LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM AND ARTS / MELLLA IMPLEMENTATION
Enclosure 2 – Attachment 13

NEDO-33698, Revision 1

Columbia Generating Station Power Range Neutron Monitoring System Design Report on Computer Integrity, Test and Calibration, and Fault Detection

January 2012

(non-proprietary version)

HITACHI

GE Hitachi Nuclear Energy

NEDO-33698
Revision 1
DRF Section 0000-0138-4855 R3
January 2012

Non-Proprietary Information-Class I (Public)

Columbia Generating Station Power Range Neutron Monitoring System Design Report on Computer Integrity, Test and Calibration, and Fault Detection

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Revision Summary

Revision	Change Summary
0	Initial Revision
1	Updated revision number in the reference section for NEDC-33685P.

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ACRONYMS AND ABBREVIATIONS

Term	Definition
APRM	Average Power Range Monitor
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CGS	Columbia Generating Station
CTP	Core Thermal Power
DI&C-ISG	Digital Instrumenation & Control-Interim Staff Guidance
EC	Engineering Change
EMC	Electromagnetic Compatibility
FMEA	Failure Modes and Effects Analysis
GAF	Gain Adjustment Factor
GEH	GE-Hitachi Nuclear Energy Americas LLC
IEEE	Institute of Electrical and Electronics Engineers
INOP	Inoperable
IRM	Intermediate Range Monitor
LAR	License Amendment Request
LPRM	Local Power Range Monitor
LTR	Licensing Topical Report
NRC	Nuclear Regulatory Commission
NUMAC	Nuclear Measurement Analysis and Control
NUREG	Nuclear Regulatory Commission Regulation
ODA	Operators Display Assembly
OPRM	Oscillation Power Range Monitor
PRM	Power Range Monitor
PRNMS	Power Range Neutron Monitoring System
RBM	Rod Block Monitor

Term	Definition
RPS	Reactor Protection System
SER	Safety Evaluation Report
SRP	Standard Review Plan
STP	Simulated Thermal Power
TS	Technical Specifications
V&V	Verification & Validation

1. INTRODUCTION

This report provides information in support of documentation submittal requirements of a Tier 2 review identified in item 1.18 of Enclosure B of "Digital Instrumenation & Control-Interim Staff Guidance," DI&C-ISG-06 (Reference 1).

The digital Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring System (PRNMS), as described in the Licensing Topical Report (LTR) NEDC-32410P-A (Reference 2), was approved by the NRC for implementation as a retrofit for Boiling Water Reactor (BWR) plants on the condition that the recommended plant-specific actions were evaluated and incorporated.

This report addresses Reference 1 Sections D.9.4.2.5, D.9.4.2.7, D.9.4.2.10, D.9.4.3.5, D.10.4.2.5.1, D.10.4.2.5.2, D.10.4.2.5.3 and D.10.4.2.7 for the Columbia Generating Station (CGS) PRNMS. In doing so, this report demonstrates compliance with Institute of Electrical and Electronics Engineers (IEEE) Standard 603-1991, Clauses 5.5, 5.7, 5.10 and 6.5, and IEEE Standard 7-4.3.2 Clauses 5.5.1, 5.5.2, 5.5.3 and 5.7.

This report provides the basis to conclude that the CGS PRNMS installation has been designed so that: (1) the system can accomplish its safety functions under the full range of applicable conditions enumerated in the design basis; (2) the capability for testing and calibration of the safety system equipment is provided while retaining the capability of the safety systems to accomplish their safety functions; (3) the safety system is designed to facilitate timely recognition, location, replacement, repair, and adjustment of malfunctioning equipment; and (4) it is possible to check, with a high degree of confidence, the operational availability of each of the sense and command feature input sensors needed for a safety function during reactor operation, including the availability of each sense and command feature needed during the post-accident period.

2. CGS PRNMS SYSTEM INTEGRITY

2.1 Analysis Objectives

This section addresses the criteria of Standard Review Plan (SRP) Chapter 7, Appendix 7.1-C, Section 5.5 and Appendix 7.1-D Section 5.5 for Computer and System Integrity, IEEE Standard 603-1991, Clause 5.5, and IEEE Standard 7-4.3.2 sub-clause 5.5.1. This section addresses Reference 1, Sections D.9.4.2.5 and D.10.4.2.5.1 for the CGS PRNMS.

2.2 Evaluation per SRP Chapter 7, Appendix 7.1-C, Section 5.5 Requirements

2.2.1 Environmental Qualification

Requirement: Per the information provided in accordance with IEEE Std 603-1991 Clauses 4.7 and 4.8, confirm that the design includes the qualification of equipment for the conditions identified in the design bases.

Per Section 5.0 of NEDC-33685P (Reference 3), the environmental conditions for the CGS PRNMS configuration are enveloped by the conditions to which the PRNMS equipment has been qualified. The qualification of the PRNMS for environmental, seismic and Electromagnetic Compatibility (EMC) was performed in two steps: first, the qualification of the PRNM instruments in accordance with their individual instrument requirements, and then, the qualification of the PRNM panel in accordance with the panel requirements (control room environments). To meet project requirements, the qualification at both the instrument and the panel level must be demonstrated. However, the qualification at the panel level is the bounding requirement to demonstrate qualification for the PRNM equipment as installed. The qualification summary report covers both instrument and panel qualification. The qualification levels of instruments mounted in those panels are included in the Qualification Summary for Energy Northwest (ENW), CGS (Reference 4).

The CGS plant-specific environmental qualification document (Reference 4) includes the results of qualification testing for the PRNM instrument and for the PRNM instruments installed in cabinets and panels in the control room environment. The environment specified for the qualification is consistent with the design basis of CGS, which addresses the design basis conditions (for example, voltage, frequency, radiation, temperature, humidity, pressure, and vibration) identified in Clause 4.7 of IEEE Standard 603-1991.

The design basis (for example, missiles, pipe breaks, fires, loss of ventilation, spurious operation of fire suppression systems, operator error, failure in non-safety-related systems) identified in Clause 4.8 of IEEE Standard 603-1991 remains the same for the PRNM retrofit application at CGS. Per Section 9.2.6 of the NEDC-33685P (Reference 3), the CGS PRNM meets the independence requirements identified in IEEE Standard 603-1991, Clause 5.6.2. Per Section 9.2.1 of the NEDC-33685P (Reference 3), the PRNMS design meets the single failure criteria and the reliability requirements as identified in IEEE Standard 603-1991, Clause 5.1.

2.2.2 System Real-Time Performance

Requirement: Confirm that system real-time performance is adequate to ensure completion of protective action within the critical points of time identified as required by Clause 4.10 of IEEE Standard 603-1991.

PRNMS response time requirements are described in Chapter 3 of NEDC 32410P-A (Reference 2). The system and equipment architecture was selected with the specific objective

of assuring the response time requirements could be met. [[

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NEDC-33690P (Reference 5) evaluated the response time of the CGS PRNMS versus the safety analysis requirements and standard criteria for digital instrumentation and controls. The response time for the PRNM has been shown by analysis and testing to be less than the required response times, and thus, the PRNM performs sufficiently to meet safety analysis requirements (Section 2.2 of Reference 5). The NUMAC PRNMS response time is adequate to meet the Limiting Response Time of RPS consistent with the guidance provided in NUREG-0800 and Branch Technical Position (BTP) 7-21, and in accordance with the IEEE Standard 603-1991, Clause 4.10 requirement for the safety system design basis.

2.2.3 Computer System Hardware Integrity

Requirement: Evaluation of computer system hardware integrity should be included in the evaluation against the requirements of IEEE Standard 603-1991.

This IEEE Standard 603-1991 requirement with guidance from IEEE Standard 7-4.3.2 1993, is the same as IEEE Standard 279 Clause 4.5, which is addressed in the LTR (Reference 2) Section 4.4.1.1.5. All equipment required to perform APRM/Oscillation Power Range Monitor (OPRM) trip functions and to assure no inadvertent bypass is designed to operate in both the normal and abnormal plant control room environment, including EMI, and under seismic loads. Refer to Section 2.2.1 above regarding hardware integrity in design basis environment.

The PRNMS is designed to achieve system integrity in digital equipment for use in safety systems with regard to: (1) design for test and calibration in accordance with IEEE Standard 7-4.3.2 Clause 5.5.2 (provided in Section 3.3 of this report) and (2) fault detection and self-diagnostics in accordance with IEEE Standard 7-4.3.2 Clause 5.5.3 (provided in Section 4.2 of this report).

2.2.4 Computer System Software Integrity

Requirement: Computer system software integrity (including the effects of hardware-software interaction) should be demonstrated by the applicant/licensee's software safety analysis activities.

Computer system software integrity is addressed by the Software Safety Plan and the Software Safety Analysis in Sections 4.4.1.9 and 4.4.2.1 of NEDC-33685P (Reference 3).

2.2.5 Safety System Failure to a Safe State

Requirement: Confirm that the design provides for safety systems to fail in a safe state, or into a state that has been demonstrated to be acceptable on some other defined basis, if conditions such as disconnection of the system, loss of energy, or adverse environments, are experienced.

The PRNM scope is limited to one-sensor system within the Reactor Trip System, and is designed to fail-safe (tripped) by hardware means like the current power range monitor (PRM) system design.

The single failure-proof design of the PRNMS, as described in the LTR (Reference 2), meets the requirements of IEEE Standard 279-1971, Clause 4.2 (Section 4.4.1.1.2 of the LTR). The failure analysis for the PRNMS is provided in Section 6 of Volume 1 of the LTR and in Appendix F of Volume 2 of the LTR.

The replacement design has been specifically designed to have the same or more conservative "fail safe" failure modes as the current system (Appendix G of Volume 2 of the LTR, Reference 2). [[

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2.2.6 Automatic Actions on Detection of Inoperable Input Instruments

Requirement: The system should, upon detection of inoperable input instruments, automatically place the protective functions associated with the failed instrument(s) into a safe state (e.g., automatically place the affected channel(s) in trip), unless the operator has already placed the affected channel in a bypass mode (this would change a two-out-of-four logic to a two-out-of-three logic).

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2.2.7 Hardware or Software Failures Detected by Self-diagnostics

Requirement: Hardware or software failures detected by self-diagnostics should place a protective function into a safe state or leave the protective function in an existing safe state. Failure of computer system hardware or software should not inhibit manual initiation of protective functions or the operator performance of preplanned emergency or recovery actions.

The testability and self-test capability of the PRNMS are provided in Section 5.3.11 of the LTR. Section 6.3.5 of Reference 2 provides the self-test coverage.

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2.2.8 Actions on Partial or Full System Initialization or Shutdown after a Loss of Power

During either partial or full system initialization or shutdown after a loss of power, control output to the safety system actuators should fail to a predefined, preferred failure state. A system restart upon restoration of power should not automatically transfer the actuators out of the predefined failure state. Changes to the state of plant equipment from the predefined state following restart and reinitialization (other than changes in response to valid safety system signals) should be under the control of the operator in accordance with appropriate plant procedures.

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2.3 Evaluation per SRP Chapter 7, Appendix 7.1-D, Section 5.5 Requirements

In addition to the system integrity criteria required by IEEE Standard 603-1991, and the guidance in Subsection 5.5 of SRP Appendix 7.1-C, IEEE Standard 7-4.3.2-2003 includes criteria in sub-clauses 5.5.1 through 5.5.3 for designs for computer integrity, test and calibration, and fault detection and self-diagnostics activities. The following are necessary to achieve system integrity in digital equipment for use in safety systems:

- Design for computer integrity (sub-clause 5.5.1 - addressed in this section)
- Design for test and calibration (sub-clause 5.5.2 - addressed in Section 3.3.1)
- Fault detection and self-diagnostics (sub-clause 5.5.3 - addressed in Sections 3.3.2 through 4.2)

2.3.1 Design for computer integrity (IEEE Std 7-4.3.2 Sub-Clause 5.5.1)

Requirement: The computer is designed to perform its safety function when subjected to conditions, external or internal, that have significant potential for defeating the safety function.

As described in Sections 2.2.1 through 2.2.8 above, the computer is designed to perform its safety function when subjected to conditions, external or internal, that have significant potential for defeating the safety function.

3. CGS PRNMS CAPABILITY FOR TEST AND CALIBRATION

3.1 Analysis Objectives

This section addresses the criteria of SRP Chapter 7, Appendix 7.1-C, Sections 5.7 and 6.5, and Appendix 7.1-D Section 5.7 for the Capability for Test and Calibration, and demonstrates compliance with these requirements. In addition, this evaluation demonstrates compliance with IEEE Standard 603-1991, Clauses 5.7 and 6.5, IEEE Standard 7-4.3.2 Clauses 5.5.2 and 5.7, and addresses Staff Guidance of DI&C-ISG-06 Sections D.9.4.2.7, D.9.4.3.5, D.10.4.2.5.2 and D.10.4.2.7 for the CGS PRNMS.

3.2 Evaluation per SRP Chapter 7, Appendix 7.1-C, Section 5.7 and 6.5 Requirements

3.2.1 Periodic Testing (Appendix 7.1-C, Section 5.7)

Requirement: Periodic testing should duplicate, as closely as practical, the overall performance required of the safety system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another.

Section 5.3.11 of the LTR (Reference 2) describes the testability and self-test capability of the PRNM system, including overlap testing from one test segment to another. The PRNMS supports the continued performance of surveillance tests per the requirements of the Technical Specifications (TS) as discussed in Section 3.2.3 of this report.

3.2.2 Test Provisions Should Address Increased Potential for Subtle System Failures (Appendix 7.1-C, Section 5.7)

Requirement: Test provisions should address the increased potential for subtle system failures such as data errors and computer lockup.

The test provisions provided to address subtle system failures are the continuous self-test and watchdog timer. Section 5.3.11 of the LTR (Reference 2), provides the testability and self-test capability of the PRNMS. Section 6.3.5 (Reference 2) provides the self-test coverage. [[

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3.2.3 Design Supports Types of Testing Required by Technical Specifications (Appendix 7.1-C, Section 5.7)

Confirm that the system design supports the types of testing required by the Technical Specifications. The system design should also support the compensatory actions required by

Technical Specifications when limiting conditions for operation are not met. The design should allow for tripping or bypass of individual functions in each safety system channel.

The PRNMS design supports testing required by TS, including channel checks, channel functional testing, channel calibrations, response time testing, and logic system functional testing. Sections 8.3.4 and 8.4.4 of the LTR (Reference 2) describe the recommended changes to channel checks, channel functional testing, channel calibrations, response time testing and how these changes are supported by the PRNMS design. Sections 8.3.5 and 8.4.5 of Reference 2 describe the recommended changes to the logic system functional testing and how these changes are supported by the design. Additionally, these sections specify the ENW action to implement changes to TS to ensure they are revised accordingly for the PRNMS design. See the CGS plant-specific responses (Reference 6), for more detailed information about the utility action taken.

The PRNMS design supports the performance of the compensatory actions required by TS when limiting conditions for operation are not met. [[

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3.2.4 Checking Operational Availability of Sensors (Appendix 7.1-C, Section 6.5)

Requirement: Means shall be provided for checking the operational availability of each sensor required for a safety function.

The PRNM design maintains the same sensor check capability for the LPRM detectors and the recirculation flow sensors as exists in the current PRM design.

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3.3 Evaluation per SRP Chapter 7, Appendix 7.1-D, Section 5.7 Requirements

3.3.1 Test and Calibration Functions Have No Adverse Effect on System Performance (IEEE Standard 7-4.3.2, Sub-clause 5.5.2)

Requirement: Test and calibration functions should not adversely affect the ability of the system to perform its safety function.

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3.3.2 Fault Detection/Self-Diagnostics and Partial System Failures (IEEE Standard 7-4.3.2, Sub-clause 5.5.3)

Requirement: Fault detection and self-diagnostics are one means that can be used to assist in detecting partial system failures that could degrade the capabilities of the computer system, but may not be immediately detectable by the system.

As described in Section 3.2.2 of this report, the test provisions provided to address subtle system failures are the continuous self-test and watchdog timer. Section 5.3.11 of the LTR (Reference 2) describes the testability and self-test capability of the PRNMS. Section 6.3.5 (Reference 2) describes the self-test coverage.

3.3.3 Use of a Non-Software Watchdog Timer

Requirement: Use of a non-software watchdog timer is critical in the overall diagnostic scheme.

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]] (Section 6.3.5 of the LTR - Reference 2).

4. CGS PRNMS REPAIR, FAULT DETECTION AND SELF-DIAGNOSTICS

4.1 Analysis Objectives

This section addresses the criteria of BTP 7-17 (failure detection, self-test and surveillance testing), and demonstrates compliance with those requirements. In addition, this evaluation demonstrates compliance with IEEE Standard 603-1991, Clause 5.10 and IEEE Standard 7-4.3.2 Clause 5.5.3, and addresses the Staff Guidance provided by DI&C-ISG-06 Sections D.9.4.2.10 and D.10.4.2.5.3 for the CGS PRNMS.

4.2 Evaluation per BTP 7-17 Requirements

The objectives of the BTP are to confirm that:

- The safety system, including self-test, is designed for in-service testability commensurate with the safety functions to be performed through all modes of plant operation. (Additional information regarding this topic is included in Section 3.2.1 of this report.)
- The positive aspects of self-test features are not compromised by the additional complexity that may be added to the safety system by the self-test features. (Additional information regarding this topic is included in Section 3.3.1 of this report.)
- Hardware and software design support the required periodic testing. (Additional information regarding this topic is included in Section 3.2.1 of this report.)
- Failure modes assumed to be detectable by the single-failure analysis are in fact detectable. Failures may be detectable by observing operational characteristics as well as other methods. (Specific information regarding this topic is included in Section 4.2.1 of this report.)

4.2.1 Failures Detected are Consistent with Assumptions in Single Failure Analyses and FMEA

Requirement: Failures detected by hardware, software, and surveillance testing should be consistent with the failure detectability assumptions of the single-failure analysis and the failure modes and effects analysis.

The automatic self-test and surveillance functions included in the PRNM are described in Section 6.3 of the LTR (Reference 2), which includes a discussion of the self-test coverage and the methods used to confirm that the self-test functions are operating. The self-test functions are integrated into the main PRNM equipment, and are designed to the same qualification, independence, integrity, single failure and Verification & Validation (V&V) requirements. The overall self-test design and surveillance provisions are consistent with the guidance of the BTP. The PRNMS failure analysis is described in Section 6 and Appendix F of the LTR (Reference 2).

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]] The self-test coverage is described in Section 6.3.5 of the LTR (Reference 2). The PRNMS failure analysis considers equipment failures, and functional failures, as described in Sections 6.2 and 6.3 of the LTR (Reference 2).

4.2.2 Self Test of Computer System on System Initialization

Requirement: The design includes self-test features to confirm computer system operation on system initialization.

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4.2.3 Continuous Self-Testing of Computer System

Requirement: The system includes continuous self-testing. Self-tests include monitoring memory and memory reference integrity, using watch-dog timers or processors, monitoring communication channels, monitoring central processing unit status, and checking data integrity.

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Continuous self-testing, use of watchdog timers, monitoring of communication channels and monitoring central processing unit status, et al. are discussed in Sections 3.2.2, 3.3.2 and 3.3.3 of this report. Section 5.3.11 of the LTR (Reference 2), provides the testability and self-test capability of the PRNMS; Section 6.3.5 (Reference 2) provides the self-test coverage. These sections demonstrate that the monitoring tasks listed in this requirement are performed.

4.2.4 Design Maintains Independence, Integrity and Meets Single-Failure Criterion

Requirement: The design of automatic self-test features should maintain channel independence, maintain system integrity, and meet the single-failure criterion during testing. The scope and extent of interfaces between software that performs protection functions and software for other functions such as self-test should be designed to minimize the complexity of the software logic and data structures. The safety classification of the hardware and software used to perform automatic self-testing should be equivalent to that of the tested system unless physical, electrical, and communications independence are maintained such that no failure of the test function can inhibit the performance of the safety function.

The automatic self-test and surveillance functions included in the PRNM are described in the LTR (Reference 2), which includes a discussion of the self-test coverage and the methods used to confirm that the self-test functions are operating. The self-test functions are integrated into the main PRNM equipment, and are designed to the same qualification, independence, integrity, single failure and V&V requirements. The overall self-test design and surveillance provisions are consistent with the guidance of the BTP.

4.2.5 Benefit of Self-Test Not Compromised by Complexity

Requirement: The positive aspects of self-test features should not be compromised by the additional complexity that may be added to the safety system by the self-test features. The improved ability to detect failures provided by the self-test features should outweigh the increased probability of failure associated with the self-test feature.

The automatic self-test and surveillance functions included in the PRNM are described in the LTR (Reference 2), which includes a discussion of the self-test coverage and the methods used to confirm that the self-test functions are operating. The self-test functions are integrated into the main PRNM equipment, and are designed to the same qualification, independence, integrity, single failure and V&V requirements. The overall self-test design and surveillance provisions are consistent with the guidance of the BTP.

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]] (Section 5.3.3.1 of the LTR, Reference 2).

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4.2.6 Self-Test Functions Verified

Requirement: Self-test functions should be verified during periodic functional tests.

The self-test function is verified during the channel functional test steps of the periodic surveillance tests described in Section 4.2.7. For details of the CGS plant specific actions taken to verify the APRM self-test functions during channel functional tests, refer to Section 8.3.4.2.4 of Reference 6.

4.2.7 System Supports Periodic Surveillance Testing per TS

Requirement: Systems should be able to conduct periodic surveillance testing consistent with the technical specifications and plant procedures. As delineated in Regulatory Guide 1.118, periodic

testing consists of functional tests and checks, calibration verification, and time response measurements.

Section 5.3.11 of the LTR (Reference 2) describes the testability and self-test capability of the PRNMS. The PRNMS supports the continued performance of surveillance tests per the requirements of the Technical Specifications, which include [[

]] For details of the CGS plant-specific actions taken regarding Technical Specification surveillance testing requirements, see Reference 6.

4.2.8 Indication of Bypassed Protective Action

Requirement: If the protective action of some part of a protection or safety system is bypassed or deliberately rendered inoperative for testing, that fact should be continuously indicated in the control room. Provisions should also be made to allow operations staff to confirm that the system has been properly returned to service.

For information regarding indication of bypasses, see Section 9.2.8.3 of NEDC-33685P (Reference 3), which discusses compliance with IEEE Standard 603-1991, Clause 5.8.3. [[

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4.2.9 Tests Should Not Require Makeshift Test Setups

Requirement: Test procedures for periodic tests should not require makeshift test setups. For digital computer-based systems, makeshift test setups, including temporary modification of code or data that must be appropriately removed to restore the system to service, should be avoided.

Per Section 8.3.5.3 of the LTR (Reference 2), [[

]] so the risk of problems caused by the normal operation of the system is greatly reduced.

Per Section 6.3.4, Table Note 3 of the LTR (Reference 2), [[

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4.2.10 Automatic Tests Credited with Performance of Surveillance Tests

Requirement: If automatic test features are credited with performing surveillance test functions, provisions should be made to confirm the execution of the automatic tests during plant operation. The capability to periodically test and calibrate the automatic test equipment should also be provided. The balance of surveillance and test functions that are not performed by the automatic test feature should be performed manually to meet the intent of Regulatory Guide 1.118. In addition, the automatic test feature function should conform to the same requirements and considerations (e.g., test interval) as the manual function.

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4.2.11 Safety Classification/Quality of the Hardware/Software Used for Periodic Testing

Requirement: The safety classification and quality of the hardware and software used to perform periodic testing should be equivalent to that of the tested system. The design should maintain channel independence, maintain system integrity, and meet the single-failure criterion during testing. Commercial digital computer-based equipment used to perform periodic testing should be appropriately qualified for its function.

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4.2.12 Compensatory Action on Detection of Any Failed or Inoperable Component

Requirement: The design should have either the automatic or manual capability to take compensatory action on detection of any failed or inoperable component. The design capability and plant technical specifications, operating procedures, and maintenance procedures should be consistent with each other.

The design capability of the PRNMS to take either automatic or manual compensatory action on detection of any failed or inoperable component is addressed in Sections 2.2.5, 2.2.6 and 2.2.7, and throughout this document.

Plant operating and maintenance procedures will be updated during the implementation phase for consistency with changes in the system design capability and the plant TS in accordance with plant procedures and per the Engineering Change (EC) process. The procedural changes that

reflect the PRNMS man-machine interface are validated during operator training on the replacement system.

4.2.13 Plant Procedures Specify Manual Compensatory Actions

Requirement: Plant procedures should specify manual compensatory actions and mechanisms for recovery from automatic compensatory actions.

Plant operating and maintenance procedures will be updated during the implementation phase for consistency with changes in the system design capability and the plant TS in accordance with plant procedures and per the EC process. The procedural changes that reflect the PRNMS man-machine interface are validated during operator training on the replacement system.

As discussed in Sections 4.2.8, 4.2.12, and 4.2.14, the PRNMS provides the controls and indications necessary to perform all required compensatory actions relative to the PRNMS.

4.2.14 Operator Notification of Detected Failures

Requirement: Mechanisms for operator notification of detected failures should comply with the system status indication provisions of IEEE Standard 603-1991 and should be consistent with, and support, plant technical specifications, operating procedures, and maintenance procedures.

Operator notification of detected failures and channel trip and bypass indications comply with the system status indication provisions of IEEE Standard 603-1991 [[

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5. REFERENCES

1. NRC Interim Staff Guidance DI&C-ISG-06, Revision 1, Digital Instrumentation & Controls; Licensing Process, January 19, 2011 (ADAMS Accession No, ML110140103).

2. NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function, Licensing Topical Report," October 1995 (Including SER); (Including Supplement 1, November 1997).

3. GE Hitachi Nuclear Energy, "Digital I&C-ISG-06 Compliance for Columbia Generating Station NUMAC Power Range Monitoring Retrofit Plus Option III Trip Function," NEDC-33685P, Revision 1, January 2012.

4. NUMAC Power Range Neutron Monitoring (PRNM) Components 268X1331TCG001, 268X1332TCG001, G002 268X1333TCG001 Qualification Summary for Energy Northwest (ENW), Columbia Generating Station (CGS), Revision 1.

5. GE Hitachi Nuclear Energy, "Columbia Generating Station Power Range Neutron Monitoring System Response Time Analysis Report," NEDC-33690P, Revision 0, November 2011.

6. GE Hitachi Nuclear Energy, "Columbia Generating Station Plant-Specific Responses Required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A)," Report# 0000-0101-7647-R3, October 2011.

List of Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal intended or planned actions, are provided for information purposes, and are not considered to be regulatory commitments.

COMMITMENT	TYPE	SCHEDULED COMPLETION DATE
Submit the following Digital I&C-ISG-06 specified Phase II information: • 2.1 Safety Analysis (D.4.4.2.1) • 2.2 V&V Reports (D.4.4.2.2) • 2.3 As-Manufactured, System Configuration Documentation (D.4.4.2.3) • 2.4 Test Design Specification (D.4.4.2.4) • 2.5 Summary Test Reports (Including FAT) (D.4.4.2.4) • 2.6 Summary of Test Results (Including FAT) (D.4.4.2.4) • 2.7 Requirement Traceability Matrix (D.9.4.2) • 2.8 FMEA (D.9.4.2.1.1) • 2.9 System Build Documents (D.4.4.3.5) • 2.14 System Response Time Confirmation Report (D.9.4.2.4) • 2.15 Reliability Analysis (D9.4.2.15, D10.4.2.15) • 2.16 Setpoint Calculations (D.9.4.3.8) • 2.17 Software Tool Analysis Report (D.10.4.2.3.2) • Commercial Grade Dedication Report(s) (D.10.4.2.4.2)	One Time	June 29, 2012.
Incorporate Limiting Trip Setpoint values, and the methodology for determining these values, into the Licensee Controlled Specifications.	One Time	Prior to startup from outage that installs the PRNM modification, (currently planned for spring 2013).

Description and Evaluation of the Proposed TS Changes

Subject: Technical and Regulatory Evaluation of License Amendment Request to Change Technical Specifications (TS) in Support of ARTS / MELLLA Implementation

1.0 TECHNICAL EVALUATION

1.1 APRM and RBM Allowable Values (AVs)
1.2 APRM Allowable Value Setdown Requirement
1.3 Summary of Safety Analyses in Attachment 1
1.4 ARTS Related Changes to RBM
1.5 ATWS Analysis in consideration of Information Notice 2001-13 and Impact on Standby Liquid Control (SLC) System
1.6 Standby Liquid Control (SLC) Boron-10 Enrichment Increase
1.7 Application of TSTF-493 and Impact to Setpoints for Proposed ARTS / MELLLA Changes
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2.0 REGULATORY EVALUATION

2.1 Applicable Regulatory Requirement/Criteria
2.2 Precedent
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3.0 REFERENCES

ATTACHMENTS to Enclosure 3:

1. NEDC-33507P, Revision 1, "Columbia Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," January 2012 (proprietary version)

2. NEDO-33507, Revision 1, "Columbia Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," January 2012 (non-proprietary version)

1.0 Technical Evaluation

Energy Northwest is planning to implement the **A**verage Power Range Monitor (APRM) / **R**od Block Monitor (RBM) / **T**echnical **S**pecifications, referred to collectively herein as ARTS, and Maximum Extended Load Line Limit Analysis (MELLLA) improvements in conjunction with the hardware changes being introduced with the Power Range Neutron Monitor (PRNM) upgrade discussed in Enclosures 1 and 2 of this License Amendment Request (LAR).

The expanded operating domain includes changes for ARTS / MELLLA consistent with approved operating domain improvements at other BWRs. The current ELLLA power-flow upper boundary is modified to include the operating region bounded by the MELLLA boundary line which passes through the 100% current licensed thermal power and 80.7% rated core flow point (see Figure 1-1 of Attachment 1 to this enclosure). The power-flow region that is above the current ELLLA boundary is referred to as the MELLLA region. As part of ARTS / MELLLA, the current flow-biased RBM would be replaced by a power-dependent RBM with the upgrade to the digital PRNM System discussed in Enclosure 2. The change from the flow-biased RBM to the power-dependent RBM would also require new Allowed Values (AVs).

The ARTS / MELLLA application is evaluated on a plant-specific basis via a safety and system response 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 limits and transient/accident performance. The effect of this operating mode relative to fuel dependent analyses has been evaluated to confirm compliance with the required fuel thermal margins during plant operation. For subsequent reload cycles, Columbia Generating Station (CGS) will include the ARTS / MELLLA operating condition in the reload analysis. Attachment 1 of this enclosure presents the results of the safety analyses and system response evaluations for the non-fuel dependent tasks. Attachment 1 also presents the assumptions and conclusions that will be validated or updated for the fuel dependent tasks performed for operation of CGS in the region above the current 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 operating domain would be extended to 80.7% flow at rated power for two loop operation. To accommodate this expanded operating domain, the APRM flow-biased Simulated Thermal Power (STP) AV would be revised. The APRM clamp will be unchanged. The MELLLA region would not be used for single loop operation.

The improvements associated with ARTS include:

- Existing power and flow dependent MCPR thermal limits, similar to that used by BWR/6 plants, are validated for ARTS.

- The APRM trip setdown and Total Peaking Factor (TPF) are replaced by more direct power-dependent and flow-dependent Linear Heat Generation Rate (LHGR) 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 operating limits.

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

- 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.

1.1 APRM and RBM AVs

Although it is part of the CGS design configuration and TS, the APRM flow-biased STP AV is not credited in any specific CGS safety analysis. The proposed AV change would permit 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 that the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and fuel thermal-mechanical design limits are not exceeded, when appropriate power-dependent trip setpoints are used in the RBM. Other transients are analyzed further in Sections 3.0 and 7.0 of Attachment 1 to this Enclosure.

The APRM flow-biased STP AV varies as a function of reactor recirculation loop flow, but is clamped such that it is always less than the APRM neutron flux-high AV. The proposed change is described further in Section 1.2 below. Justification for making this change is provided in Sections 3.0 and 7.0 of Attachment 1 to this enclosure.

The flow-biased RBM AVs will be replaced by power-dependent AVs. 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. The proposed change is described further in Section 1.4 below. Justification for making this change is provided in Section 4.0 of Attachment 1 to this enclosure.

1.2 APRM Allowable Value Setdown Requirement

LCO 3.2.4 currently requires the APRM flow-biased STP AV to be reduced when the Fraction of Rated Thermal Power (FRTP) is less than the Maximum Fraction of Limiting Power Density (MFLPD). The 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 flow-biased STP AV, the APRM gains may be adjusted such that the APRM readings are greater than or equal to 100% times MFLPD. The CGS normal operating practice is to adjust APRM gains when required to meet LCO 3.2.4. Each APRM channel is typically bypassed while the required gain adjustment is made.

The basis for this setdown requirement originated from the Hench-Levy Minimum Critical Heat Flux Ratio (MCHFR) thermal limit criterion. The GE Critical Quality - Boiling Length correlation (GEXL) Critical Power Ratio has since replaced the Hench-Levy Critical Heat Flux Ratio as the approved means of determining departure from nucleate boiling.

The following criteria were applied to support removal of the APRM trip setdown requirement, LCO 3.2.4, "APRM Gain and Setpoint":

- MCPR SL shall not be violated as a result of any AOO.
- All fuel thermal-mechanical design bases shall remain within the licensing limits.
- Peak cladding temperature and maximum cladding oxidation fraction following a Loss of Coolant Accident (LOCA) shall remain within the limits specified in 10 CFR 50.46.

Power and flow dependent adjustments to the MCPR and LHGR thermal limits have been determined using NRC approved analytical methods identified in TS 5.6.3. These adjustments will ensure that the three criteria discussed above are met during operation at other than rated conditions without the APRM trip setdown. Justification for making this change is provided in Sections 3.0 and 7.0 of Attachment 1 to this enclosure.

1.3 Summary of Safety Analyses Included in Attachment 1

Safety analyses performed in support of the proposed changes are described in Attachment 1 to this enclosure. These changes include fuel performance event evaluations (Sections 3.0 and 4.0), an evaluation of vessel overpressure protection (Section 5.0), an evaluation of thermal-hydraulic stability (Section 6.0), an evaluation of the loss-of-coolant accident (Section 7.0), containment response evaluations (Section 8.0), reactor internals integrity evaluations (Section 9.0), an evaluation of an anticipated transient without scram (Section 10.0), an evaluation of steam dryer and separator performance (Section 11.0), and high energy line break evaluations (Section 12.0). A description of planned testing is included in Section 13.0. The following technical analysis summarizes or supplements the information in Attachment 1.

Attachment 1, Section 1.0, Introduction, and Section 2.0, Overall Analysis Approach, provide a description and background for the implementation of ARTS / MELLLA at CGS. The content of Sections 1.0 and 2.0, relative to fuel dependent evaluations, describes the approach CGS is taking to justify and implement the ARTS / MELLLA bases. The assumptions and conclusions described in Section 1.0 and 2.0 for fuel dependent evaluations are based upon the CGS Cycle 20 core design using GE14 and ATRIUM-10 fuel and in some cases on existing analyses for plants similar to CGS.

The content of Attachment 1, Sections 1.0 and 2.0, relative to non-fuel dependent evaluations, describes the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases and reflect the CGS configuration. The assumptions and conclusions described in Sections 1.0 and 2.0 relative to non-fuel dependent evaluations are applicable for CGS.

Attachment 1, Sections 3.0 - Fuel Thermal Limits, 4.0 - Rod Block Monitor System Improvements, 5.0 - Vessel Overpressure Protection, and 6.0 - Thermal-Hydraulic Stability, describe particular aspects of the implementation of ARTS / MELLLA for CGS Cycle 20. These sections describe fuel dependent evaluations which summarize the approach taken to justify and implement the ARTS / MELLLA bases. The assumptions and conclusions for the fuel dependent evaluations are based upon CGS Cycle 20 core design using GE14 and ATRIUM-10 fuel.

Attachment 1, Section 7.0, Loss-of-Coolant Accident Analysis, describes a fuel-dependent evaluation. Analysis in this section is based on a full core of GE14 fuel, which was determined to be conservative with respect to ATRIUM-10 fuel for Emergency Core Cooling Systems (ECCS) LOCA analysis and was representative of the CGS Cycle 20 core. The content of this section describes the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases and reflects the CGS plant configuration.

Attachment 1, Section 8.0, Containment Response, describes a non-fuel dependent evaluation. This section describes the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases and reflects the CGS plant configuration. The assumptions and conclusions described are applicable for CGS.

Attachment 1, 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. Section 9.0 describes the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases and reflects the current CGS plant configuration. The assumptions and conclusions described are applicable for CGS. Evaluation of the results from Section 9.1 indicates that the existing non-fuel dependent CGS Increased Core Flow (ICF) bases are bounding relative to the MELLLA application and therefore no fuel-dependent evaluations will be required to justify the ARTS / MELLLA bases.

Attachment 1, 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 CGS plant specific evaluation using inputs related to the CGS Cycle 20 core. The contents of this section describe the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases, and is discussed further in Section 1.5 below.

Attachment 1, 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. These sections describe the approach Energy Northwest is taking to justify and implement the ARTS / MELLLA bases and reflect the CGS plant configuration. The assumptions and conclusions described are applicable for CGS.

Attachment 1, Section 13.0, Testing, describes the planned testing which will be performed in support of the ARTS / MELLLA implementation.

1.4 ARTS Related Changes to RBM

Additional changes to the TS introduced with ARTS improvements include LCO 3.3.2.1, "Control Rod Block Instrumentation," to reflect power-dependent RBM setpoints and elimination of the downscale trip (see TS Markups in Enclosure 1).

1.4.1 Deletion of RBM Downscale Function

The deletion of the RBM Downscale Function is intended to simplify the TS by deleting a Function that has no significant value due to differences between the original analog equipment and the replacement digital system.

The effect of the differences between analog equipment and the digital equipment on the RBM Downscale Function was not addressed at the time the NUMAC PRNM LTR was prepared, so this deletion was not addressed in the LTR. The originally intended RBM Downscale Function would 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 Allowed Values (AVs). 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 RBM Downscale Function. In an 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 will result 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 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 ability to detect 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 PRNM RBM, with ARTS incorporated, there is no incremental value or benefit provided by the RBM Downscale Function. Consistent with the overall thrust of the improved Standard Technical Specifications to eliminate "no value" requirements, the RBM Downscale Function is being removed from the TSs and from the related discussion in the Bases. The RBM Inop Function is being retained in TSs. Removal of the RBM Downscale function was approved by the NRC for implementation of ARTS/MELLLA at Nine Mile Point 2 (Reference 1).

1.4.2 Additional Discussion of TS Table 3.3.2.1-1 Changes

This table would be modified to change from a flow-biased RBM to a power-dependent RBM consistent with the ARTS improvement. NUREG-1433, (Reference 2), reflects specifications that include ARTS RBM limits, and the changes proposed for CGS are modeled after the NUREG. Deviations from the NUREG format would include specifying the RBM AVs for Low Power Range - Upscale, Intermediate Power Range - Upscale, and High Power Range - Upscale in the COLR. The relocation to the COLR would also include the MCPR limits which define when the RBM is required to be operable, as a function of power level.

Consistent with the implementation of the ARTS improvement, the RBM AVs are modified to reflect ARTS limits. With the implementation of ARTS logic in the RBM, the AVs for the RBM will be located in the COLR, rather than the TS Table 3.3.2.1-1 to allow for these values to be modified on a cycle specific basis as needed. These changes are similar to those previously approved for Susquehanna (Reference 8).

The current exception contained within note (a) listed in the APPLICABLE MODE OR OTHER SPECIFIED CONDITIONS column of Table 3.3.2.1-1, for when a peripheral control rod is selected, will be maintained in the new applicability notes (a) through (c) for the RBM Functions. The RBM will continue to be automatically bypassed if a peripheral control rod is selected. This exception is consistent with the ARTS based RBM applicability notes previously approved for Cooper (Reference 4). Additionally, notes (a) through (c) have been written based on APRM STP input, the digital signal that is actually used in the NUMAC RBM, not thermal power as specified in the NUREG.

Notes (d) and (e) reflect application of actions to address the industry setpoint methodology issue as specified in TSTF-493. See Section 1.7 for discussion on

application of TSTF-493 to the proposed ARTS / MELLLA improvements and related TS changes.

1.4.3 Deletion of Applicability of SR 3.3.2.1.4 to RBM Inop Function in TS Table 3.3.2.1-1

The RBM Inop Function is not affected by the proposed implementation of ARTS / MELLLA. As discussed in Section 4.2 of Attachment 1, a count of active LPRMs is made automatically and the RBM channel is declared inoperable if too few detectors are available.

The current CGS TS Table 3.3.2.1-1 note (a), requires the RBM Inop Function 1.b to be operable when thermal power is ≥ 30% RTP and no peripheral control rod is selected. The current SR 3.3.2.1.4 also requires verification that the RBM Inop Function is not bypassed when THERMAL POWER is ≥ 30% RTP and when a peripheral control rod is not selected. Under the conditions where the RBM Inop Function is allowed to be bypassed, the RBM is not required to be operable, and hence no surveillance is needed to demonstrate the system "Inop function" is operable.

Consistent with the changes described above, the CGS ARTS / MELLLA application proposes a revised SR 3.3.2.1.4 and revised Table 3.3.2.1-1 function applicability notes. The revised SR is worded such that that the SR excludes the RBM Inop Function. The revised function applicability notes in Table 3.3.2.1-1 require that the RBM Inop Function is not bypassed at APRM Simulated Thermal Power ≥ 28% RTP with MCPR less than the limits specified in the COLR and no peripheral control rod selected.

The deletion of the applicability of SR 3.3.2.1.4 to the RBM Inop Function is consistent with the improved Standard Technical Specifications presented in Reference 2. These proposed changes are consistent with the changes approved for Nine Mile Point 2 (Reference 1).

1.4.4 TS 5.6.3, Core Operating Limits Report (COLR)

The addition of TS 5.6.3.a.5 identifies the requirements of TS 3.3.2.1 (RBM) that have been added to the COLR specification. This approach is discussed above in section 1.4.2 of this enclosure.

1.5 ATWS Analysis in Consideration of Information Notice 2001-13 and Impact on Standby Liquid Control (SLC) System

For ARTS / MELLLA, the ATWS analysis is revised to reduce the number of pumps required for ATWS mitigation from two pumps to one. LCO 3.1.7, "Standby Liquid Control (SLC) System," will continue to require both pumps to be operable. ATWS analysis for MELLLA conditions shows that the pump flow rate of 82.4 gpm (two pump flow) can be reduced to 41.2 gpm (one pump flow) if the enrichment of Boron-

10 in the SLC solution is correspondingly doubled above the current required enriched boron levels. This change is being made to improve reliability (one pump credited compared to two) and to maintain margin to the SLC relief valve setpoint. The original SLC design criterion for the maximum pump discharge pressure was based on the lowest relief valve setpoint for the main steam Safety Relief Valves (SRVs) operating in the relief mode. This has been generally replaced by the use of plant specific ATWS transient pressure results occurring during the time the SLC system is analyzed to be in operation in consideration of NRC Information Notice 2001-13 (Reference 5).

For ARTS / MELLLA operation, the maximum reactor upper plenum pressure during the limiting transient is 1155 psig, resulting in a two-pump SLC system discharge pressure of 1322 psig. This would reduce margin to the SLC system relief valve setpoint. To maintain relief valve margin, Energy Northwest intends to use only one SLC pump for ATWS mitigation. The single-pump flow rate is lower than two-pump, resulting in lower pipe line flow losses. The maximum SLC system pump discharge pressure depends primarily on the Safety Relief Valve (SRV) setpoints. The maximum SLC system pump discharge pressure during the limiting ATWS event using one SLC system pump is 1209.5 psig. This value is based on a peak reactor vessel upper plenum pressure of 1155 psig that occurs during the limiting ATWS event after SLC system initiation. These values reflect the implementation of NRC Information Notice 2001-13 (Reference 5) assumptions. This reactor vessel pressure is within the previously analyzed pump design pressure of 1365 psig. The pump discharge pressure for ARTS / MELLLA remains within the design capability of the SLC pumps.

The SLC pump relief valve setpoint margin is calculated as the difference between the relief valve setpoint and the maximum SLC pump discharge pressure. Generally, a margin of 75 psig provides sufficient margin against inadvertent relief valve lifting. The 75 psig is based on an allowance for a 3% relief valve setpoint drift (1372 psig setpoint) and a 30 psig allowance for a SLC pump pressure pulse. For ARTS / MELLLA operation during the limiting ATWS event, the maximum single-pump SLC discharge pressure is 1209.5 psig. This results in a minimum relief valve setpoint margin of 162.5 psig, which meets the required margin of greater or equal to 75 psig. Therefore, there is adequate margin to prevent the SLC relief valve from lifting during SLC system operation to address the lessons learned in NRC Information Notice 2001-13 (Reference 5).

In the event that the SLC system is initiated before the time that the reactor pressure recovers from the first transient peak, resulting in opening of the SLC pump relief valves, the reactor pressure must reduce sufficiently to ensure SLC pump relief valve closure. The analysis indicates that the reactor pressure reduces sufficiently from the first transient peak to allow the SLC pump relief valves to close.

Attachment 1, Section 10.0, discusses the results of the ATWS analysis performed for ARTS / MELLLA conditions. The ATWS analysis resulted in a peak upper plenum pressure that is 102 psig greater than the current analysis. The increased

upper plenum pressure is offset by the reduced system pressure losses from injection with one SLC pump instead of two. The net result is no change in the pump discharge pressure (which is specified in SR 3.1.7.6). The increase in peak upper plenum pressure is not due to implementation of ARTS / MELLLA, but rather to differences in the modeling assumptions used in the revised ATWS analysis as discussed above.

The current and proposed changes to the SLC system parameters are shown below.

	Current Parameter	Proposed Parameter
SLC Flow Rate (gpm)	82.4 (two-pump)	41.2 (one-pump)
Minimum Boron Concentration (weight percent)	13.6	13.6
Boron-10 Enrichment (atom percent)	22*	44

* Amendment 221 (Reference 6) justified 22 atom percent, however 44 atom percent was conservatively implemented in anticipation of future improvements such as this LAR.

The ATWS analysis results for ARTS / MELLLA show acceptable suppression pool temperatures for SLC injection with one pump if the enrichment of Boron-10 in the sodium-pentaborate solution is increased to 44 atom percent. Peak suppression pool temperature increases to 187.4°F, which is less than the design limit of 204.5°F.

The SLC continues to meet the requirements contained in 10 CFR 50.62(c)(4) for SLC injection capability for ATWS events. The combination of the neutron absorber boron enrichment of 44 atom percent, minimum solution concentration of 13.6 weight percent, and minimum SLC pump flow rate of 41.2 gpm exceeds the equivalency in control capacity of 86 gallons per minute of 13 weight percent sodium-pentaborate solution for a 251-inch inside diameter reactor vessel contained in 10 CFR 50.62(c)(4). Section 1.6 below provides additional discussion of compliance with the equivalency requirement contained in 10 CFR 50.62(c)(4).

CGS has implemented Alternative Source Term in accordance with 10 CFR 50.67. The SLC system is credited to inject sodium-pentaborate solution into the RPV in response to a LOCA to control the pH in the suppression pool for dose mitigation. The LOCA analysis only credits one SLC pump for injection and is therefore not affected by the change in pump flow rate credited in the ATWS analysis. The pH of sodium-pentaborate is determined by the concentration of the solution, which remains at least 13.6 weight percent, so the change in Boron-10 enrichment has no

impact on the LOCA analysis. The changes to SLC described above do not affect its ability to meet its design function.

1.6 Standby Liquid Control (SLC) Boron-10 Enrichment Increase

The NRC approved Amendment 221 (Reference 6) for CGS that justified a sodium pentaborate solution enrichment of 22 atom percent boron-10 to achieve an equivalent concentration of 780 ppm natural boron in the reactor. During the application for the license amendment request that resulted in approval of Amendment 221, Energy Northwest proposed to conservatively double the analytically determined atom percent boron-10 requirements in order to support planned future improvements at the site. The enrichment was conservatively increased to allow for anticipated future improvements, such as operation in the MELLLA domain as proposed by this LAR.

The SLC is designed to shut down the reactor from rated power conditions to cold shutdown in the postulated situation that some or all of the control rods cannot be inserted. This manually operated system pumps a sodium pentaborate solution into the vessel, to provide neutron absorption and achieve a sub-critical reactor condition. The SLC system is designed to inject over a wide range of reactor operating pressures.

An increase in Boron-10 enrichment is required to support the change from two-pump to one-pump SLC system operation for the ATWS analysis described above. Since credited flow rate is reduced in half (from 82.4 to 41.2 gpm), Boron-10 enrichment doubles (from 22 to 44 atom percent). 10 CFR 50.62(c)(4) requires that each BWR have a SLC system with a minimum flow capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent sodium pentaborate solution. NEDE-31096-P-A (Reference 7) provides guidance for boron equivalency determinations. Equation 1-1 of that document was used to demonstrate injection capacity equivalency as follows:

$$\frac{Q}{86} * \frac{M_{251}}{M} * \frac{C}{13} * \frac{E}{19.8} \geq 1$$

where:

Q	=	SLC system flow rate, gpm;
M_{251}	=	the mass of water in the reactor vessel and recirculation system at rated conditions for the reference plant (a 251 inch diameter reactor vessel), lbm;
M	=	the mass of water in the reactor vessel and recirculation system at rated conditions, lbm;
C	=	expected sodium pentaborate solution concentration, weight percent; and
E	=	minimum expected Boron-10 isotope enrichment, atom percent.

Since CGS has a 251 inch diameter reactor vessel, the value of M_{251}/M is equal to 1. Applying the values of the remaining parameters, which were assumed in the ATWS analysis, yields:

$$\frac{41.2}{86} * 1 * \frac{13.6}{13} * \frac{44.0}{19.8} = 1.11 \geq 1$$

Thus, this requirement of 10 CFR 50.62 is satisfied.

Accordingly, this calculation confirms that the currently approved TS SR 3.1.7.9, which verifies that sodium pentaborate enrichment is ≥ 44 atom percent Boron-10 prior to addition to the SLC tank, ensures SLC system operation remains consistent with analytical bases. This change does not have any impact on SLC operation or the ability of the system to perform its shutdown function. Operation within the Acceptable Operation region of TS Figure 3.1.7-1, with a sodium pentaborate enrichment of ≥ 44 atom percent Boron-10 in accordance with SR 3.1.7.9, will achieve the required concentration equivalent to 780 ppm natural boron in the reactor core.

There are no significant impacts of the new sodium pentaborate solution on the mechanical and electrical aspects of the SLC system. The SLC pump, motor, and system valves are capable of delivering the required minimum flow rate to the reactor vessel under worst case postulated operating conditions. Since operation of only one pump is required, the margin between the maximum pump discharge pressure and the nominal setpoint of the pump discharge relief valve is maintained. This is mainly due to the reduced system back pressure resulting from the lower pipe line flow losses for one-pump compared to two-pump operation.

The NRC has previously approved similar changes for Susquehanna regarding the number of SLC pumps required for ATWS mitigation (Reference 8).

1.7 Application of TSTF-493 and Impact to Setpoints for Proposed ARTS / MELLLA Changes

The NRC approved Revision 4 of TSTF-493 via issuance of a model application for adoption on April 30, 2010 (Reference 9). Using the guidance of Appendix A of TSTF-493, Energy Northwest has applied the actions identified to this LAR; the results being that the two notes specified in the TSTF are applied to channel calibration SR 3.3.2.1.5 for the following RBM functions listed in TS Table 3.3.2.1-1 as follows:

- Rod Block Monitor - Low Power Range - Upscale (1.a)
- Rod Block Monitor - Intermediate Power Range - Upscale (1.b)
- Rod Block Monitor - High Power Range - Upscale (1.c)

In order to implement this change, Energy Northwest will revise the Licensee Controlled Specifications (LCS) to include the Limiting Trip Setpoint values and the

methodologies used for determining these setpoints prior to the startup from the refueling outage that this modification is installed. Energy Northwest plans on implementing the TSTF-493 required notes without deviation, for the affected setpoints.

The Bases for TS 3.3.2.1 describe the application of the notes to SR 3.3.2.1.5 as applied to RBM Functions 1.a, 1.b, and 1.c. Draft marked-up pages of the affected TS Bases are provided in Attachment 3 of Enclosure 1, for information only. In addition, Energy Northwest calibration procedures for these RBM functions will be revised to reflect the instruction given in the above notes.

1.8 Conclusion

With the above changes the CGS TS appropriately reflect the implementation of ARTS / MELLLA improvements, ensuring design requirements and acceptance criteria are met.

Incorporation of the ARTS / MELLLA improvements as described above will increase operating flexibility in power ascension and operation at rated power. Replacement of the APRM setdown requirement with more direct power and flow dependent thermal limits will reduce the need for manual AV or gain adjustments and allow for more direct thermal limits administration. This will improve the human/machine interface, update thermal limits administration, and provide more direct protection of plant limits.

Reducing the amount of SLC pumps required to meet ATWS mitigation requirements provides increased operational flexibility in meeting required surveillance requirements while continuing to ensure that the SLC system satisfies 10 CFR 50.62 requirements for all future proposed core reloads.

2.0 Regulatory Evaluation

2.1 Applicable Regulatory Requirements / Criteria

2.1.1 10 CFR Part 50

10 CFR 50.36, "Technical Specifications," provides the regulatory requirements required in the TS. As stated in 10 CFR 50.36, TS include SRs to assure that the LCOs are met. The proposed TS changes would revise SRs, LCOs, Required Actions and Completion Times, as applicable, for each change in APRM and RBM functions and related LCOs.

The CGS Neutron Monitoring System was designed and licensed to the General Design Criteria (GDC) specified in 10 CFR 50 Appendix A. The GDCs related to the proposed changes are discussed below.

- Criterion 10 – "Reactor design." The reactor core and associated coolant, control, and protection systems shall 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.

- Criterion 12 – "Suppression of reactor power oscillations." 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.

- Criterion 13 – "Instrumentation and control." Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems with prescribed operating ranges.

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

- Criterion 21 – "Protection system reliability and testability." The protection system shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

- Criterion 22 – "Protection system independence." The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design

techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

- Criterion 24 – "Separation of protection and control systems." The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

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

- Criterion 29 – "Protection against anticipated operational occurrences." The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

- Criterion 50 – "Containment design basis." The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall 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. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by §50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

The SLC system was designed and licensed to the following reactivity control related GDCs:

- Criterion 26 - "Reactivity control system redundancy and capability." Two independent reactivity control systems of different design principles shall be provided....The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power

changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

- Criterion 27 - "Combined reactivity control systems capability." The reactivity control system shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity change to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Other applicable Regulations:

- 10 CFR 50.46 sets forth acceptance criteria for the performance of the Emergency Core Cooling Systems (ECCS) following postulated Loss of Coolant Accidents (LOCAs). 10 CFR 50 Appendix K describes the required and acceptable features of evaluation models used to calculate ECCS performance.

- In 10 CFR 50.62, requirements for reduction of risk from ATWS events are specified. Paragraph (c)(4) of 10 CFR 50.62 states, in part that "Each boiling water reactor must have a standby liquid control (SLC) system with the capability of injecting into the reactor pressure vessel a borated water solution at such a flow rate, level of boron concentration and boron-10 isotope enrichment, and accounting for reactor pressure vessel volume, that the resulting reactivity control is at least equivalent to that resulting from injection of 86 gallons per minute of 13 weight percent sodium pentaborate decahydrate solution at the natural boron-10 isotope abundance into a 251-inch inside diameter reactor pressure vessel for a given core design."

- In 10 CFR 50.36(d)(1)(ii)(A), the NRC states, in part, that "where a limiting safety system setting (LSSS) is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded."

- In 10 CFR 50.36(c)(3), the NRC states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation would be within safety limits, and that the limiting conditions for operation would be met."

- Regulatory Guide (RG) 1.105, "Setpoints for Safety-Related Instrumentation," describes a method that the NRC staff finds acceptable for use in complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and will remain within the TS limits.

- TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions," (Reference 9) addresses NRC concerns that the TS requirements for Limiting Safety System Settings (LSSS) may not be fully in compliance with the intent of 10 CFR 50.36.

Energy Northwest has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria and finds that implementation of ARTS / MELLLA and changing the required number of operating pumps for the SLC system continue to meet all regulatory requirements. Based on the Technical Evaluation provided above in Section 1.0, the proposed TS amendment:

1. Does not alter the design or function of any reactivity control system;
2. Does not result in any change in the qualifications of any component; and
3. Does not result in the reclassification of any component's status in the areas of shared, safety related, independent, redundant, and physical or electrical separation.

Hence, there is reasonable assurance the health and safety of the public remain unaffected following approval of this change.

2.2 Precedents

Precedents are discussed in the relevant sections above where the specific changes are described. In addition, the NRC has approved the TS improvements of ARTS / MELLLA with other plants that have also implemented the Power Range Neutron Monitor (PRNM) upgrade similar to what CGS is proposing with this LAR. Two such plants are Nine Mile Point Unit 2 and Susquehanna 1 and 2 (References 1 and 10 respectively).

2.3 Significant Hazards Consideration

Energy Northwest 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," as discussed below:

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

Response: No.

The proposed change eliminates the APRM flow-biased STP setdown requirement and substitutes power and flow dependent adjustments to the 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 SL will not be violated as a result of any AOO, and that the fuel thermal and mechanical design bases will be maintained.

The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased STP AV and by the replacement of the current flow-biased RBM with a new power-dependent RBM. As discussed in the Technical Evaluation Section 1.0 above, and Attachment 1, operation in the MELLLA expanded operating domain will not increase the probability or consequences of previously analyzed accidents. The APRM and RBM are not involved in the initiation of any accident, and the APRM flow-biased STP function is not credited in any CGS safety analyses. The proposed change will not introduce any initial conditions that would result in NRC approved criteria being exceeded and the APRM and RBM will remain capable of performing their design functions.

The SLC system is provided to shutdown the reactor without reliance on control rod movement, to mitigate anticipated transients without scram (ATWS) events and provide suppression pool pH control following a LOCA. As such, SLC is not considered an initiator of an ATWS event, LOCA or any other analyzed accident. The revised SLC pump flow rate and increased Boron-10 enrichment continue to meet the shutdown requirement of SLC. The changes do not reduce the ability of the SLC system to respond to or mitigate an ATWS event or LOCA. Nor do these changes increase the likelihood of a system malfunction that could increase the consequences of an accident.

Based on the above discussion, it is concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2.3.2 Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change eliminates the APRM flow-biased STP setdown requirement and substitutes power and flow dependent adjustments to the MCPR and 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.

The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased STP AV and the replacement of the current flow-biased RBM with a new power-dependent RBM. Changing the formulation for the APRM

flow-biased STP AV and changing 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. While not credited for MCPR SL protection, the APRM flow-biased STP AV and associated scram trip setpoint will continue to provide a redundant trip for the credited trip functions (such as APRM Fixed Neutron Flux - High or Reactor Pressure - High). The power-dependent RBM will prevent rod withdrawal when the power-dependent RBM rod block setpoint is reached, thus protecting MCPR SL. No new failure mechanisms, malfunctions, or accident initiators are being introduced by the proposed change. 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 kind of accident from any accident previously evaluated.

The proposed change to the SLC pump flow rate credited in the ATWS analysis, in conjunction with the increased enrichment of Boron-10 in the sodium pentaborate solution, is consistent with the functional requirements of the ATWS rule (10 CFR 50.62). These proposed changes do not involve the installation of any new or different type of equipment, do not introduce any new modes of plant operation, and do not change any methods governing normal plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

2.3.3 Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed change eliminates the APRM flow-biased STP setdown requirement and substitutes power and flow dependent adjustments to the MCPR and LHGR thermal limits. Replacement of the APRM setdown requirement with power and flow dependent adjustments to the MCPR and LHGR thermal limits will continue to ensure that margins to the fuel cladding SL 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 SL will not be violated as a result of any AOO, and that the fuel thermal and mechanical design bases will be maintained.

The proposed change also expands the power and flow operating domain by relaxing the restrictions imposed by the formulation of the APRM flow-biased STP AV and the replacement of the current flow-biased RBM with

a new power-dependent RBM. The APRM flow-biased STP AV and associated scram trip setpoint will continue to initiate a scram, providing a redundant trip that is not credited for protection of MCPR SL. 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 remain within the licensing limits during a control rod withdrawal error event and to ensure that the MCPR SL will not be violated as a result of a control rod withdrawal error event. Operation in the expanded operating domain will not alter the manner in which SLs, Limiting Safety System Setpoints (LSSSs), or limiting conditions for operation are determined. AOOs and postulated accidents within the expanded operating domain will continue to be evaluated using NRC approved methods. The 10 CFR 50.46 acceptance criteria for the performance of the ECCS following postulated LOCAs will continue to be met.

The proposed change to the SLC flow rate, in conjunction with the increased Boron-10 enrichment in the sodium pentaborate solution, credited in the ATWS analysis continues to meet accident analyses limits. The proposed change is consistent with the functional requirements of the ATWS rule (10 CFR 50.62) and the flow rate credited for LOCA suppression pool pH control. The ability of the SLC system to respond to and mitigate an ATWS event or LOCA is not affected.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

2.4 Conclusions

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

3.0 REFERENCES

1. NRC Letter to NMP2, "Nine Mile Point Nuclear Station, Unit No. 2 – Issuance of Amendment Re: Implementation of ARTS/MELLLA (TAC No. MD5233)," dated February 27, 2008 (ADAMS Accession No ML080230230).

2. NUREG-1433, Rev. 3, "Standard Technical Specifications General Electric Plants, BWR/4," dated March 2004.

3. NRC letter to Exelon Nuclear, "Peach Bottom Atomic Power Station, Units 2 and 3 – Issuance of Amendment Re: Activation of Oscillation Power Range Monitor Trip (TAC Nos. MC2219 and MC 2220)," dated March 21, 2005 (page 4 of SE) (ADAMS Accession No. ML050270020).

4. NRC to Nebraska Public Power District, "Cooper Nuclear Station – Issuance of Amendment Re: Revise Technical Specification Surveillance Requirement 3.3.2.1.4 and Table 3.3.2.1-1 (TAC No. MC0629)," dated December 22, 2004 (ADAMS Accession No. ML043630055).

5. NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin," dated August 10, 2001.

6. NRC to Mark E. Reddemann (Energy Northwest), "Columbia Generating Station – Issuance of Amendment Re: Increased Boron Concentration in Standby Liquid Control System (TAC NO. ME4789)," May 18, 2011 (ADAMS Accession No. ML 111170370)

7. NEDE-31096-P-A, "Anticipated Transients Without Scram; Response to NRC ATWS Rule 10 CFR 50.62," dated February 1987.

8. NRC to PPL Susquehanna, LLC, " Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendment Re: Standby Liquid Control System (TAC Nos. MD1424 AND MD1425)," dated February 28, 2007 (ADAMS Accession No. ML070390215).

9. "Model Application for Adoption of TSTF Traveler TSTF-493, Revision 4, 'Clarify Application of Setpoint Methodology for LSSS Functions' Option A, Addition of Surveillance Notes," April 30, 2010 (ADAMS Accession No. ML100710442).

10. NRC to PPL Susquehanna, LLC, "Susquehanna Steam Electric Station, Units 1 and 2 - Issuance of Amendment Re: Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) Implementation (TAC Nos. MC9040 and MC9041)," dated March 23, 2007 (ADAMS Accession No. ML070720675).