this channel. The RBM setpoint chosen is the one which blocks rod withdrawal before violation of the Safety Limit Minimum Critical Power Ratio (SLMCPR) based on the response of the least responsive channel with maximum allowable LPRM failures. However, when this setpoint is implemented at the plant, both RBM channels typically will be in operation and the number of failed LPRMs will be less than assumed in the analysis. The more responsive channel actually blocks rod withdrawal at much shorter withdrawal increments and unnecessarily restricts control rod movements. This results in complicated and time-consuming plant maneuvers to reach the full-power rod pattern. Therefore, the correlation between RBM response and thermal margin change is improved by reassigning the LPRMs making up the two RBM channel signals.

When a control rod is selected, rod withdrawal is blocked by the current RBM until the proper LPRM signals have been routed to the averaging electronics and a variable gain has been applied to the channel responses, which normalizes them to read the same as the reference APRM channels (Figure 4-1). Normalization of the signal and trips to the reference APRM provides a method of mapping RBM setpoints over a broad range of power and flow (Figure 4-3). Three flow-biased trip settings are provided; the one selected is determined by the power and recirculation drive flow at the time of selection. At a given flow, the RBM trip setting immediately above the APRM measured power is selected for enforcement. If the APRM measured power is within the 2% reset band immediately below the two lower trip settings, the next higher RBM trip setting is automatically selected for enforcement. Similarly, manual reset of the lower trip to the next higher trip is allowed when the local power reaches the 2% band as a result of rod withdrawal. In this case, the operator verifies that adequate thermal margins exist before resetting the trips. These reset features are a necessary result of the normalization of the signals to the APRM. If the APRM power is just below the trip, random noise in the signals may cause the trip to be exceeded and no withdrawal will be possible. Since the flow-biased trip settings are roughly parallel to the flow control lines, it would be very difficult to increase core power above an RBM trip setting without the reset features. Resets are possible only for the two lower trip settings; the high trip cannot be reset. Since the highest trip setting cannot be reset, another direct consequence of the normalization of the RBM signals to the reference APRM is that control rod withdrawal is not permitted when the reference APRM exceeds the highest RBM trip setting.

Figure 4-3 illustrates an ideal startup path in which rated power is attained without control rod movement after recirculation flow has been increased above the minimum pump speed.
Figure 4-3 also shows the relationship between the RBM trip settings and the ideal startup path relative to the highest RBM trip setting. Because these two lines cross at low flow, the RBM prevents withdrawal of control rods necessary to attain the ideal startup path. Table 4-1 summarizes the limitations of the current SSES RBM system, the effects of these limitations, and the proposed improvements to the system.

4.2 New RBM System Description

The improved RBM System will:

- Eliminate the restrictions imposed on core power by the current flow-referenced RBM trips (this function is fulfilled by the APRM flow-biased rod block), and
- Enhance operator confidence in the system by reducing the frequency of nonessential rod blocks and by making the occurrence of rod blocks more predictable and therefore avoidable.

Advances in electronics make it possible to efficiently specify system performance requirements that were not possible in the mid-1960s. The ARTS improvement takes advantage of these advances to make changes in the SSES RBM hardware that controls the trip logic and LPRM averaging to enhance the instrumentation accuracy and to improve the signal to thermal margin correlation. These improvements are integrated in the NUMAC Power Range Neutron Monitoring System (PRNMS) that is planned to be installed at the SSES units.

A more direct trip logic is implemented (Figure 4-4). Instead of calibrating to the APRM, the RBM signals are calibrated to a fixed (constant) reference signal. As in the original system, an RBM downscale trip level is defined to detect abnormally low signal levels. The upscale trip levels are set at a fixed level above the reference and will vary as step functions of core power. This will allow longer withdrawals at low powers where thermal margins are high and allow only short withdrawals at high power. Once tripped, recalibration is allowed only by deselecting the rod (typically accomplished by selecting another rod or "Rod Select Clear") and reselecting the rod. Reselection will result in a recalibration to the reference signal.

A number of alternatives to the current LPRM assignment were studied by General Electric. Figure 4-2 illustrates the current LPRM assignments. The new assignment scheme (Figure 4-5) provides the best grouping to achieve the following objectives:

- Similarity of channel responses,
- High response to rod motion (allows higher setpoints, which reduces the effect of random signal noise, calibration inaccuracies, and instrument drift),
- Less restrictive MCPR limits with high setpoints,
- High availability (tolerance of LPRM failures), and
- Ease of implementation.

While the A level LPRMs will no longer be used in the RBM signals, they will remain in place for all other functions and displays. The basis for this is that the A level response has minimum significance for bundle power increases (level A response has significance only for shallow rod withdrawal).

Individual channel responses are compared in Figure 4-6 for a typical high worth control rod withdrawal. This figure demonstrates the high degree of similarity of channel response for the new assignments and the low degree of similarity existing with current assignments.

To the maximum extent possible, while achieving the above objectives, the new RBM System design meets the same requirements as the previous RBM System. The only exceptions are the

sharing of LPRM signals from the "C" level detectors by both RBM channels and the calibration of the RBM signals to isolated, fixed reference signals instead of isolated APRM reference signals. As for the current system, the new RBM System is fail safe for failed LPRM input signals. As for the current system, a count of active LPRMs is made automatically and the RBM channel declared inoperative if too few detectors are available.

The impact on the availability of the new RBM System due to the sharing of the "C" level detectors has been shown to be small compared to the benefits of the improved signal response.

The new RBM System possesses readily predictable behavior, and will limit the thermal margin reduction during rod withdrawals, but does not restrict rod withdrawals on the basis of core power level (see comparison between Figures 4-3 and 4-7). The limitations on core power levels imposed by the APRM flow-biased rod block, remains unchanged.

The RWE evaluations necessary to establish the CPR limit and the trip setpoints for each power interval are discussed in the following subsections.

4.3 Rod Withdrawal Error Analysis

4.3.1 Analysis

The Control Rod Withdrawal Error (CRWE) transient is currently analyzed during the reload fuel licensing analysis for Susquehanna. The Framatome ANP, Inc.[1] (FANP) CRWE methodology which is currently employed for SSES is discussed in Reference 33. The CRWE 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 Minimum Critical Power Ratio (MCPR) and increases the Linear Heat Generation Rate (LHGR) in the core limiting fuel rods. The CRWE transient is terminated by control rod blocks which are initiated by the Rod Block Monitor (RBM) system.

The CRWE analyses are performed with the FANP MICROBURN-B2 reactor simulator code (Reference 19). The ANFB-10 critical power correlation (Reference 34) is used to calculate the MCPR values for the FANP ATRIUM™-10 fuel.

The object of the CRWE analysis is to determine bounding values for the power dependent CRWE 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.

The rod withdrawal calculations are performed in full core geometry and the initial rod patterns are based on the projected rod patterns for the specific cycle being analyzed. The rod withdrawal error rods are selected based on limiting, i.e., minimum CPR margin, fuel assemblies located

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1 Framatome ANP, Inc. is an AREVA and Siemens company.

near the inserted error rods. In calculating the limiting CRWE MCPR and LHGR results as a function of reactor power and RBM set point, the assumed allowable bypass/failure condition is one RBM channel bypassed and not more than one-half of the LPRM detectors bypassed in the RBM channel in service. The CRWE analysis assumes no xenon and assumes that the plant could be operating in either an A or B sequence control rod pattern. Limiting control rod patterns are conservatively used along with power/flow conditions that are representative of operation in the MELLLA domain to generate the thermal limits for the CRWE.

A non-statistical ARTS CRWE analysis has been performed using SSES and fuel related input and methods consistent with the Unit 2 Cycle 13 licensing analysis. Analyzed conditions support plant operation in the Maximum Extended Load Line Limit Analysis (MELLLA) region of the power and flow operating map. The CRWE MCPR results have been tabulated as a function of the RBM setpoints for the ATRIUM™-10 fuel. The Susquehanna ARTS analytical unfiltered RBM setpoints are shown in Table 4-2. The trip setpoints are grouped in sets in Table 4-2 where a given set is selected for a specific cycle (for example, setpoints 108 (HTSP), 113 (ITSP), and 118 (LTSP) are one set). The bounding CRWE MCPR results are shown in Table 4-3 as a function of percent rated power/flow and the RBM setpoint values. The CRWE MCPR values in Table 4-3 are for an SLMCPR value of 1.09. For other SLMCPR values, the CRWE MCPR values are adjusted by the difference in the SLMCPR. The bounding CRWE MCPR results at all power levels are shown as a function of RBM setpoint in Figure 4-8.

The upgraded performance of the ARTS RBM system significantly reduces the severity of the CRWE event when compared to other AOO events for selected RBM setpoints. The LPRM assignments make the ARTS RBM system more sensitive to rod withdrawals. The CRWE MCPR operating limits can be compared with the limiting cycle specific transient $MCPR_p$ limit and SLMCPR to verify that the CRWE is a non-limiting event for a specific set of RBM setpoints. For example, representative Susquehanna Unit 2 Cycle 13 $MCPR_p$ and CRWE MCPR limits are shown on Figure 4-9. A comparison of the curves on Figure 4-9 shows that the CRWE MCPR limit is bounded by the $MCPR_p$ limit for the RBM setpoints of 108, 113, and 118.

At low reactor powers, the CRWE event is far from limiting as shown on Figure 4-9. 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.

The SSES CRWE calculations confirm that the transient LHGR limits for the ATRIUM™-10 fuel are not exceeded in an unblocked CRWE event and therefore cladding strain induced fuel damage and fuel melting are precluded.

4.3.2 Sensitivity Analyses

4.3.2.1 Peripheral Rod Groups

The CRWE results discussed above are based on rod withdrawals occurring in interior four rod cells surrounded by four LPRM strings. The RBM cells near the core periphery may possess fewer than four control rods and have one, two, or three LPRM strings. The location of the LPRM strings and the control rods in the SSES core are shown in Figure 4-11.

Selected cases were evaluated to verify that the CRWE MCPR results for control rods surrounded by four RBM LPRM strings are applicable to control rods that are surrounded by fewer than four RBM LPRM strings. The control rods with less than four LPRM strings in an RBM channel are located near the core periphery where the missing string is located away from the control rod position. The rod block monitor rod group geometries and error rod locations are shown on Figure 4-12.

To evaluate the effect of the number of LPRM strings on the CRWE MCPR results, representative MICROBURN-B2 MCPR results were tabulated for four, three, two, and one LPRM strings. The ΔCPR /initial MCPR responses for the four LPRM string cases and the geometries with fewer LPRM strings are shown on Table 4-4 for a RBM setpoint of 108%. The peripheral geometry RBM response is slightly better (lower ΔCPR/initial MCPR) because the missing LPRM strings are further from the error rod locations. These results show that the CRWE MCPR results documented in this report bound those for a rod withdrawal near the core periphery with fewer than four LPRM strings input to the RBM system.

4.3.2.2 LPRM Failures

The LPRM failure assumptions are not fuel related. The FANP CRWE results shown in this section are for a non-statistical ARTS analysis using the methodology outlined in Reference 33. The limiting MCPR and LPRM response is calculated as a function of control rod withdrawal for the fuel in the core. In calculating the limiting CRWE MCPR results as a function of reactor power, flow, and RBM set point, the assumed allowable bypass/failure condition is one RBM channel bypassed and not more than one-half of the LPRM detectors bypassed in the RBM channel in service. For each CRWE calculation, the RBM response is calculated for each bypass condition and the most limiting condition is selected at each RBM setpoint to obtain the bounding MCPR results.

4.3.2.3 Effect of RBM Signal Filter on CRWE

Optional capability is included to filter the RBM signal to reduce signal noise levels. The filter time constant is adjustable up to a maximum value of 0.5 ± 0.05 seconds. The design of the control rod drive system is for a normal speed of 3 ± 0.6 inches/second. When the filter is utilized, the filtered signal lags the unfiltered signal. For a ramp input, the asymptotic time lag will equal the time constant of the filter.

FANP has evaluated the effect of the signal filter on the ARTS RBM setpoints. The RBM response data including the results for Susquehanna were evaluated with and without a time constant filter. The data base included results at all analyzed power levels. The evaluation was performed for a maximum filter time constant of 0.55 seconds and a maximum control rod withdrawal speed of 3.6 inches/second. The maximum rod withdrawal speed results in the maximum rod withdrawal distance as a function of RBM setpoint with filter. The difference between the filtered and unfiltered setpoints is subtracted from the analytical setpoint values to assure that the CRWE results are valid. The evaluated setpoint reduction results are summarized statistically in Table 4-5. For each evaluated condition, the uncertainty on the setpoint result is small.

The FANP analysis supports the following setpoint reduction values for filter lag effects:

For RBM setpoints ≤ 108%, subtract 0.6%.

For 108% < RBM Setpoints ≤ 116%, subtract 0.8%.

For 116% < RBM Setpoints ≤ 124%, subtract 1.0%.

For 124% < RBM Setpoints ≤ 127%, subtract 1.2%.

4.4 Filter and Time Delay Settings

The new RBM system provides an adjustable time delay T_{d1} and a filter to allow field optimization of the system to actual signal noise characteristics. The adjustable delay, T_{d1} , is from the time the signal is nulled to the reference signal to the time the signal is passed to the trip logic (rod withdrawal is not restricted during this period). The filter on the RBM signal, T_{c1}, smoothes the averaged LPRM signal to reduce trips due to signal noise. The APRM filtered signal (i.e., known as the simulated thermal power) is input to the power-dependent trip selection logic.

The purpose of the adjustable delay, T_{d1}, is to allow a plant that is within thermal limits to withdraw a control rod at least a single notch despite extremely noisy signals that would normally block rod withdrawal. The design intent of time delay (T_{d1}) is for application with extreme signal noise characteristics. For those cases, the signal noise may be too severe for a filtering system to handle adequately (i.e., the required filter time lag setpoint penalty would result in setpoints too low to be operationally acceptable). Therefore, specifications of standard RBM setpoints coupled with this time delay would assure that at least one 6-inch notch control rod withdrawal could be made on each rod selection. If T_{d1} is utilized, thermal margin analyses are performed based on unrestricted continuous rod withdrawal during the T_{d1} period. The inclusion of this feature is considered totally consistent with the ARTS objective of eliminating unnecessary RBM rod block alarm on normal rod maneuvers with thermal margin to improve the human factors of the RBM System.

The SSES ARTS RBM/RWE licensing bases supports the adjustable RBM filter time constant (T_{c1}) with the applicable setpoint adjustments defined in Section 4.3.2.3. If greater than minimum RBM filtering is utilized, the nominal maximum time constant of 0.5 second is recommended. A set of representative trip setpoints and power intervals, based on Unit 2 Cycle 13 analysis results, are listed in Table 4-6 and depicted in Figure 4-10. Symbols used in Table 4-6 are explained in Table 4-7.

4.5 RBM Operability Requirement

The RBM System design objective is to block erroneous control rod withdrawal initiated by the operator before the safety limit MCPR is violated. When any control rod in the core will violate this limit upon complete withdrawal, operability of the RBM System is required. The RBM System basis is limited to consideration of single control rod withdrawal errors and does not accommodate multiple errors.

Based on the calculated unblocked MCPR results for SSES Unit 2 Cycle 13, the following limiting MCPR values were determined to provide the required margin for full withdrawal of any control rod: The results are based on a 1.09 SLMCPR.

For power < 90% rated: MCPR ≥ 1.71 (dual loop operation)

For power < 90% rated: MCPR ≥ 1.75 (single loop operation)

For power ≥ 90% rated: MCPR ≥ 1.47 (dual loop operation)

Whenever operating MCPR is below the preceding values, the RBM System must be operable; whenever the operating MCPR is above these values, complete RBM bypass is supported. The assumed single loop SLMCPR is 1.11 and greater than 90% power is not attainable with single loop operation. For SLMCPR values different than 1.09 (two loop) and 1.11 (single loop), the above MCPR values for RBM bypass need to be multiplied by the ratio of the SLMCPR values to the appropriate limit of 1.09 or 1.11.

4.6 Conclusion

The NUMAC hardware and Technical Specification implementation of ARTS will:

- Eliminate the restrictions imposed on gross core power by the current flow-referenced RBM trips (this function will be fulfilled by the APRM flow-biased rod block).
- Enhance operator confidence in the system by reducing the frequency of nonessential rod blocks and by making the occurrence of rod blocks more predictable and therefore avoidable.
- Upgrade the performance of the system such that it will be unlikely that the RWE will be the limiting transient. The RWE transient MCPR is determined by the rod block setpoints. These setpoints will be selected based on the OLMCPR, as established by other AOOs.

Table 4-1 Rod Block Monitor System Improvements

Current Design	Effect	Improvements
Non-Optimum LPRM Assignment	Divergent Channel Response Low Trip Setpoints Unnecessary Rod Blocks	Optimize LPRM Assignments
Normalization to APRM	Erratic Trip Setpoints	Normalize Initial Signal to Fixed Reference
Flow-Biased Trips	Unnecessary Rod Blocks	Power-Biased Trips (Like BWR/6) Relative to Fixed Reference
Reset Capability	Gross Core Power Limited	Renormalize on Rod Select Only

Table 4-2 Susquehanna RBM Instrumentation Setpoints

Power Setpoint	Analytical Setting[2]
LPSP	30.0
IPSP	65.0
HPSP	85.0

Analytical RBM Trip Setpoints (Unfiltered)		
LTSP	ITSP	HTSP
118	113	108
121	116	111
124	119	114
127	122	117

See Table 4-7 for function definitions

2 Analytical setpoint in % of reference power level.

Table 4-3 Rod Withdrawal Error Analysis Results

Setpoint	Power/Flow	CRWE MCPROL	Maximum Value
108	100/100	1.34	
108	85/72	1.32	
108	65/45	1.37	1.37
108	40/45	1.28	
111	100/100	1.37	
111	85/72	1.37	
111	65/45	1.41	1.41
111	40/45	1.32	
113	100/100	1.39	
113	85/72	1.39	
113	65/45	1.43	1.43
113	40/45	1.35	
116	100/100	1.41	
116	85/72	1.44	
116	65/45	1.45	1.45
116	40/45	1.41	
121	100/100	1.41	
121	85/72	1.46	
121	65/45	1.50	1.50
121	40/45	1.47	
124	65/45	1.50	
124	40/45	1.71	1.71
127	40/45	1.71	1.71

Table 4-4 CRWE Analysis Results For Peripheral Rod Groups

(108% SETPOINT)

Number of LPRM Strings	Normal Number of LRRM Inputs	Channel A		Channel B	
		Δ CPR/Initial MCPR		Δ CPR/Initial MCPR	
		Mean	Std. Dev.	Mean	Std. Dev.
4	8	0.086	0.014	0.080	0.012
3 (Case 1)	6	0.075	0.012	0.075	0.012
3 (Case 2)	6	0.084	0.016	0.072	0.010
2	4	0.077	0.021	0.070	0.010
1	2	0.071	0.025	0.050	0.019

Table 4-5 RBM Signal Filter Setpoint Adjustment

Power Level (%)	RBM Channel	Number of times evaluated	RBM Setpoint (%)	Mean Signal Difference Where Unfiltered Signal equals Setpoint	Standard Deviation of Difference
100	1	1056	108	0.0048	0.0009
100	2	1056	108	0.0052	0.0011
40	1	767	118	0.0081	0.0023
40	2	766	118	0.0078	0.0022

Table 4-6 RBM System Setup Without RBM Filter

	Trip Level Setting (Note a)	
Function	Analytical Limit (AL)	Allowable Value (AV)
LPSP	30.0	28.0
IPSP	65.0	63.0
HPSP	85.0	83.0
LTSP	118.0	115.6
ITSP	113.0	110.6
HTSP	108.0	105.6
DTSP	N/L(Note b)	N/L(Note b)
T_{d1}	N/L(Note b)	N/L(Note b)
T_{c1}	N/L(Note b) Filtered (≤ 0.55 s) Unfiltered (0 s)	N/L(Note b) Filtered (≤ 0.55 s) Unfiltered (0 s)

Note (a): Trip Setpoint function numbers in % of Reference Level. Power Setpoint function numbers in % Rated Thermal Power.

Note (b): N/L - No Limitations; means either that the setpoint does not affect the RWE analysis or that the range is restricted by design to values considered in the RWE analysis.

Table 4-7 RBM Setup Setpoint Definitions

AL	Analytical limit
AV	Allowable value
NTSP	Nominal trip setpoint
LPSP	Low power setpoint; RBM trips automatically bypassed below this level
IPSP	Intermediate power setpoint
HPSP	High power setpoint
LTSP	Low trip setpoint
ITSP	Intermediate trip setpoint
HTSP	High trip setpoint
DTSP	Downscale trip setpoint
T_{d1}	Adjustable Time delay that delays passing RBM filter signal to RBM trip logic after signal has been nulled successfully to reference signal.
T_{c1}	Adjustable RBM signal filter time constant. Adjustment within the hardware capability must be consistent with the basis of the setpoints.
Reference Level	The level the RBM is automatically calibrated to upon control rod selection.

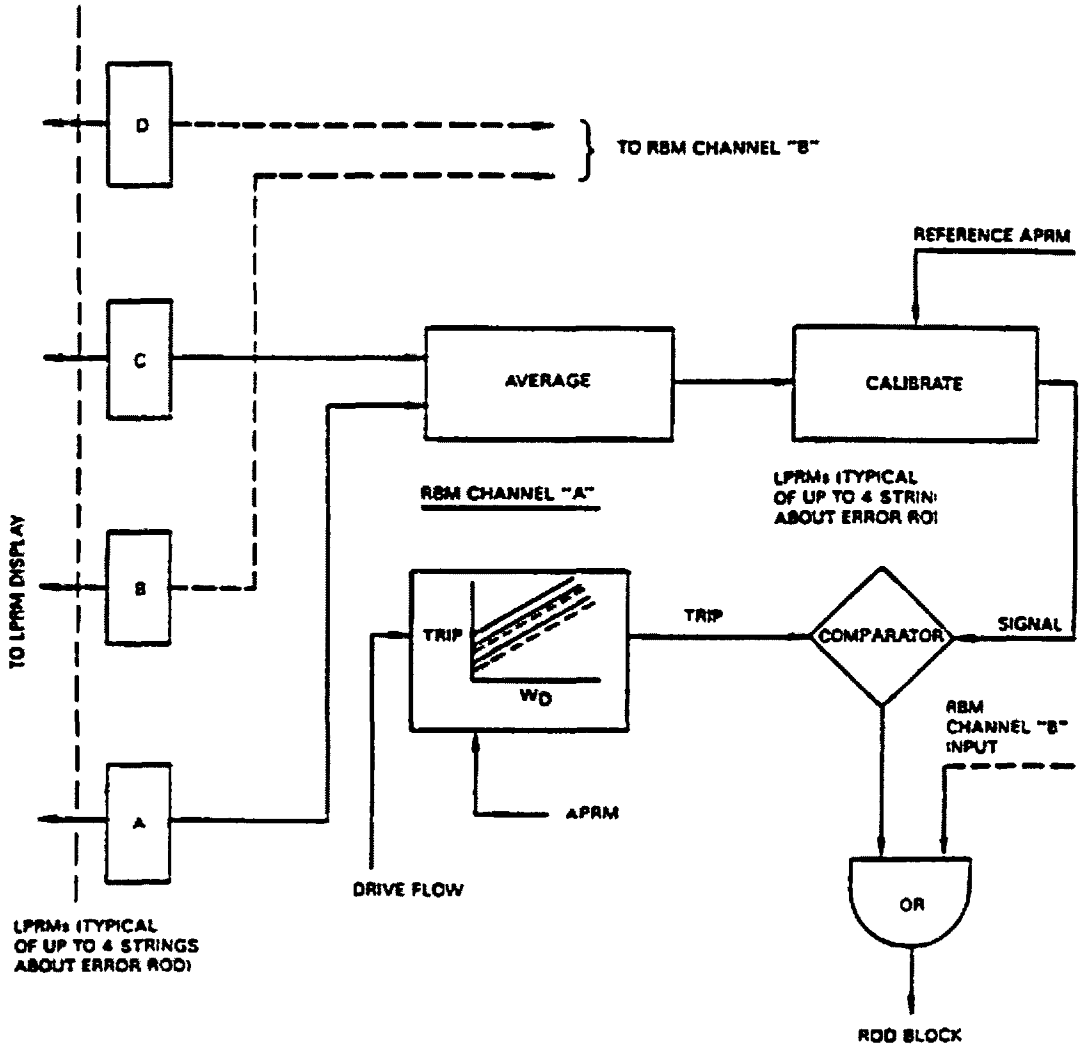

Figure 4-1 Illustration of Current Flow-Dependent RBM with AC/BD LPRM Assignment

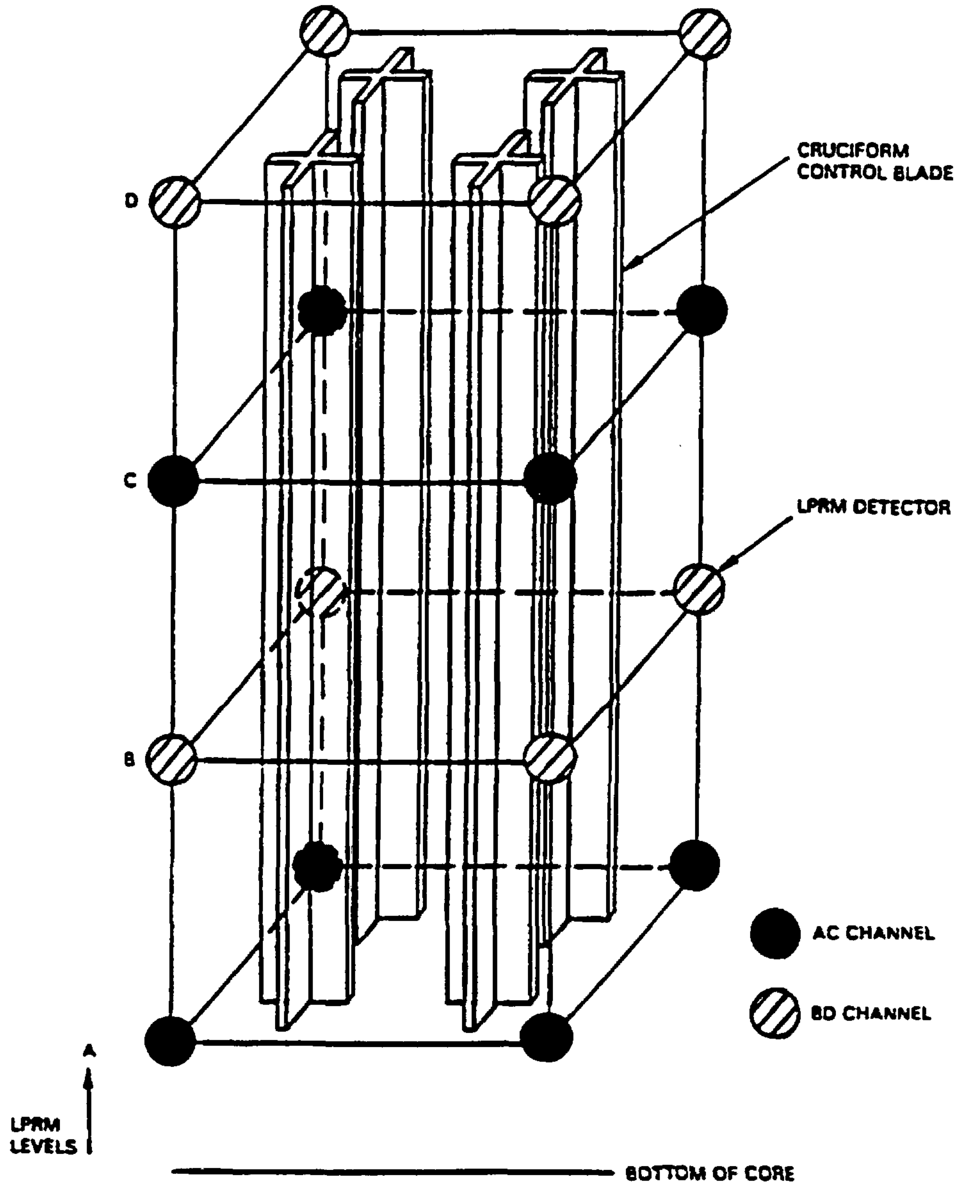

Figure 4-2 RBM Current AC/BD LPRM Assignment

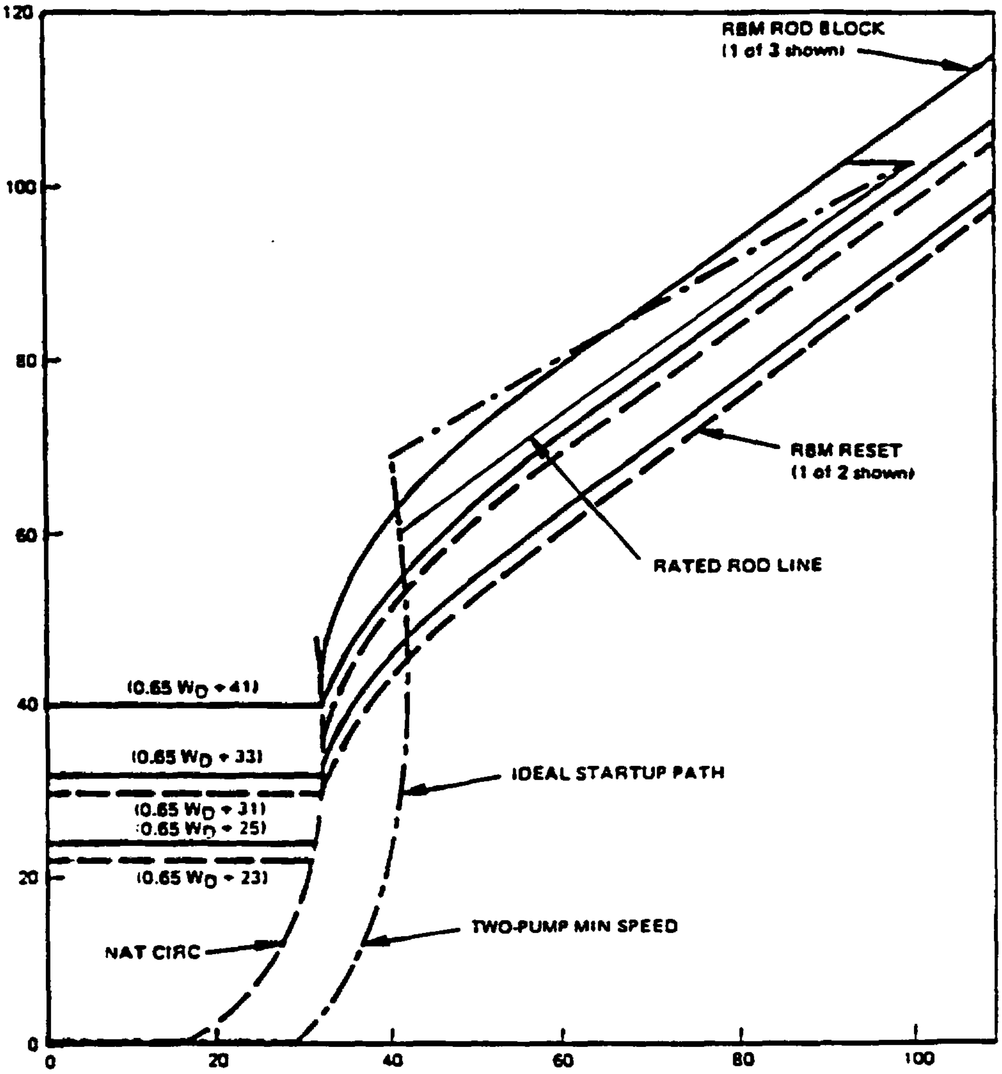

Figure 4-3 Current RBM System Configuration Limits (Typical for 106 Setpoint)

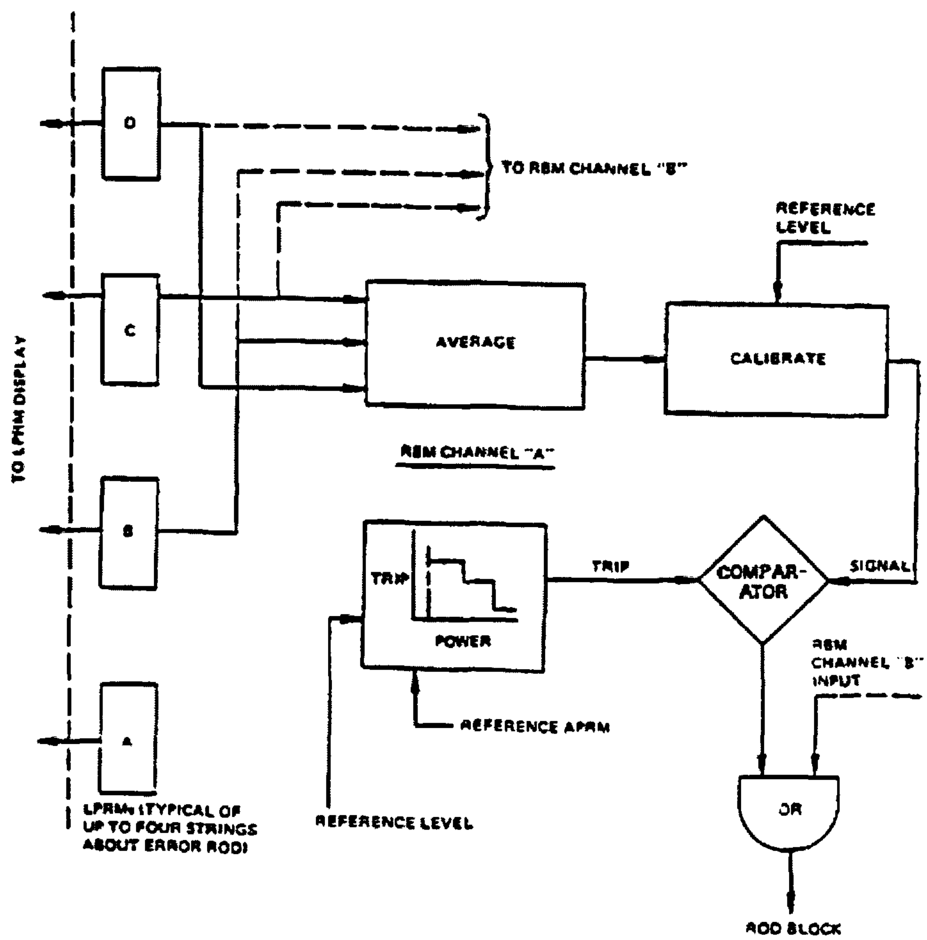

Figure 4-4 Illustration of New Power-Dependent RBM System with $BCCD_1/BCCD_2$ LPRM Assignment

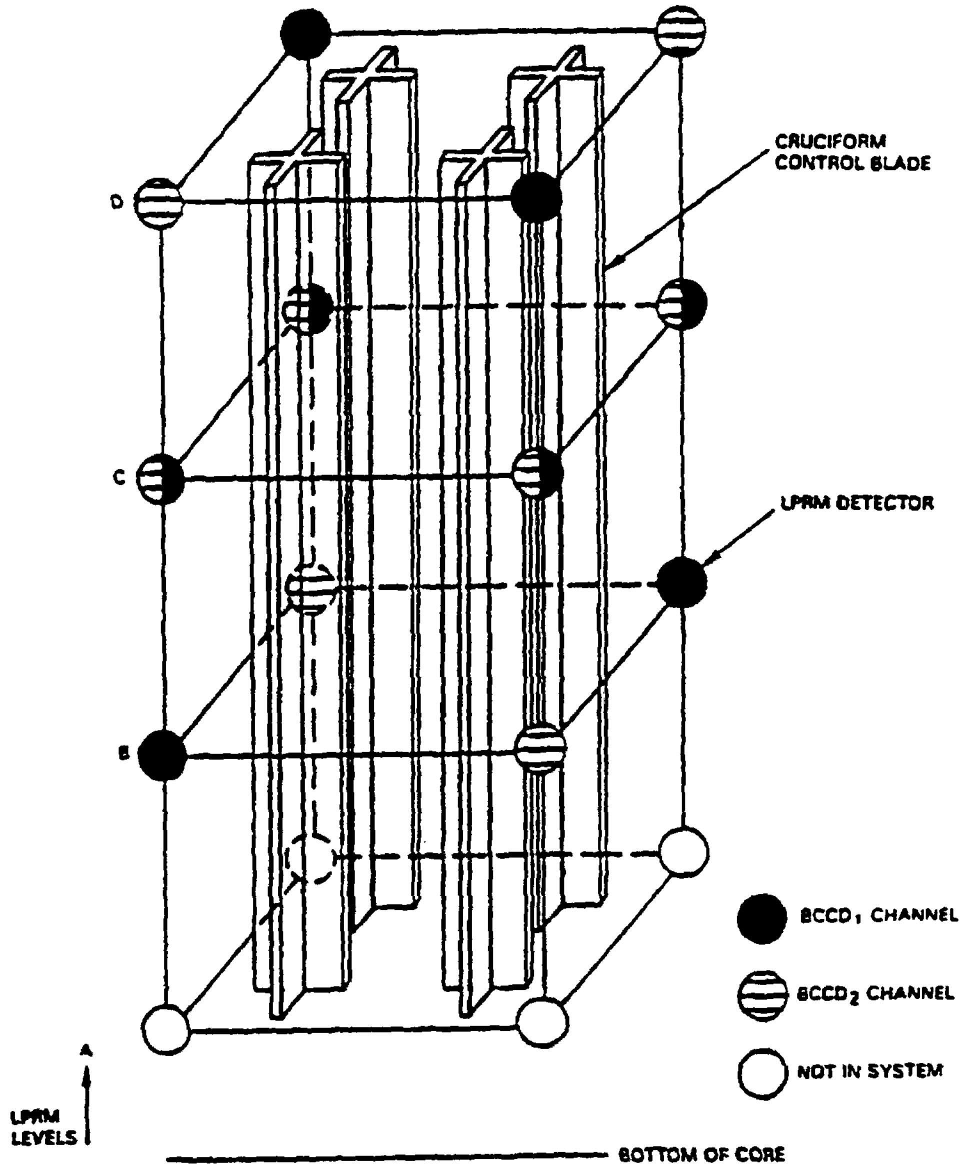

Figure 4-5 New RBM $BCCD_1/BCCD_2$ LPRM Assignment

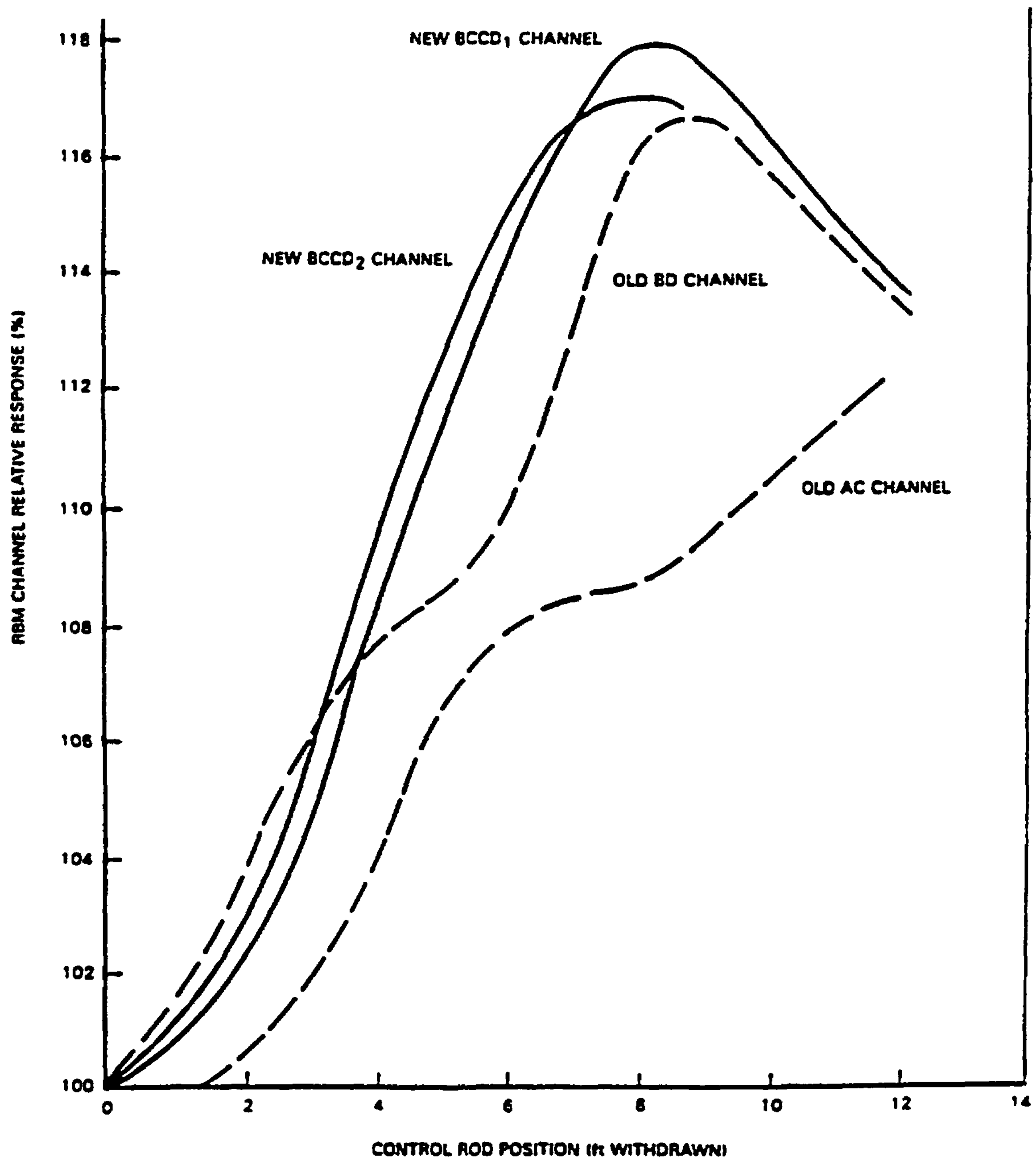

Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs)

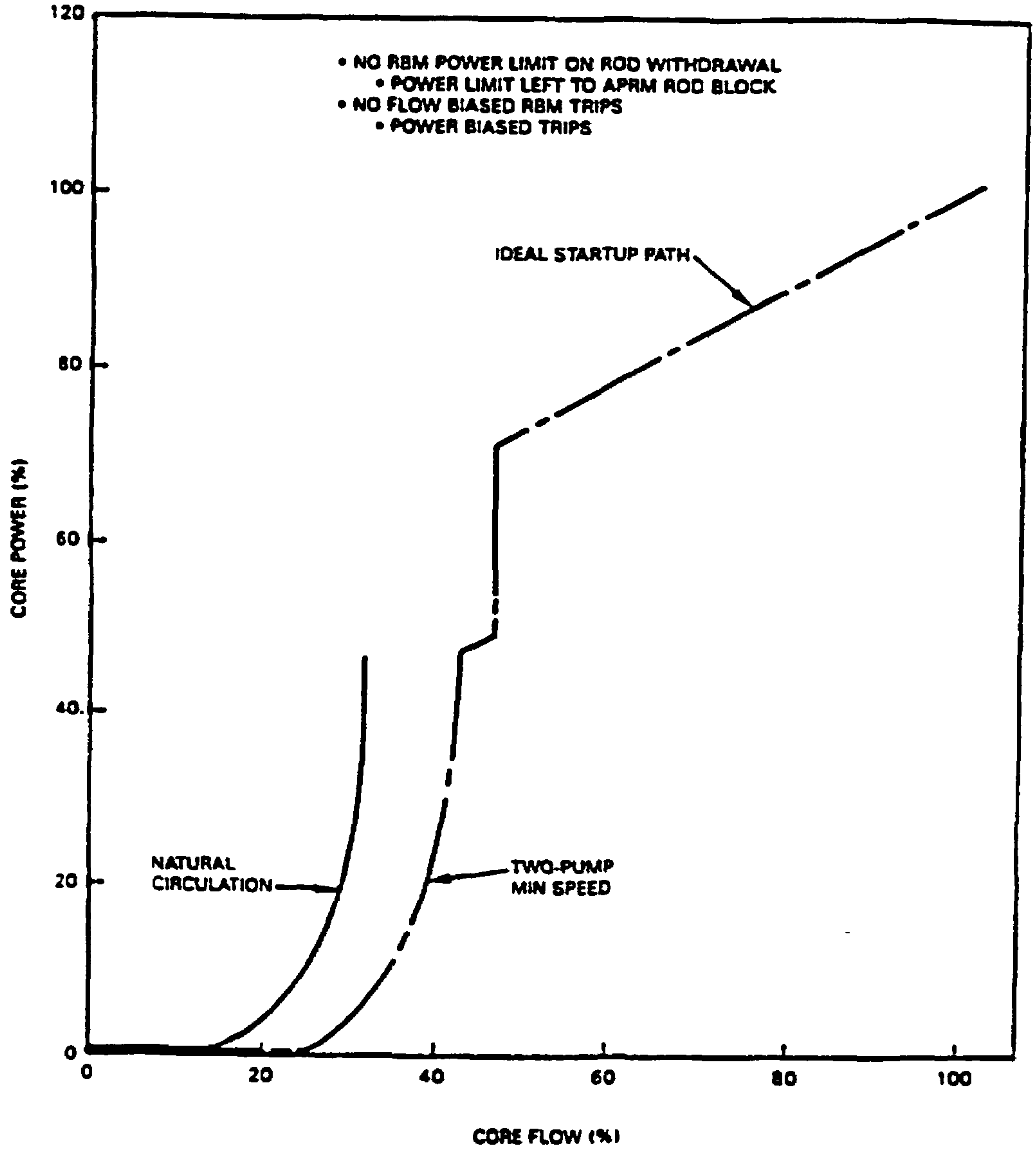

Figure 4-7 New RBM System Core Power Limit (Typical)

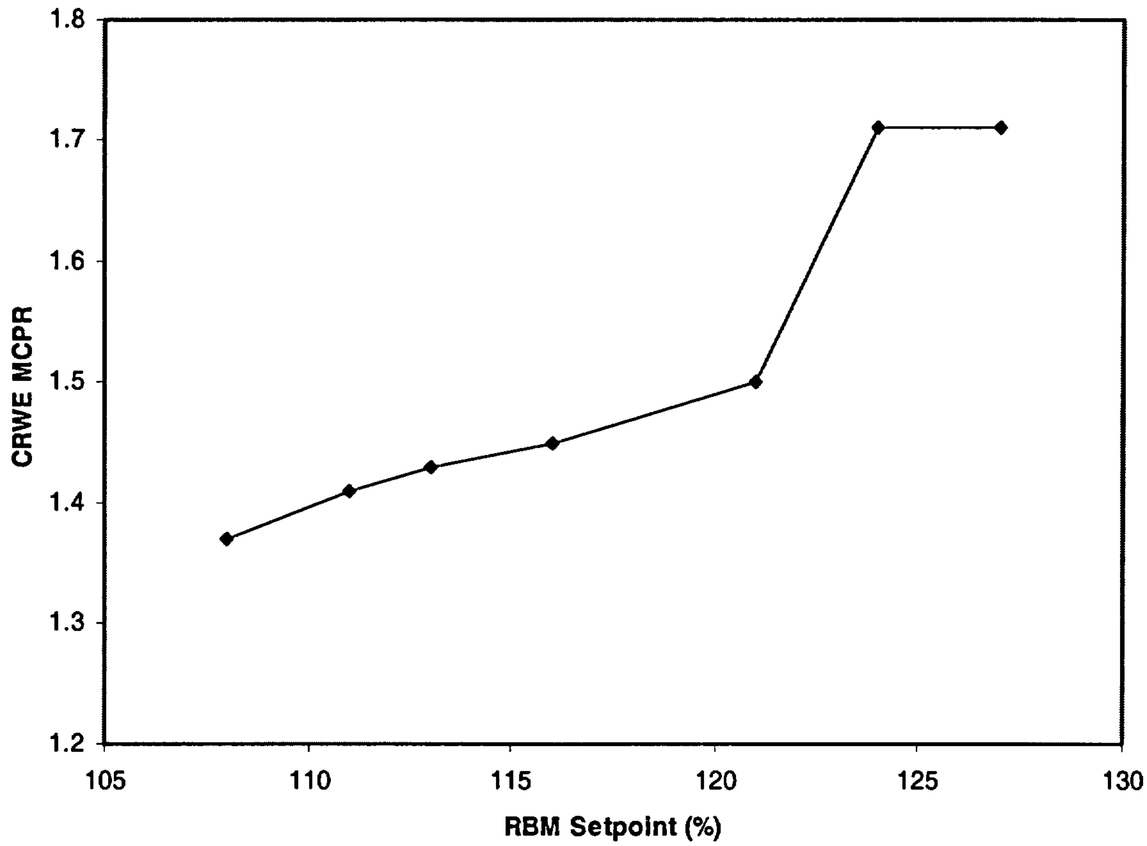

Figure 4-8 Representative CRWE MCPR Requirement Versus RBM Setpoint

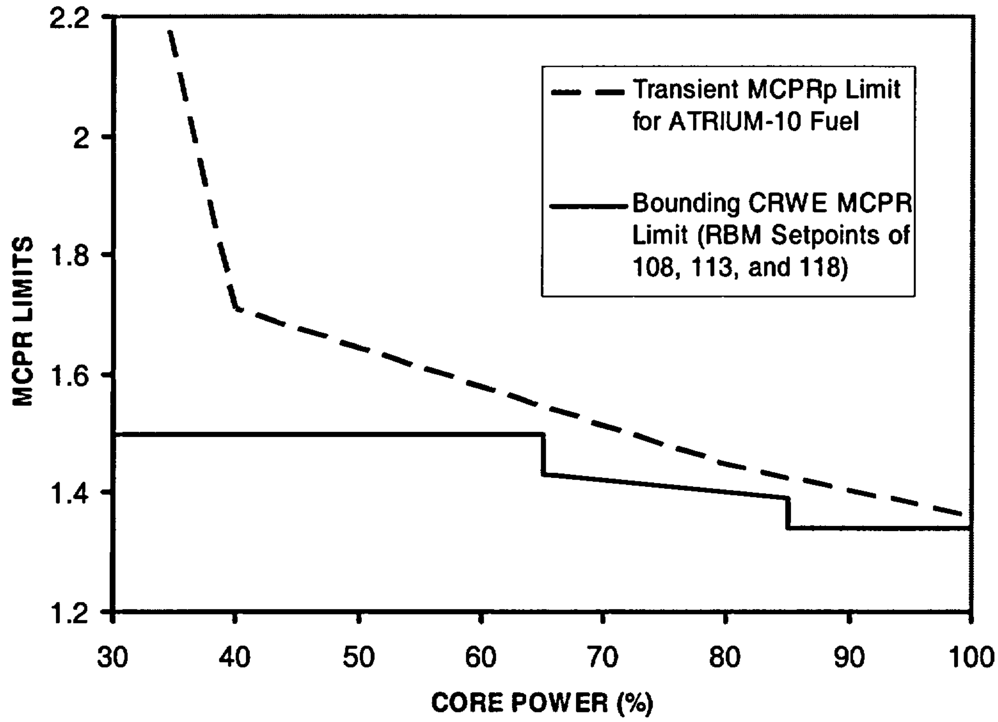

Figure 4-9 Representative CRWE MCPR and $MCPR_p$ Results for ARTS

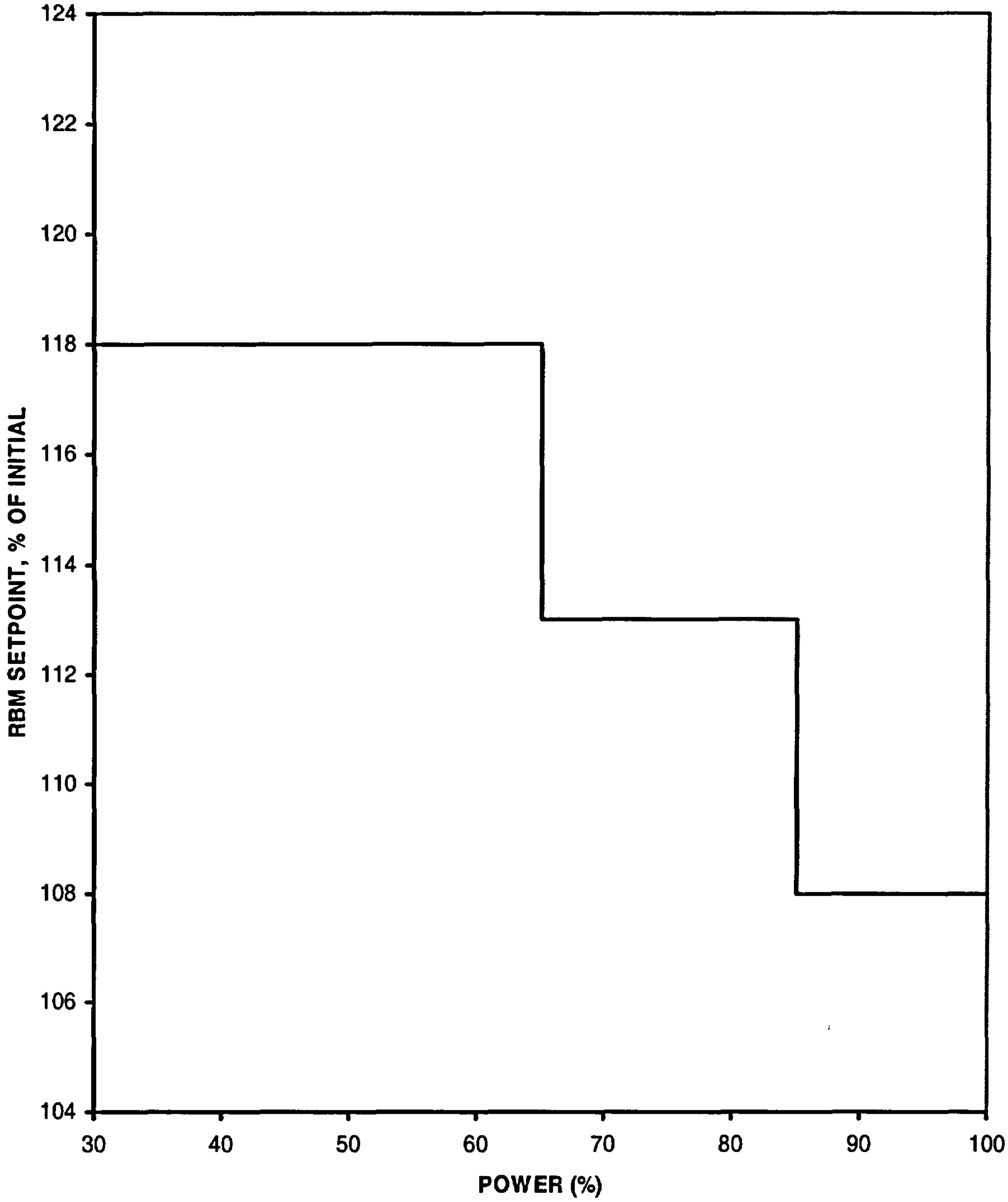

Figure 4-10 Representative RBM Analytical Setpoint Versus Power (without Filter)

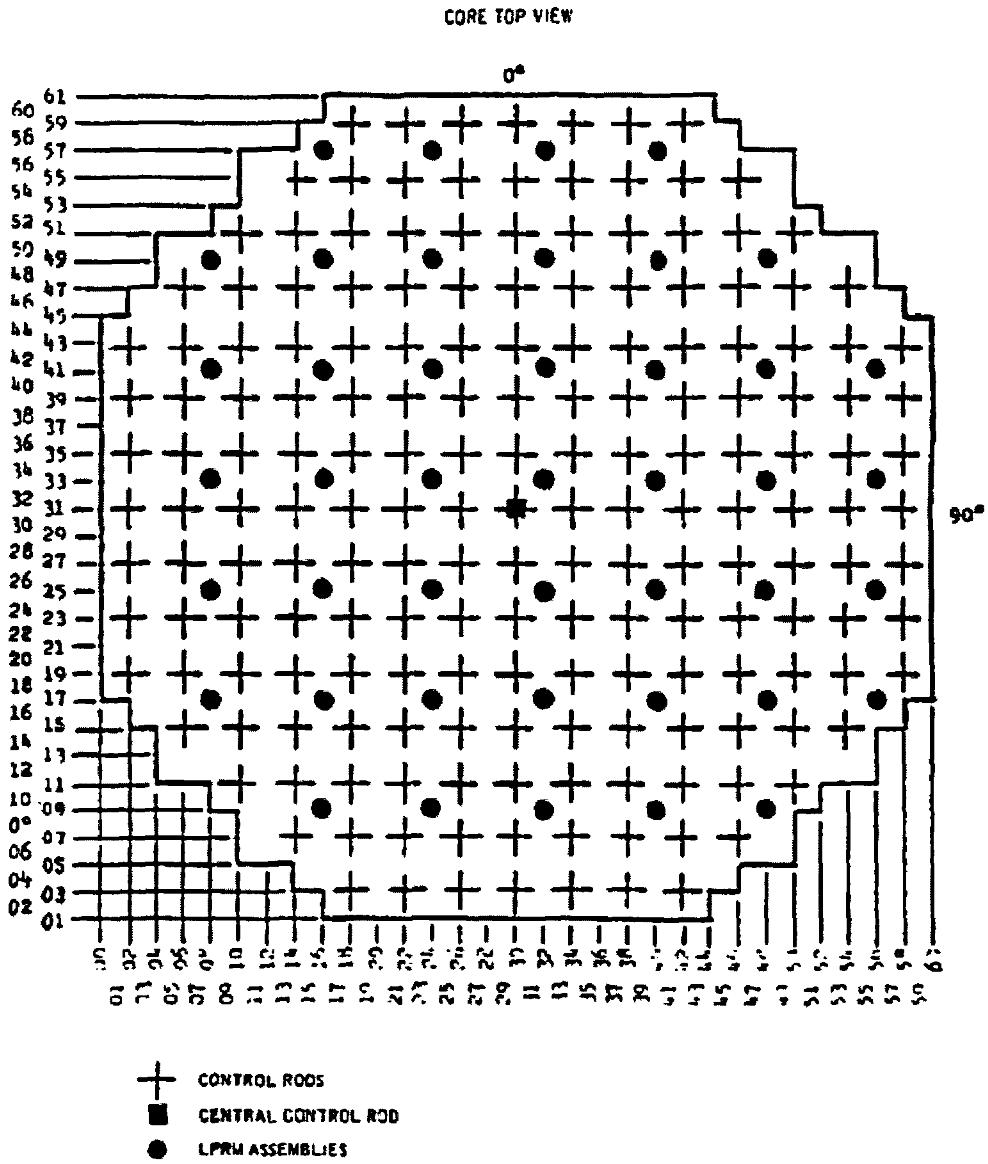

Figure 4-11 SSES Neutron Monitoring System

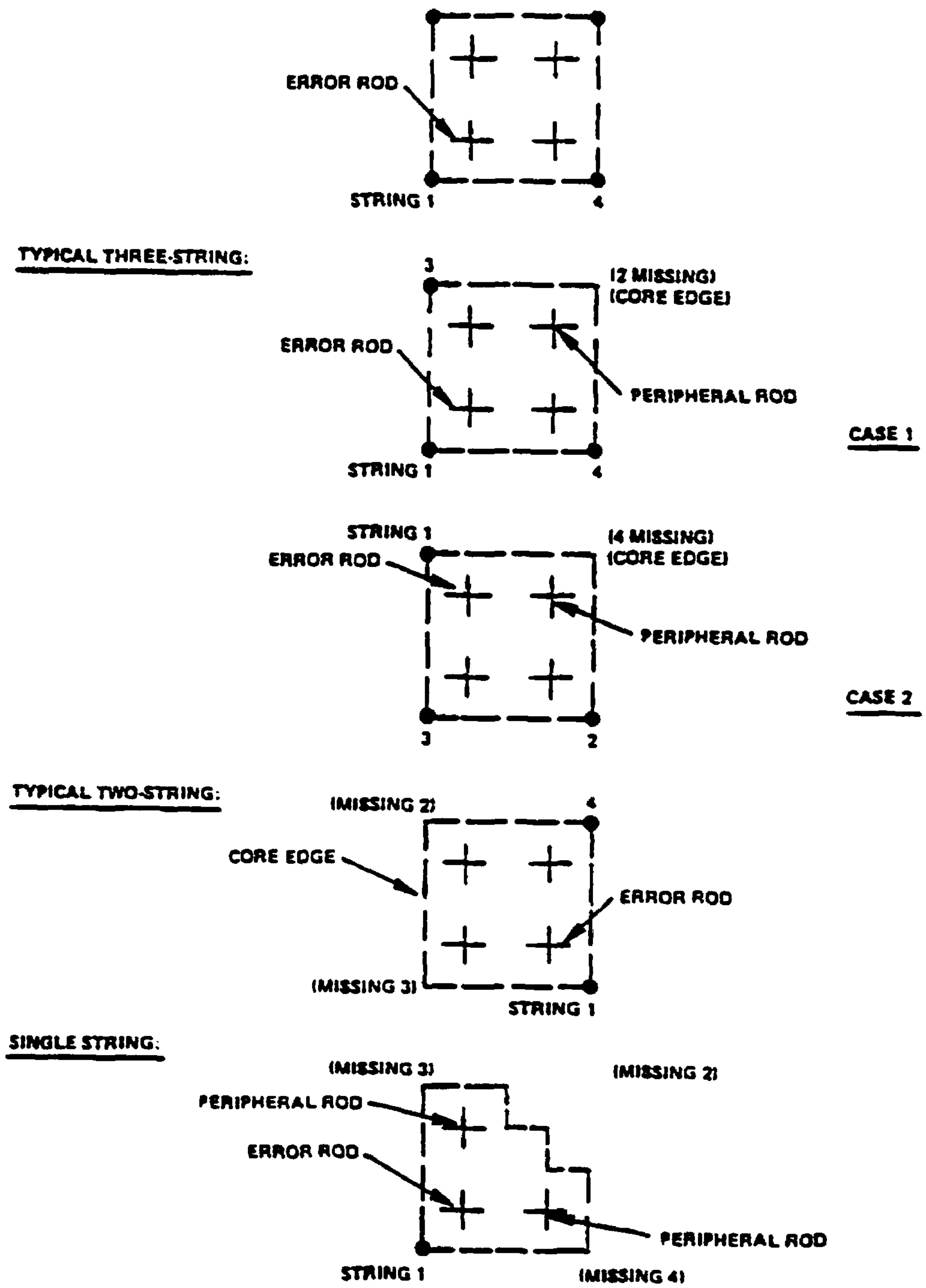

Figure 4-12 Rod Block Monitor Rod Group Geometries

5.0 VESSEL OVERPRESSURE PROTECTION

The MSIV closure with a flux scram (MSIVF) event is used to determine the compliance to the ASME Pressure Vessel Code. This event was previously analyzed at the 100.6%P/108%F state point for the SSES Unit 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). 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 ship overpressure protection analysis. The results from the SSES Unit 2 Cycle 13 overpressure analysis are presented in Table 5-1.

Table 5-1 SSES Unit 2 Cycle 13 Overpressure Analysis Results

Initial Power / Flow (%Rated)	Peak Steam Dome Pressure (psig)	Peak Vessel Pressure (psig)	Peak Steam Line Pressure (psig)
100.6 / 108	1278	1308	1277

6.0 THERMAL-HYDRAULIC STABILITY

6.1 Introduction

SSES is operating with ATRIUM™-10 fuel for Unit 2 Cycle 13. SSES implemented the Option III (Reference 7) stability solution beginning in Cycle 12 for Unit 2. This section presents the effect of the MELLLA operating domain expansion on stability for SSES.

6.2 Option III Evaluation

Option III is a detect and suppress solution which combines closely spaced Local Power Range Monitor (LPRM) detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed 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 (References 7 and 44). This algorithm provides an instrument setpoint designed to trip the reactor before an oscillation can increase to the point where the Safety Limit Minimum Critical Power Ratio (SLMCPR) is exceeded.

The Option III stability reload licensing basis specifies the methodology to calculate the limiting Operating Limit Minimum Critical Power Ratio (OLMCPR) required to protect the SLMCPR for instability events. Selection of an appropriate instrument setpoint is based on the OLMCPR to provide adequate SLMCPR protection.

The PBDA setpoint calculation requires the use of the regional DIVOM (which is defined as the Delta CPR over Initial MCPR Versus the Oscillation Magnitude) curve, determined for SSES on a cycle specific basis.[3] Because the Boiling Water Reactor Owners' Group (BWROG) DIVOM guidelines (Reference 43) specifically provides a suitable DIVOM slope on a cycle specific basis, the Option III solution is fully capable of supporting SSES operation in the MELLLA domain with ATRIUM™-10 fuel.

6.3 Alternate Means to Detect and Suppress

Alternate means to detect and suppress thermal hydraulic instabilities are to be used should the Option III system be declared inoperable. These are manual actions based on the previously employed Interim Corrective Actions (ICAs, Reference 9) and are employed in accordance with the Technical Specifications and the Technical Requirements Manual. The ICAs restrict plant operation in the high power, low core flow region of the BWR power/flow-operating map and contain specific operator actions to respond to a reactor entering the defined restricted regions.

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[3] Evaluations by GE have shown that the generic DIVOM curves specified in NEDO-32465-A (Reference 7), might not be conservative for some plants that have implemented Stability Option III. Specifically, a non-conservative deficiency has been identified for high peak bundle power-to-flow ratios in the generic regional mode DIVOM curve. The deficiency results in a non-conservative slope of the associated DIVOM curve so that the Option III trip setpoint may be too high. GE has made a Part 21 Notification (Reference 8) on this issue.

ICAs provide appropriate guidance to reduce the likelihood of instability and to enhance early detection in the very unlikely event that some stability threshold is exceeded in spite of the ICA guidelines. PPL has committed to review the applicability of the ICA regions on a cycle-specific basis, and take appropriate action to revise the ICA regions if needed. The effectiveness of the ICA regions in preventing reactor instability will be evaluated using the NRC-approved MICROBURN-B2/STAIF methodology (Reference 24) with the NRC approved STAIF stability acceptance criterion for the initial SSES MELLLA cycle and each subsequent reload cycle.

6.4 Conclusion

The SSES MELLLA operating domain expansion complies with the current licensing requirements for stability Option III. The Option III solution is fully capable of supporting SSES 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 ICA regions effectiveness evaluation is fully capable of supporting continued operation, because the evaluation will be performed on a cycle specific basis.

7.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

The current licensing basis LOCA analysis for SSES (Reference 30) supports operation in the MELLLA domain. The initial conditions for the SSES LOCA analysis are listed in Table 7-1.

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 operation in the MELLLA region. The limiting Design Basis Accident (DBA) was evaluated to show that the estimated PCT remains below the acceptance limits. The initial conditions for the SSES LOCA analysis that were used in this determination are listed in Table 7-1. The PCT results are presented in Table 7-2 for ATRIUM™-10 fuel. The maximum local oxidation is less than 2%. The core-wide metal-water reaction is 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 incorporation of ARTS/MELLLA.

Table 7-1 DBA LOCA Initial Conditions for SSES ARTS/MELLLA

Plant Parameter	Maximum Core Flow	Minimum Core Flow
Core Thermal Power (MWt / % CLTP)	3510 / 100.6	3510 / 100.6
Vessel Steam Output (Mlbm/hr)	14.56	14.55
Core Flow (% of RCF)	108	81
Vessel Steam Dome Pressure (psia)	1050.1	1050.0
Maximum RSLB Area (ft^2)	3.503	3.503

Table 7-2 DBA LOCA Results for SSES ARTS/MELLLA with ATRIUM™-10 Fuel[(a)]

Core Flow	PCT (°F)
108%F (Maximum)	1950
81%F (MELLLA)	1945

Notes:

(a) The assumed single failure is failure of the low-pressure coolant injection.

8.0 CONTAINMENT RESPONSE

8.1 Approach/Methodology

This section evaluates the trend effects of ARTS/MELLLA containment pressure and temperature response and the containment LOCA hydrodynamic loads (pool swell, condensation oscillation and chugging) for SSES. The analysis presented here either demonstrates that sufficient conservatism and margin in the containment hydrodynamic loads currently defined for SSES is available to compensate for any variance in these loads due to the extended operating domain, or that the currently defined loads are not affected. The SRV discharge load evaluation would normally consider any increases in the SRV opening setpoints. Because the ARTS/MELLLA operating domain does not require changes to the SRV setpoints, the pressure related SRV loads do not change.

The procedure used for this evaluation follows the methodology used to evaluate the containment LOCA hydrodynamic loads for the SSES (4.5%) and Appendix K and (1.4%) Power Uprates in References 1 through 4.

8.1.1 Short-Term Pressure/Temperature Response

The short-term containment response covers the blowdown period during which the maximum drywell pressure, wetwell pressure, and maximum drywell to wetwell differential pressure occur. A sensitivity study was performed for various cases that cover the full extent of SSES operation in the MELLLA domain. The objective of the study is to demonstrate that SSES operation in the MELLLA domain will not result in exceeding the containment design limits as stated in the SSES FSAR. The results of the study are also used for evaluating the various containment hydrodynamic loads.

The short-term containment pressure and temperature response up to approximately 30 seconds for a DBA LOCA was analyzed for the following four cases:

- Case 1 - 102% of Pre-TPO RTP/100% core flow
- Case 2 - 102% of Pre-TPO RTP/108.00% core flow (ICF)
- Case 3 - 102% of Pre-TPO RTP/81.90% core flow (MELLLA)
- Case 4 - 65.6% of 3510 MWt (102% of Pre-TPO RTP)/41.70% core flow (Minimum Pump Speed)

The Stretch Uprate RTP, or Pre-Thermal Power Optimization (TPO), was 3441 MWt. Therefore, the power level used in the first three analysis cases is 102% of 3441 MWt or 3510 MWt. The Minimum Pump Speed (MPS-MELLLA) point is performed at a power level of 65.60% of 3510 MWt.

These cases were selected to conservatively cover the full extent of the MELLLA power/flow boundary including the Increased Core Flow (ICF) region. The ICF condition is included to provide a consistent set of MELLLA and ICF analyses that can be used to evaluate the effect of the different operating conditions, and to demonstrate that peak pressures occur in the ICF, versus the MELLLA operating condition.

8.1.2 Long-Term Pressure and Temperature

The long-term pressure and temperature response is not affected by MELLLA operation. The long-term containment response is a function of initial reactor power level and the amount of sensible energy stored initially in the reactor vessel fluid and metal components. MELLLA operation does not change the reactor power level (i.e. the rated thermal power) and it does not increase the initial reactor vessel fluid and metal components sensible energy, which would result in a more severe long-term containment response.

8.1.3 LOCA Containment Hydrodynamic Loads

The SSES LOCA containment hydrodynamic loads assessment includes the following:

- Submerged Boundary Load During Vent Clearing
- Pool Swell Loads
- LOCA Steam Condensation Pool Boundary Loads
- Downcomer Lateral Tip Loads
- LOCA Submerged Structure Loads

Plant operation in the ARTS/MELLLA region changes the mass flux and the subcooling of the break flow, which may affect the containment short-term LOCA response and subsequently the containment hydrodynamic loads. These loads were previously defined using the generic methodology from the Mark II Containment Program as described in Reference 10 and accepted by the NRC in References 11 and 12. The plant-specific dynamic loads are also defined in the Design Assessment Report (DAR) for SSES (Reference 13). The current evaluation of these loads to SSES is described in the PPL Loads Reports for the Stretch (4.5%) and Appendix K (1.4%) power uprates.

8.2 Assumptions and Initial Conditions

The following initial containment conditions are used in the ARTS/MELLLA Sensitivity Study.

Parameter	Value
Drywell Pressure (psig)	2
Wetwell Pressure (psig)	2
Drywell Temperature (°F)	120
Suppression Pool Temperature (°F)	90
Drywell humidity (%)	20
Wetwell humidity (%)	100

The following assumptions were used in the study of the short-term containment response for SSES operation in the ARTS/MELLLA domain. In addition, some of the important initial conditions are also listed below.

1. Reactor power generation is assumed to cease concurrently with the time of the accident initiation. There is no delay period.
2. The break being analyzed is an instantaneous double-ended rupture of a recirculation suction line. This results in the maximum discharge rates to the drywell.
3. GE's LAMB computer code (Reference 14) is used to calculate the break flow rates and break enthalpies. These values are then used as inputs to the M3CPT computer code (References 15 and 16) to calculate the containment pressure and temperature response. The drywell pressure response from M3CPT is then used as input to the PICSM computer code (References 11 and 35) to calculate the wetwell pressurization during pool swell.
4. The vessel blowdown flow rates are based on the Moody Slip flow model. (Reference 17)
5. The MSIVs start closing at 0.5 seconds (the delay is associated with the maximum instrument signal response) after the accident. They are fully closed in the shortest possible time of 2.0 seconds following closure initiation.
6. No credit is taken for passive structural heat sinks in the containment. Steam condensation on structures and components in the containment is therefore conservatively neglected.
7. For the DBA-LOCA cases that are analyzed to obtain the maximum pressure and temperature response, the initial drywell and wetwell pressure and drywell relative humidity are selected so as to maximize the initial mass of non-condensable gases. Wetwell humidity is assumed to be 100%.
8. The wetwell airspace is in thermal equilibrium with the suppression pool at all times. This assumption maximizes the wetwell airspace temperature and pressure.
9. The flow of liquid, steam, and air in the vent system is assumed to be a homogenous mixture based on the instantaneous mass fractions in the drywell.
10. The feedwater flow is assumed to begin to coast down at 2.5 seconds and entirely stop at 42.0 seconds. This assumption is consistent with the current analysis of the FSAR.

8.3 Analyses Results

8.3.1 Short-Term Pressure/Temperature Response

The four cases listed above were analyzed as part of the sensitivity study to determine the limiting operating condition. A comparison of the results for the cases in Table 8-1 shows that the limiting operating condition (with respect to peak Drywell pressures, Drywell temperature, and Wetwell pressure) is Case 2 (i.e. 102%P/108%F - ICF). For the Downward Slab Differential Pressure, Case 3 (102%P/81.9%F – MELLLA) was 0.1 psi higher than Case 1 (102%P/100%F – Normal). This 0.1 psi increase in the peak drywell-to-wetwell airspace differential pressure is within the conservativisms of the analytical methods. The NRC review of the topical report (Reference 45) on the critical flow model used states that these analytical methods are conservative.

8.3.2 LOCA Related Hydrodynamic Loads

A description of the LOCA loads and the methodologies for specifying the loads are found in the SESS DAR (Reference 13). The LOCA hydrodynamic load specifications used in the original plant design, augmented by the PPL Loads Reports for the stretch and Appendix K uprates, were reviewed to determine if operation at MELLLA conditions will have an effect on the LOCA loads.

8.3.2.1 Submerged Boundary Loads During Vent Clearing

The vent clearing phenomenon following a LOCA results from the clearing of water from the main vent downcomers due to drywell pressurization. As a result of this phenomenon, pressure loads are produced on the containment basemat and the submerged wetwell walls. The results of the study conclude that the initial drywell pressurization rate for the MELLLA operating domain is bounded by the normal operating conditions. Therefore, the current boundary loads are not affected by MELLLA conditions.

8.3.2.2 Pool Swell Loads

Pool swell loads, which occur during vent clearing from a LOCA, are the result of the rise of the suppression pool surface, which increases the pressure in the wetwell air space. Pool swell loads act on the wetwell boundary and impacts both the drag and fall back loads on the wetwell components located within the pool swell zone. Pool swell loads are a function of the initial drywell pressurization rate during a LOCA. The results of the study conclude that the initial drywell pressurization rate for the MELLLA operating domain is bounded by the normal operating conditions.

8.3.2.3 LOCA Steam Condensation Pool Boundary Loads

After the initial pool-swell transient resulting from a postulated LOCA in a BWR, steam with decreasing amounts of non-condensable gases is vented from the drywell into the wetwell. During such steam venting, condensation-driven oscillations have been observed in related experiments.

Two types of condensation-driven oscillations occur. The first type, Condensation Oscillation (CO), occurs during the earlier portion of the blowdown of a large break LOCA, when the steam mass flux and air content in the steam are high. CO is characterized by sinusoidal pressure oscillations in the drywell and wetwell system. These CO are followed by the second type of condensation-driven oscillations, referred to as chugging. The pressure oscillations during chugging are associated with the rapid collapse of the steam bubble at the vent exit and typically exhibit a pressure spike, followed by a damped ringout which has predominant frequency components at the vent and pool natural frequency (Reference 12).

The load definition for SSES is based on the full scale LOCA steam condensation tests conducted by Kraftwerk Union (KWU) at their GKM II-M test facility. The SSES Design Assessment Report (DAR) (Reference 13) provides a description of the test matrix and results.

8.3.2.3.1 Condensation Oscillation (CO) Submerged Boundary Loads

CO loads result from oscillation of the steam-water interface that forms at the vent exit during a LOCA during the period of high vent water vapor mass flow rate. This occurs after pool swell. The CO loads include loads on submerged boundaries and submerged structures. Generally, the CO load increases with higher suppression pool temperature and/or higher vent mass flow rate. A comparison of the break flow (and hence vent flow) for the MELLLA conditions with the flow used to document the adequacy of the CO loads for stretch uprate, indicates very similar flow, and both are enveloped by the vent flow calculated for the GKM II-M test. Hence, the CO load developed for the GKM-II-M test data remains bounding for SSES at MELLLA conditions.

8.3.2.3.2 Chugging Submerged Boundary Loads

Chugging occurs when the steam (water vapor) mass flux through the vents during a LOCA is not high enough to maintain a steady steam/water interface at the vent exit. Chugging loads result from a collapse of steam (water vapor) bubbles that form at the vent exit, and they include loads on the suppression pool boundary and submerged structures.

The chugging loads are dominated by the Main Steam Line Break (MSLB) and smaller steam breaks. Since the vessel pressure is unchanged for MELLLA at full power, the MSLB break flow and consequently the steam vent flow will be the same for operation in the MELLLA domain. Since the chug amplitude is proportional to the vent steam mass flux, which is not changed for MELLLA, the design pressure traces selected from the smaller steam break tests are conservative for the MELLLA domain. Therefore, the chugging load developed for the GKM-II-M test data remains bounding for SSES at MELLLA conditions.

8.3.2.4 Downcomer Lateral Tip Loads

Chugging produces a lateral load at the downcomer exit. The chugging loads are dominated by the MSLB and smaller steam breaks. As concluded in Section 8.3.2.3.2, the current chugging loads bound the MELLLA operating conditions; therefore, the current downcomer lateral tip loads remain bounding for SSES at MELLLA conditions.

8.3.2.5 LOCA Submerged Structure Loads

8.3.2.5.1 Downcomer Jet Loads

The clearing of the downcomers following the design basis LOCA produces a water jet load on submerged structures located beneath the downcomers. Reference 11 describes this phenomenon as a water discharge in the form of a narrow jet whose transverse dimension remains approximately the size of the exit diameter. A review of the original downcomer jet load evaluation found that no submerged structures were located underneath the downcomers

8.3.2.5.2 LOCA Air Bubble Loads

The analysis for the stretch uprate indicates a decrease in air bubble pressure of 14% from the original value used in the SSES DAR (Reference 13). This is due to the fact that the DAR was not based on NUREG 0808 criteria (Reference 12). When the same criteria were applied to the

stretch uprate and the original design basis conditions, the same wetwell pressure was calculated. Since the MELLLA analysis wetwell pressure is lower than the wetwell pressure evaluated for the stretch uprate, the original DAR analysis is valid and conservative for MELLLA conditions.

8.3.2.5.3 CO and Chugging Submerged Structure Drag Load

The original CO and chugging submerged structure drag load evaluation used the same acoustic model of the SSES suppression pool and design pressure traces for sourcing as the CO and chugging submerged boundary load methodology (Reference 12). However, the pressures are calculated at the submerged structure surfaces, instead of containment boundary.

As described in sections 8.3.2.3.1 and 8.3.2.3.2 of this report, the current CO and chugging loads are conservative for MELLLA. Thus, the pressures calculated at the submerged structure surfaces are also conservative for MELLLA.

8.4 Conclusion

The above analysis and evaluation demonstrates that the existing LOCA containment hydrodynamic load definition bounds the loads calculated for SSES operation in the MELLLA domain. The containment parameter peak values obtained from the MELLLA analyses are within the design limits. Plant operation in the ARTS/MELLLA region has no significant effect on the peak drywell and wetwell pressures, or on the peak differential pressure between the drywell and the wetwell.

8.5 Reactor Asymmetric Loads

The reactor asymmetric loads during the Design Basis Accident (DBA) Loss-Of-Coolant Accident (LOCA) include the annulus pressurization (AP) loads, the jet reaction loads/jet impingement loads, and the pipe whip loads. This section describes the effects of MELLLA on these loads.

The following line breaks in the annulus region (reactor pressure vessel (RPV) to shield wall) were evaluated for the effects of MELLLA:

- Recirculation Suction Line Break (RSLB)
- Feedwater Line Break (FWLB)
- Main Steam Line Break (MSLB)

The methodology for calculating the current RSLB blowdown mass and energy release profile for AP loads is the conservative methodology documented in Reference 36. Using this methodology, the mass and energy release profile calculated for the MELLLA conditions exceeds the SSES design calculation. Specifically, blowdown mass release at the MELLLA minimum pump speed condition exceeds the SSES design calculation. Therefore, a more realistic blowdown mass and energy release profile for this condition was determined using the GE code LAMB for the AP load analysis. In addition, credit was taken for the lower operating steam dome pressure at the lower power level. The LAMB code considers the pipe break separation time history, but ignores the fluid inertia effect, providing conservative results.

LAMB has been used in several plant licensing applications to calculate the blowdown mass flow rate and energy profile in the event of an RSLB and has been accepted for licensing applications for power/flow map extension (MELLLA) associated with BWR extended power uprates (Reference 37, Section G.1.1). The LAMB methodology has been used to calculate the mass and energy releases for short-term containment response analysis for several applications. LAMB results at the MELLLA minimum pump speed condition are bounded by the SSES design calculation.

The AP loads, the jet reaction loads/jet impingement loads, and the pipe whip loads would occur during the time periods following the double ended guillotine break of the recirculation suction line, and are combined for the evaluation of the structural integrity of the RPV, reactor internals, the biological shield wall, control rod drive (CRD) mechanism, and the piping systems that are connected to the RPV and penetrate the biological shield wall. Since the mass and energy releases at the off-rated conditions associated with ARTS/MELLLA have been shown to be bounded by the current analysis, these analyses were not performed.

The AP mass and energy release analysis was performed over the range of power/flow conditions associated with the MELLLA boundary. When the same basis is used as in the calculation of record, the results were determined to be bounded by the current basis.

For the FWLB, the inlet conditions of pressure and enthalpy for feedwater remain unchanged. Therefore, the current calculation of record is still applicable.

For the MSLB, the current calculation of record is still applicable as the dome pressure remains unchanged.

Table 8-1 Summary of Sensitivity Analysis Results

Case	Drywell Pressure	Drywell Temperature	Wetwell Pressure	Slab Differential Pressure Downward
	(psig)	(°F)	(psig)	(psid)
1) 102%P/100%F (Normal)	47.8	294.0	36.7	25.8
2) 102%P/108%F (ICF)	**47.9**	**294.1**	**36.7**	25.7
3) 102%P/81.9%F (MELLLA)	47.2	293.1	36.5	**25.9**
4) 67%P/42%F (MELLLA-MPS)	43.6	289.1	31.4	20.7

Note: "P" in the table refers to the Pre-TPO RTP, while the "F" refers to rated core flow.

9.0 REACTOR INTERNALS INTEGRITY

9.1 Reactor Internal Pressure Differences (RIPDs)

The RIPDs across the reactor internal components and the fuel channels in the MELLLA condition are bounded by the uprated ELLLA (87% of Rated Core Flow (RCF)) and the Increase Core Flow (ICF) (108% of RCF) conditions due to the lower core flow condition in MELLLA (81.9% of RCF). Thus, no new RIPDs, fuel bundle lift and Control Rod Guide Tube (CRGT) conditions are generated by the MELLLA operating domain. The current RIPD basis in Reference 1 remains applicable to the MELLLA condition.

9.2 Acoustic and Flow-Induced Loads

The acoustic and flow-induced loads are contributing factors to the SSES design basis load combination in the Faulted condition. The acoustic loads are imposed on the reactor internal structures as a result of the propagation of the decompression wave created by the assumption of an instantaneous Recirculation Line Suction Break (RSLB). The acoustic loads affect the core shroud, core shroud repair components, core shroud support, and jet pumps. The flow-induced loads are imposed on the reactor internal structures as a result of the fluid velocities from the discharged coolant during an RSLB. The flow-induced loads affect the core shroud and jet pumps.

9.2.1 Approach/Methodology

Major components in the vessel annulus region, the shroud, shroud support, and jet pumps were evaluated for the bounding RSLB acoustic and flow-induced loads representing the MELLLA conditions.

The flow-induced loads were calculated for an RSLB utilizing the specific SSES geometry and fluid conditions applied to a reference BWR calculation. The loads were calculated by applying scaling factors that account for plant-specific geometry differences (e.g., size of the shroud, reactor vessel, and recirculation line) and thermal-hydraulic condition differences (e.g., downcomer subcooling) from the reference plant. The reference calculation was based on the GE methods utilized to support NRC Generic Letter 94-03 that was issued to address the shroud cracks detected at some BWRs.

The acoustic loads on the jet pumps and shroud applied for SSES represent SSES-specific plant geometry configuration and operating conditions. The bounding natural frequencies for the jet pumps and shroud along with the bounding subcooling are applied. For acoustic loads on the shroud support, generic bounding BWR loads based on the GE approved methods were used for the flow-induced load calculation.

For SSES, the most limiting subcooling condition is at the intersection of the minimum pump speed and the MELLLA or uprated ELLLA maximum power boundary line. The initial thermal hydraulic conditions including the subcooling at this point are applied to the reference BWR calculation, along with the SSES geometry, to determine the plant specific flow-induced loads.

9.2.2 Input Assumptions

The following assumptions and initial conditions were used in the determination of the acoustic and flow-induced loads for the MELLLA operation.

Initial Conditions	Bases/Justifications
100.6%P / 100%F	Consistent with the SSES current licensing basis.
100.6%P / 87%F	ELLLA corner at rated power at nominal rated feedwater temperature.
100.6%P / 81.9%F	MELLLA corner at rated power at nominal rated feedwater temperature.
70.4%P / 41%F	Minimum pump speed point on the uprated ELLLA boundary line at nominal rated feedwater temperature.
65.6%P / 41.7%F	Bounding power/flow state point for MELLLA; minimum pump speed point on the MELLLA boundary line at nominal rated feedwater temperature

9.2.3 Results

The flow-induced loads for the shroud and jet pumps are shown in Table 9-1. SSES-specific flow-induced load multipliers for off-rated conditions to be applied to the baseline loads are also documented. The maximum acoustic loads on the shroud and jet pumps are shown in Table 9-2. The maximum acoustic loads on the shroud support are shown in Table 9-3. These loads were used to determine the structural integrity of these components.

The acoustic and flow-induced loads in the MELLLA condition (at the RTP and 81.9% RCF) are slightly higher than the current uprated ELLLA condition (at the RTP and 87% RCF) due to the increased subcooling in the downcomer associated with the MELLLA condition. From ELLLA to MELLLA, the downcomer subcooling increases thereby increasing the critical flow and the mass flux out of the break in a postulated RSLB. As a result, the acoustic and flow-induced loads in MELLLA conditions increase slightly.

However, the loads at the minimum pump speed in the current plant ELLLA map are higher than the loads in the MELLLA map because the maximum rod line in the uprated ELLLA domain is extrapolated to low flow, although the APRM Rod Block boundary is lower. This extrapolated line crosses the MELLLA boundary line between the minimum pump speed and the minimum MELLLA core flow at 100% RTP, which results in greater subcooling for the recirculation minimum pump speed in the uprated ELLLA domain.

9.3 Structural Integrity Evaluation

The structural integrity of the reactor internals was evaluated for the loads associated with MELLLA operation for the SSES considering the current design basis evaluations. The loads considered for MELLLA include Dead Weights, Seismic Loads, RIPDs, Acoustic and Flow induced Loads due to Loss of Coolant Accidents, Safety Relief Valve (SRV), Annulus Pressurization (AP) loads, Jet Reaction loads, Thermal loads, Flow Loads and Fuel Lift loads. The limiting flow conditions and thermal conditions were considered. The RPV internals are not certified to the ASME Code; however, the requirements of the ASME Code are used as guidelines in their design basis analysis. The following RPV internal components were evaluated:

- Core Plate
- Top Guide
- Control Rod Drive Housing
- Control Rod Guide Tube
- Orificed Fuel Support
- Jet Pumps
- Core Spray Line and Sparger
- Access Hole Cover
- Shroud Head and Steam Separator Assembly
- Shroud
- Shroud Support

Acoustic loads for MELLLA condition have increased for shroud and shroud support with respect to CLTP. The existing shroud load definitions were reviewed, and the current defect evaluations of record remain valid. All other loads for MELLLA conditions remain unchanged with respect to CLTP, or remain bounded by those of CLTP. The changes in the flow and thermal loads for MELLLA with respect to CLTP are negligible (<4%). All stresses and fatigue usage factors for CLTP were determined to remain applicable in the MELLLA conditions. The existing flaw evaluations for the Feedwater Sparger, and the Flaw Handbook for the Core Spray Line, were reviewed for the ARTS/MELLLA loads, and were found to remain valid for the ARTS/MELLLA conditions. All the RPV internals remain structurally qualified for operation in the MELLLA condition.

9.4 Reactor Internals Vibration

9.4.1 Approach/ Methodology

To ensure that the flow-induced vibration (FIV) response of the reactor internals is acceptable, a single reactor for each product line and size undergoes an extensively instrumented vibration test during initial plant startup. After analyzing the results of such a test and assuring that all responses fall within acceptable limits of the established criteria, the tested reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20. Other reactors of the same product line and size are classified as non-prototype and undergo a less rigorous confirmatory test.

Browns Ferry Unit 1 (BF-1) was designated as the prototype plant for BWR4, 251-inch diameter reactors in accordance with Regulatory Guide 1.20. FIV testing was performed at BF-1 and data was collected during plant start-up between December 1972 and March 1974. The critical reactor internals were instrumented with vibration sensors and the reactor was tested up to 100% core flow at 100% rod line and at increased core flow up to 113% at 50% rod line. The RCF for BF-1 is 2.5% higher than SSES. This data was used in the SSES evaluation.

SSES is currently licensed to operate at an ICF of up to 108% of RCF (108 Mlbs/hr) at 100% of RTP. For MELLLA operation, the rated power output remains the same, but core flow is reduced to 81.9% of RCF at 100% of RTP as shown in Figure 1-1.

9.4.2 Inputs/Assumptions

The following inputs/assumption were used in the reactor internals vibration evaluation:

Parameter	Input
Plant data selected for flow induced vibration evaluation	BF-1 was designated as the prototype plant for BWR4, 251-inch diameter reactors in accordance with Regulatory Guide 1.20. Therefore, BF-1 FIV data collected during plant start-up between December1972 and March 1974 was used. The reactor was tested up to 100% core flow at 100% rod line, and at increased core flow up to 113% of RCF at 50% rod line.
Target plant conditions in the MELLLA region selected for component evaluation	RTP of 3489 MWt and 81.9% of RCF at 100% of RTP (114% rod line) with balanced flow conditions.
GE stress acceptance criterion of 10,000 psi is used for all stainless steel components	Limit is lower than the more conservative value allowed by the current ASME Section III design codes for the same material and is bounding for all stainless steel material. The ASME Section III value is 13,600 psi for service cycles equal to 10^{11}.

9.4.3 Analyses Results

Because the vibration levels generally increase as the square of the flow and MELLLA flow rates are lower than RTP flow rates with power remaining unchanged, RTP vibration levels bound those at MELLLA conditions.

The reactor internals vibration measurements report for the prototype plant (BF-1) was reviewed to determine the components likely to have significant vibration at the MELLLA conditions. Only the jet pump sensing lines are affected by MELLLA. Based on the analysis, the recirculation pump vane passing frequency (VPF) will not have an adverse effect on the jet pump sensing lines during MELLLA operating conditions.

For the shroud, shroud head, separators, and the steam dryer, the vibrations are a function of the steam flow, which at MELLLA conditions is bounded by the steam flow at RTP. The feedwater sparger vibrations are a function of the feedwater flow and are bounded at MELLLA conditions, by the feedwater flow at RTP.

The lower plenum components (Control Rod Guide Tube (CRGT), Incore Guide Tube (ICGT)) and the jet pumps are dependent on the core flow, because the vibration levels are generally proportional to the square of the flow. Therefore, these components experience reduced vibration due to the reduction in core flow during MELLLA operation. Therefore, the vibration levels of those components at MELLLA conditions are bounded by those at RTP conditions.

The jet pump riser braces were evaluated for possible resonance due to VPF pressure pulsations. The jet pump riser braces natural frequencies are well separated from the recirculation pump VPF during MELLLA conditions and will not have any increased vibrations.

The FIV evaluation is conservative for the following reasons:

- The GE criteria of 10,000 psi peak stress intensity is more conservative than the ASME allowable peak stress intensity of 13,600 psi for service cycles equal to 10^{11};
- The modes are absolute summed; and
- The maximum vibration amplitude in each mode is used in the absolute sum process, whereas in reality the vibration amplitude fluctuates.

Therefore, the FIV will remain within acceptable limits.

9.5 Conclusion

The analyses documented in this section demonstrate that, from an FIV viewpoint, the reactor internals structural mechanical integrity is maintained to provide SSES safe operation in the MELLLA domain.

Table 9-1 Flow-induced Loads on Shroud and Jet Pumps for SSES

Component	Parameter	Loads (1)
Shroud	Baseline Force (kips)	213.954
	Baseline Moment at the Shroud Centerline (10^6 in-lbf)	16.727
Jet Pump	Baseline Force (kips)	14.215
	Baseline Moment at the Jet Pump Centerline (10^6 in-lbf)	0.711
Component	**Operating Condition**	**Load Multiplier**
Jet Pump /Shroud	100.6%P / 100%F	1.0000
	100.6%P / 87%F (ELLLA)	1.0690
	100.6%P / 81.9%F (MELLLA)	1.1034
	70.4%P / 41 %F (extrapolated ELLLA)	1.4718
	65.6%P / 41.7%F (MELLLA)	1.4113

Note (1): Loads at rated conditions (100.6% power/100% core flow).

Table 9-2 Maximum Acoustic Loads on Shroud and Jet Pumps

Component	Conditions	Force (kips)	Effective Force (kips)	Moment (10^6 in-lbf)	Effective Moment (10^6 in-lbf)
Shroud	70.4%P / 41 %F (extrapolated ELLLA)	2712.727	1342.083	362.500	145.614
	65.6%P / 41.7%F (MELLLA)	2712.728	1342.087	362.501	145.615
Jet Pump	70.4%P / 41 %F (extrapolated ELLLA)	35.534	30.770	1.927	1.607
	65.6%P / 41.7%F (MELLLA)	35.534	30.770	1.927	1.607

Table 9-3 Maximum Acoustic Loads on Shroud Support

Component	Parameter	Unit	Loads
Shroud Support	Total Vertical Force	Kips	2200
	Moment at the Shroud Support Plate Outside Edge Nearest the Break	10^6 in-lbf	324
	Half Period	Sec	0.037

10.0 ANTICIPATED TRANSIENT WITHOUT SCRAM

10.1 Approach/Methodology

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 the approved licensing methodology (Reference 18), to demonstrate that ATWS acceptance criteria are met for operation in the MELLLA region.

The ATWS analysis takes credit for ATWS-RPT and SLCS, but assumes that ARI fails. If reactor vessel and fuel integrity are maintained, then the ATWS-RPT setpoint is adequate. If containment integrity is maintained, then the SLCS injection rate is adequate.

MELLLA conditions provide the greatest effect on peak vessel pressure and peak long-term containment response (suppression pool temperature and containment pressure). The analysis is based on an initial power level of 100% of RTP and the corresponding MELLLA minimum core flow of 81.9% of RCF.

Three ATWS events were re-evaluated at the most limiting MELLLA point (100% of RTP and 81.9% of RCF) with ARI assumed to fail, thus requiring the operator to initiate SLCS injection for shutdown. These events were: (1) closure of all MSIVs (MSIVC), (2) Pressure Regulator Failure (Open) to maximum steam demand flow (PRFO), and (3) Loss Of Offsite Power (LOOP).

The Inadvertent Opening of a Relief Valve (IORV) event was also considered, but found to be non-limiting. For the IORV event, the availability of the main condenser reduces the severity of the peak suppression pool temperature increase. The absence of reactor vessel isolation avoids the vessel pressurization; therefore, the peak vessel pressure remains at or below the initial value. A reactor power excursion does not occur and the fuel does not experience boiling transition. Hence, the Peak Cladding Temperature (PCT) is bounded by MSIVC and PRFO events.

The LOOP event is less limiting than the MSIVC in the short term due to the initiation of recirculation and condensate pump coastdown at time zero, which effectively reduces the severity of the initial power surge. The generic ATWS analysis in Reference 40 has shown that this event does not produce limiting results for pool temperature, containment pressure, peak reactor pressure, or PCT.

The following ATWS acceptance criteria were used to determine acceptability of the SSES operation in the MELLLA region (based on the results of the MSIVC and PRFO transients – the LOOP is non-limiting for these evaluations):

1. Fuel integrity:
    - Maximum clad temperature < 2200° F
    - Maximum local clad oxidation < 17%
2. RPV integrity:
    - Peak RPV pressure < 1500 psig (ASME service level C)
3. Containment integrity:
    - Peak suppression pool bulk temperature < 210 °F
    - Peak containment pressure < 53 psig

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 also assessed.

Because the core initial minimum critical power ratio (MCPR) is considered in setting the bundle initial conditions for the PCT evaluation, a reasonable estimate of CPR for ATRIUM™-10™ is required. In addition, boiling transition is expected during the limiting PCT calculations. Therefore, a reasonable means of predicting this point is required. The best estimate available method for both of these purposes utilizes the GEXL97 correlation, previously approved for application to ATRIUM™-10™ fuel at another plant (Reference 38). This correlation is only used in the ATWS PCT calculations, and is used within the approved range of applicability.

10.2 Input Assumptions

Along with the initial operating conditions and equipment performance characteristics given in Table 10-1, the following assumptions were used in the analysis:

Analytical Assumptions	Bases/Justifications
The reactor is operating at 3489 MWt (100% of RTP)	Consistency with SSES current licensing basis
Initial core flow is 81.9% of RCF	Lowest core flow at rated power range to maximize the initial void fraction in the coolant, and thus more severe pressurization transient consequences. The lower initial core flow also reduces the effectiveness of the high pressure RPT function.
Both Beginning-Of-Cycle (BOC) and End-Of-Cycle (EOC) nuclear dynamic parameters were used in the calculations	Consistency with generic ATWS evaluation bases
Dynamic void and Doppler reactivity are based on a full core of ATRIUM™-10 fuel	ATWS analyses are performed conservatively compared to a nominal basis, which bounds cycle to cycle variation
The relief mode of the dual function Dual Mode Safety/Relief Valves (DS/RV) is used in the analysis to limit peak vessel pressure	Consistency with generic ATWS evaluation bases
MSIV closure starts at event initiation (time zero) for the MSIVC event	Consistency with generic ATWS evaluation bases
The LOOP event is assumed to be a loss of all auxiliary power transformers at event initiation	Consistency with generic ATWS evaluation bases

10.3 Analyses Results

Table 10-2 presents the results for the MSIVC and PRFO events. The limiting ATWS event for peak vessel pressure, Peak Clad Temperature (PCT), suppression pool heatup, and containment pressure is the PRFO. The peak vessel bottom pressure for this event is 1288 psig at 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 significantly less than the ATWS limit. The fuel cladding oxidation is insignificant and 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 psia that occurs during the 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 per NRC Information Notice 2001-13.

10.4 Conclusion

The results of the ATWS analysis performed for SSES to support operation in the MELLLA region show that the maximum values of the key performance parameters (reactor vessel pressure, suppression pool temperature, and containment pressure) remain within the applicable limits. Therefore, SSES operation in the MELLLA region has no adverse effect on the capability of the plant systems to mitigate postulated ATWS events in the expanded operating region.

Table 10-1 Operating Conditions and Equipment Performance Characteristics for ATWS Analyses

Parameter	Current Analysis
Dome Pressure (psig)	1034
Core Flow (Mlb/hr / % rated)	81.9 / 81.9
Core Thermal Power (MWt / %NBR)	3489 / 100.0
Steam / Feed Flow (Mlb/hr / %NBR)	14.44 / 100
Sodium Pentaborate Solution Concentration in the SLCS Storage Tank (% by weight)	13.6
Nominal Boron 10 Enrichment (atom %)	19.8
SLCS Injection Location	lower plenum
Number of SLCS Pumps Operating	2
SLCS Injection Rate (gpm)	82.4 total
SLCS Liquid Transport Time (sec)	30
Initial Suppression Pool Liquid Volume (ft^3)	122410
Initial Suppression Pool Temperature (°F)	90
Number of RHR cooling loops	2
RHR heat exchanger effectiveness (Btu/sec-°F)	317.5 each
Service Water Temperature (°F)	88
Transient time at which the RHR suppression pool cooling is initiated (seconds for first train/second train)	1100/1600
High Dome Pressure ATWS-RPT Setpoint (psig)	1170
DS/RV Capacity – per valve (lbm/hr) / Reference Pressure (psig) / Accumulation (%)	883950 / 1175 / 3
SRV Configuration	16 DS/RV (0 OOS)

Table 10-2 Summary of ATWS Calculation Results

Event	Exposure	Peak Vessel Pressure (psig)	Peak Cladding Temperature (°F)*	Peak Suppression Pool Temp (°F)	Peak Containment Pressure (psig)
MSIVC	BOC	1282	718	180.2	9.7
MSIVC	EOC	1287	1262	205.0	15.8
PRFO	BOC	1284	983	180.2	9.8
PRFO	EOC	1288	1420	207.1	16.5

* The fuel clad oxidation is insignificant and is less than 17%.

11.0 STEAM DRYER AND SEPARATOR PERFORMANCE

The ability of the steam dryer and separator to perform their design functions during MELLLA operation was evaluated. MELLLA decreases the core flow rate, resulting in an increase in separator inlet quality for constant reactor thermal power. These factors, in addition to core radial power distribution, affect the steam separator-dryer performance. Steam separator-dryer performance was evaluated to determine the effect of MELLLA on the steam dryer and separator operating conditions, the entrained steam (i.e., carryunder) in the water returning from the separators to the reactor annulus region, the moisture content in the steam leaving the RPV into the main steam lines, and the margin to dryer skirt uncovery.

The evaluation concluded that the performance of the steam dryer and separator remains acceptable (e.g., moisture content ≤ 0.1 weight %) in the MELLLA region.

12.0 HIGH ENERGY LINE BREAK

The following HELBs were evaluated for the effects of MELLLA:

- Main steam lines in the main steam tunnel
- Reactor Core Isolation Cooling (RCIC) steam line
- High Pressure Core Inspection (HPCI) steam line
- Reactor Water Cleanup (RWCU) line

The effect of increased subcooling due to MELLLA was evaluated based on the HELB mass/energy release profiles assumed in the current SSES design basis. Analyses were performed at rated conditions, and MELLLA conditions at minimum pump speed for the break locations listed above, taking into account the changes in enthalpy and pressure at each operating condition. Analysis of these power/flow points has shown that the blowdown mass/energy release profile at MELLLA conditions is bounding. The mass and energy releases for the following HELBs are unchanged from the pre-MELLLA conditions: Main steam line in the main steam tunnel, RCIC steam line, and HPCI steam line.

The mass and energy release profiles assumed in the current SSES design basis analysis for the RWCU line HELB analysis were reviewed at the break locations for the MELLLA conditions listed above, and the MELLLA conditions were found to be higher than the analysis of record. However, the higher mass release profiles do not result in subcompartment temperatures or pressures, which exceed existing design allowable limits.

13.0 TESTING

Required pre-operational tests (i.e., PRNMS and recirculation system flow calibrations) will be performed in preparation for operation at the MELLLA conditions with the ARTS improvements. Routine measurements of reactor parameters (e.g., APLHGR, LHGR, MAPLHGR, MLHGR, and MCPR) will be taken within a lower power test condition in the MELLLA region. Core thermal power and fuel thermal margin will be calculated using accepted methods to ensure current licensing and operational practice are maintained.

Measured parameters and calculated core thermal power and fuel thermal margin will be used to project those values at the CLTP test conditions. The core performance parameters will be confirmed to be within limits to ensure a careful monitored approach to CLTP in the MELLLA region.

The PRNMS will be calibrated prior to ARTS/MELLLA implementation. The APRM flow-biased scram and rod block setpoints will be calibrated consistent with the MELLLA implementation and all APRM trips and alarms will be tested. The power dependent setpoints of the RBM will also be calibrated consistent with the ARTS implementation.

Acceptable plant performance in the MELLLA power-flow range will be confirmed by inducing small flow changes through the recirculation flow control system. Control system changes are not expected to be required for MELLLA operation, with the possible exception of tuning following evaluation of testing. Subsequently, the recirculation system flow instrumentation calibration will be confirmed near CLTP within the MELLLA operating domain.

Steam separator and dryer performance will be evaluated by measuring the main steam line moisture content. See Section 11.0 "Steam Dryer and Separator Performance". The evaluation will be conducted near the CLTP/MELLLA boundary corner. Other test condition power/flow operating points may be tested as deemed appropriate prior to the CLTP/MELLLA boundary corner test to demonstrate the test methodology or to determine the steam moisture content at the power/flow condition.

14.0 TRAINING

General Electric will conduct a training session for plant personnel at the Susquehanna site. The attendees will include the Operations Training Instructors, who will develop the training program for the plant operators.

The General Electric training session will include discussion on the following topics:

Background on Thermal Limits and Operating Map

Current Licensing Criteria

ARTS Limits Development

MELLLA Evaluation

Technical Specifications

Impact of Revised Limits on Operating Margins

Expanded Use of Operating Map

Review of Previous SSES Startups/Power Ascensions

PRNM System

The plant operators are scheduled to receive an introduction to PRNMS prior to implementation. A detailed course in ARTS/MELLLA and its impact on plant systems will be scheduled prior to implementation in late 2006 or early 2007.

I&C technicians will receive formal training in ARTS during the PRNMS factory acceptance testing for Unit 2 equipment.

15.0 REFERENCES

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2. PPL Letter PLA-4055, George T. Jones (PPL) to C. L. Miller (NRC) "Susquehanna Steam Electric Station Proposed Amendment No. 117 to License No. NPF-22: Power Uprate With Increased Flow," dated November 24, 1993.

3. PPL Letter PLA-4173, Robert G. Byram (PPL) to C. L. Miller (NRC) "Susquehanna Steam Electric Station Proposed Amendment No. 168 to License No. NPF-14: Power Uprate With Increased Flow," dated July 27, 1994.

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Attachment 4 to PLA-5931

Revisions to

Plant-Specific Evaluations Required By

NUMAC PRNM Retrofit Plus
Option III Stability Trip Function
Topical Report (NEDC-32410P-A)

For ARTS Implementation

Susquehanna Steam Electric Station (SSES)
Units 1 & 2

Revisions to

Plant-Specific Evaluation Required By

NUMAC PRNM Retrofit Plus
Option III Stability Trip Function
Topical Report (NEDC-32410P-A)

For ARTS Implementation

EVALUATION OF SSES NUMAC POWER RANGE NEUTRON MONITOR (PRNM) COMPARED TO NUCLEAR MEASUREMENT ANALYSIS AND CONTROL (NUMAC) LICENSING TOPICAL REPORT (LTR) SECTIONS

The PPL SSES PRNMS project installation is planned in two phases. Phase 1, planned for incorporation during the Spring 2006 outage for Unit 1 and Spring 2007 for Unit 2, was described in a PPL PRNMS Digital Upgrade submittal (Reference 1).

Phase 1 includes a PRNMS digital upgrade that retains the previously approved "non-ARTS" version of the Rod Block Monitor (RBM). A full description of the Phase 1 PRNM project installation and all Technical Specification changes associated with Phase 1 were included in the prior submittal.

This Phase 2 ARTS/MELLLA submittal describes the equipment and Technical Specifications changes that are different from the configuration described in the prior Phase 1 submittal configuration, i.e., a NUMAC PRNM system including the ARTS logic. To support the power ascension plan for Extended Power Uprate, the ARTS/MELLLA implementation is scheduled during the Unit 2 Spring 2007 outage, and prior to the Unit 1 Spring 2008 outage.

The fundamental logic and setpoint changes to implement ARTS and supporting analyses and justifications are covered in Attachment 3 of this submittal. The NUMAC PRNM equipment and system, as described in the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report NEDC-32410P-A including Supplement 1 (References 2 and 3) and previously reviewed and approved by the NRC, is designed to handle, with minor hardware modifications, ARTS RBM logic. The Phase 1 submittal specifically discussed applicability of the NUMAC LTRs to the non-ARTS configuration as applied at SSES. This attachment 4 addresses only the changes in the NUMAC LTR applicability resulting from changing from non-ARTS to ARTS logic.

The implementation of ARTS logic in the NUMAC PRNM will be managed as a change from the previously completed non-ARTS NUMAC PRNM system. All software changes necessary will undergo full verification and validation activities fully equivalent to those performed for the Phase 1 installation. The specific equipment changes necessary are:

a. Replacement of the firmware in the two RBM channels, specifically in the two RBM Chassis, to remove the non-ARTS flow-biased RBM logic and replace it with the power-based trip logic. This change involves changing the basic trip logic plus the user interface (user display) to provide for different types of setpoints (power dependent vs. flow-biased) and minor changes to the readouts. Part of the change is to modify the RBM logic so that it uses APRM Simulated Thermal Power (filtered flux) instead of APRM flux for the automatic bypass and setpoint selection logic. This change reduces signal noise and the risk of unnecessary nuisance rod block alarms. The logic change also updates the status outputs to the process computer to reflect the power dependent vs. flow-biased RBM setpoints. The basic ARTS logic for the SSES RBM is the same as that previously applied at several BWRs with currently installed NUMAC PRNM systems. The change is accomplished by replacing the currently installed plug-in firmware (memory chips) with new ones on two modules in each of the two RBM chassis.

b. Replacement of the firmware in the two RBM Operator Display Assemblies (ODA) to provide status indication and displays for the power-based RBM trips in place of the non-ARTS flow-biased trip. The change is accomplished by replacing the currently installed plug-in firmware (memory chips) with new ones on one module in each of the two RBM ODA units.

c. Disconnecting and disabling two RBM "push to set-up" switches, one per RBM channel, and eight associated status lights, four for each RBM channel. These switches and associated status lights, which allow the operator to manually "step-up" the rod block limit in the current non-ARTS RBM logic, are not used in the ARTS logic. This change is accomplished by disconnecting the signal from the RBM chassis and either removing the unused equipment or marking it as not used.

d. Installing two jumpers in the PRNM panel, one in each rod block circuit, to permanently bypass (remove from the logic) the recirculation flow comparison rod block signal. As described in the PRNM NUMAC LTRs, the recirculation flow comparison rod block function is not required for the ARTS RBM.

e. Modify slightly the Multi-Vendor Data (MVD) (interface between the PRNM system and the process computer) to reflect the power-based instead of flow-biased RBM setpoints, the status of which is transmitted from PRNM.

f. Modify slightly the process computer data base to reflect the power-based instead of flow-biased setpoints.

g. Update the APRM STP flow-biased RPS trip and rod block setpoints to reflect the ARTS limits, and install the ARTS RBM setpoints.

Required changes to the Technical Specifications are as outlined in the enclosure to this submittal (PLA 5931).

The prior Phase 1 submittal included the SSES-specific responses to all "Utility Actions Required" items in the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report NEDC-32410P-A including Supplement 1. Those responses remain unchanged for the Phase 1 PRNMS. The following Phase I utility action responses have been revised to incorporate responses for the proposed change to ARTS. In the following table, the Utility Action Required identified is as stated in the Phase 1 submittal. The section numbers and Utility Actions Required listed below are from the Topical Report. In addition to the SSES-specific information, the table also includes additional justification information where the Topical Report does not specifically cover the SSES configuration. Responses apply for both SSES Unit 1 and SSES Unit 2. Only responses that change from those included in the prior Phase 1 submittal are included here.

Section No.	Utility Action Required	Response
2.3.4	<u>Plant Unique or Plant-Specific Aspects</u> Confirm that the actual plant configuration is included in the variations covered in the Power Range Neutron Monitor (PRNM) Licensing Topical Report (LTR) [NEDC-32410P-A, Volumes 1 & 2 and Supplement 1], and the configuration alternative(s) being applied for the replacement PRNM are covered by the PRNM LTR. Document in the *plant-specific licensing submittal* for the PRNM project the actual, current plant configuration of the replacement PRNM, and document confirmation that those are covered by the PRNM LTR. For any changes to the plant operator's panel, document in the submittal the human factors review actions that were taken to confirm compatibility with existing plant commitments and procedures.	The current plant configuration (after Phase 1) and the modification to the PRNM to implement the ARTS logic are included in the PRNM LTR as follows: (Applicable LTR sections are listed.) <u>No change for ARTS addition:</u> <u>Current</u> APRM 2.3.3.1.2.2 RBM 2.3.3.2.2.1 Flow Unit 2.3.3.3.2.2 Rod Control 2.3.3.4.2.2 Panel Interface 2.3.3.6.2.2 <u>For this modification</u>: <u>Current</u> <u>Proposed</u> ARTS 2.3.3.5.1.3 2.3.3.5.2.1 Other than minor display changes on the NUMAC Operator Display Assemblies, there are no changes to the plant operator's panel. A Human Factors Engineering review will be considered as part of the design inputs to the modification process.
3.4	<u>System Functions</u> As part of the *plant-specific licensing submittal*, the utility should document the following: 1) The pre-modification flow channel configuration, and any changes planned (normally changes will be either adding two channels to reach four or no change planned) NOTE: If transmitters are added, the requirements on the added transmitters should be:	1) There are no changes to the flow channels for this modification. 2) There are no changes to the APRM trips. However, as part of the change to ARTS/MELLLA, the Allowable Value and setpoints for the "Simulated Thermal Power – High" will be revised. The equipment as currently designed for the Phase 1 project (non-ARTS) includes adequate range to accommodate these setpoint changes.

Section No.	Utility Action Required	Response
	• Non-safety related, but qualified environmentally and seismically to operate in the application environment. • Mounted with structures equivalent or better than those for the currently installed channels. • Cabling routed to achieve separation to the extent feasible using existing cableways and routes. 2) Document the APRM trips currently applied at the plant. If different from those documented in the PRNM LTR, document plans to change to those in the LTR. 3) Document the current status related to ARTS and the planned post modification status as: • ARTS currently implemented, and retained in the PRNM • ARTS will be implemented concurrently with the PRNM (reference ARTS submittal) • ARTS not implemented and will not be implemented with the PRNM • ARTS not applicable	ARTS is not currently implemented. The ARTS logic is implemented by the proposed change. ARTS will be implemented via replacement of NUMAC RBM EPROMs and minor plant wiring changes. SSES LCO 3.3.2.1 will be modified to be as shown in the PRNM LTR, Volume 2, Section H.1.1, except that RBM Downscale, Function 1.e, will not be included. (See additional discussion and justification in the responses to LTR Section 8.5.1.4 and the discussion following this table.)
7.6	<u>Impact on UFSAR</u> The plant-specific action required for FSAR updates will vary between plants. In all cases, however, existing FSAR documents should be reviewed to identify areas that have descriptions specific to the current PRNM using the general guidance of Sections 7.2 through 7.5 of the PRNM LTR to identify potential areas impacted. The utility should include in the *plant-specific licensing submittal* a statement of the plans for updating the plant FSAR for the PRNM project.	Applicable sections of the FSAR will be reviewed and appropriate revisions of those sections will be prepared and approved as part of the normal design process. Following implementation of the design modification, and closure of the design package, the FSAR will be revised as part of the routine FSAR update.

Section No.	Utility Action Required	Response
8.3.6.1	APRM-Related RPS Trip Functions - Setpoints Add to or delete from the appropriate document any changed RPS setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Tech Spec submittal information with the PRNM information in the plant-specific submittal, or reference the ARTS submittal in the PRNM submittal. In the *plant-specific licensing submittal,* identify what changes, if any, are being implemented and identify the basis or method used for the calculation of setpoints and where the setpoint information or changes will be recorded.	Only the Simulated Thermal Power – High values are affected by the proposed change. The Simulated Thermal Power – High setpoints and Allowable Values for both two-loop and single-loop operation are revised to reflect the ARTS/MELLLA limits. The Allowable Values will be included in the Tech Specs or the COLR, comparable to what is currently in the SSES Tech Specs. See the SSES Tech Spec markups for the specific changes.
8.5.1.4	APRM-Related Control Rod Block Functions - Functions Covered by Tech Specs If ARTS will be implemented concurrently with the PRNM modification, include or reference those changes in the *plant-specific PRNM submittal.* Implement the applicable portion of the above described changes via modifications to the Tech Specs and related procedures and documents. In the *plant-specific submittal,* identify functions currently in the plant Tech Specs and which, if any, changes are being implemented. For any functions deleted from Tech Specs, identify where setpoint and surveillance requirements will be documented. NOTE: A utility may choose not to delete some or all of the items identified in the PRNM LTR from the plant Tech Specs.	APRM and recirculation flow rod block functions, shown only in the SSES TRM, are unchanged, except that, consistent with the LTR for ARTS plants, the recirculation flow comparator rod block function is being deleted from the TRM. The proposed change replaces the flow-biased RBM rod blocks with power-based rod blocks. To implement this change, the RBM Rod Block Functions LCO 3.3.2.1 are modified as follows: Current RBM rod block functions: 1. Low Power Range - Upscale 2. Inop 3. Downscale For the proposed change, the following functions will replace the current RBM functions: 1. Low Power Range - Upscale 2. Intermediate Power Range - Upscale 3. High Power Range - Upscale 4. Inop

Section No.	Utility Action Required	Response
		The proposed change also modifies the RBM "auto-bypass" logic to use APRM Simulated Thermal Power (STP) from the reference APRM channel instead of unfiltered APRM flux, as is used in the non-ARTS logic. The selection of setpoints in the ARTS logic in the RBM is also based on APRM STP. This change reduces the risk of spurious rod block signals and assures a clean transition between RBM setpoints as power increases or decreases. The proposed Tech Spec and Bases change to the RBM Functions are consistent with those shown in the LTR except for deletion of the RBM Downscale Function. The Bases discussions have been expanded from those shown in the LTR to provide a more complete discussion of the functions. With the implementation of the ARTS logic in the RBM, the AVs for the RBM setpoints will be relocated from LCO 3.3.2.1 to the COLR. This change is being made because the RBM power setpoints must be reconfirmed or modified on a cycle-specific basis. In addition, the surveillance and operability requirements for each RBM "power range" Function will be modified from that shown in the PRNM LTRs (for ARTS) by revision to the notes to Table 3.3.2.1-1 and SR 3.3.2.1.4. The deletion of the RBM Downscale Function is intended to simplify the Tech Spec by deleting a Function that has no significant value due to differences between the original analog equipment and the replacement digital system. [Note: See justification following this table.]

Section No.	Utility Action Required	Response
		The surveillance and operability requirements for each RBM power range are being modified from those shown in the LTR to clarify that the requirement for each range is that the applicable limits (i.e., Low Power Range limit, Intermediate Power Range limit, and High Power Range limit) 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). The previous wording 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 STP input, the digital signal that is actually used in the NUMARC 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 correct setup of the RBM. These additional surveillance and operability requirements clarifications are consistent with the PRNM LTR and result in no functional changes in the equipment performance or operational limits. See the SSES Tech Spec and Bases markup for the specific changes.
8.5.4.1.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Channel Check</u> Delete any requirements for instrument or channel checks related to RBM and, where applicable, recirculation flow rod	APRM and recirculation flow control rod block functions and related SRs are currently shown in the SSES TRM, and are unchanged except for deletion of the SR requirements for the recirculation flow comparator rod block function

Section No.	Utility Action Required	Response
	block functions (non-ARTS plants), and APRM functions. Identify in the plant-specific PRNM submittals if any checks are currently included in Tech Specs, and confirm that they are being deleted.	(which is being deleted as discussed in 8.5.1.4 above). Consistent with the PRNM LTRs, the proposed change replaces the current SR 3.3.2.1.4 requirement, which addresses only a single operability lower limit, with an SR that addresses the operability of the three power level trips in the ARTS RBM logic. As discussed in the response for item 8.5.1.4 above, the details of SR 3.3.2.1.4 have been modified from those shown in the LTR to more clearly define the requirement. The Bases description of the SR requirements have also been expanded from those in the LTR to provide more comprehensive description of the SR requirements. See the SSES Tech Spec and Bases markup for the specific changes.
8.5.6.1	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Setpoints</u> Add to or delete from the appropriate document any changed control rod block setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Tech Spec submittal information with the PRNM information in the *plant-specific submittal,* or reference the ARTS submittal in the PRNM submittal. In the *plant-specific submittal,* identify what changes, if any, are being implemented and identify the basis or method used for calculation of setpoints and where the setpoint information or changes will be recorded.	The proposed change implements ARTS/MELLLA. The Simulated Thermal Power – High rod block values, shown only in the TRM, are modified to reflect the ARTS limits. RBM Allowable Values and setpoints are modified to reflect the ARTS limits. With the implementation of ARTS logic in the RBM, the AVs for RBM will be relocated from Tech Spec Table 3.3.2.1-1 to the COLR to allow for these values to be modified on a cycle specific basis if needed. Similarly, for Phase 2, the RBM related setpoints for the power level limits will be located in the COLR rather than the Tech Spec Table 3.3.2.1-1 and SR 3.3.2.1.4 as shown in the PRNM LTRs, also to allow for these values to be modified on a cycle specific basis if needed. See the SSES Tech Spec and Bases markup for the specific changes.

Section No.	Utility Action Required	Response
None	Core Operating Limits Report Reporting requirements Section 5.6.5 do not currently address the RBM Allowable Values.	Requirements for RBM power level Allowable Values are added in 5.6.5a with reference to LCO 3.3.2.1 See the SSES Tech Spec markup for the specific changes.
9.1.3	Utility Quality Assurance Program As part of the *plant-specific licensing submittal*, the utility should document the established program that is applicable to the project modification. The submittal should also document for the project what scope is being performed by the utility and what scope is being supplied by others. For scope supplied by others, document the utility actions taken or planned to define or establish requirements for the project, to assure those requirements are compatible with the plant-specific configuration. Actions taken or planned by the utility to assure compatibility of the GE quality program with the utility program should also be documented. Utility planned level of participation in the overall V&V process for the project should be documented, along with utility plans for software configuration management and provision to support any required changes after delivery should be documented.	Quality assurance requirements for work performed at SSES are defined and described in PPL Quality Assurance Plans. For the ARTS modification to the PRNM equipment, PPL has contracted with GE to include the following PRNM scope: 1) design, 2) hardware/software, 3) licensing support, 4) training, 5) O&M manuals and design documentation, 6) EMI/RFI qualification of equipment, and 7) NMS setpoint calculation inputs. On-site engineering work to incorporate the GE-provided design information into a Design Change Package (DCP) or provide supporting, interface DCPs will be performed per the requirements of applicable PPL/SSES procedures. Modification work to implement the DCPs will be performed per PPL/SSES procedures or PPL/SSES-approved contractor procedures. PPL has participated and will continue to participate in appropriate reviews of GE's design and V&V program for the PRNM modification. For software delivered in the form of hardware (EPROMs), PPL currently intends to have GE maintain post delivery configuration control of the actual source code and handle any changes. PPL will then handle any changes in the EPROMs as hardware changes under its applicable hardware modification procedures. All changes required to implement the ARTS modification will undergo the same level of V&V as the Phase 1 design described in the prior submittal.

ADDITIONAL SSES-SPECIFIC INFORMATION

Justification for Deletion of Rod Block Monitor Downscale, Specification 3.3.2.1, with Phase 2 (implementation of ARTS)

(Ref. Para. 8.5.1.4 response above)

The effect of the differences between the original analog equipment and the replacement digital equipment on the RBM Downscale was not addressed at the time the NUMAC PRNM LTR was prepared, so this deletion was not addressed in the LTR.

The RBM Downscale Function will 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. However, no credit is taken for the RBM Downscale Function in the establishment of the RBM upscale trip setpoints or Allowable Values.

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. Therefore, the RBM Downscale Function was in fact part of the overall inop condition detection function.

The replacement of the original analog RBM equipment with the NUMAC digital RBM, which was accomplished with the Phase 1 installation covered by the prior submittal, results in all of the original analog processing being replaced by digital processing. One effect of this change is to eliminate the types of failures that can reasonably be detected by a Downscale Function. In addition, 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.

Therefore, in the NUMAC ARTS RBM, there is no incremental value or benefit provided by the RBM Downscale Function. Consistent with the overall thrust of the Improved Tech Specs to eliminate "no value" requirements, the RBM Downscale Function is being removed from the Technical Specifications and from the related discussion in the Bases. The RBM Inop Function is being retained in Technical Specifications.

References:

1. PPL Letter PLA-5880, Britt T. McKinney (PPL) to USNRC "Susquehanna Steam Electric Station Proposed License Amendment Numbers 272 for I 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

2. Licensing Topical Report NEDC-32410P-A Volumes 1 and 2, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," dated October 1995.

3. Licensing Topical Report NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," dated November 1997.