ENERGY NORTHWEST

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GO2-12-116

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING LICENSE AMENDMENT REQUEST TO IMPLEMENT PRNM/ARTS/MELLLA**

References:
1) Letter, GO2-12-017, dated January 31, 2012, BJ Sawatzke (Energy Northwest) to NRC, "License Amendment Request to Change Technical Specifications in Support of PRNM / ARTS / MELLLA Implementation"
2) Letter dated July 23, 2012, NRC to ME Reddemann (Energy Northwest), "Columbia Generating Station – Request for Additional Information Regarding License Amendment Request to Implement PRNM/ARTS/MELLLA (TAC NO. ME7905)"

Dear Sir or Madam:

By Reference 1, Energy Northwest requested approval of a license amendment request to revise the Columbia Generating Station Technical Specifications to reflect improvements in the Average Power Range Monitor / Rod Block Monitor Technical Specifications (ARTS) and expand the facility operating domain to reflect operations using the Maximum Extended Load Line Limit Analysis (MELLLA). These improvements coincide with the installation of the digital General Electric-Hitachi (GEH) Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring (PRNM) System.

Via Reference 2, the Nuclear Regulatory Commission (NRC) requested additional information related to the Energy Northwest submittal. Transmitted herewith in Attachment 1 is the Energy Northwest response to the request for additional information.

Regulatory commitments are identified in Attachment 2. Should you have any questions or require additional information regarding this matter, please contact Mr. Z. K. Dunham, Licensing Supervisor, at (509)377-4735.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the date of this letter.

A001
NRR

Respectfully,

W. [signature] for BJ Sawatzke

B. J. Sawatzke
Vice President, Nuclear Generation & Chief Nuclear Officer

Attachments: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Senior Resident Inspector/988C
AJ Rapacz – BPA/1399
W.A. Horin – Winston & Strawn
JO Luce – EFSEC
RR Crowley - WDOH

Response to Request for Additional Information

NRC Request:

1. On page 1-2 of NEDC-33507P, Revision 1, "Columbia Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," January 2012 (nonproprietary version designated as NEDO-33507, Revision 1, available at ADAMS Accession No. ML12040A088), plants with either full or partial Average Power Range Monitor/Rod Block Monitor/Technical Specifications/Maximum Extended Load Line Limit Analysis (ARTS/MELLLA) were listed. The NRC staff requests the licensee to clarify the following:

   a. Does CGS plan to implement full or partial ARTS/MELLLA?

   b. Please provide the basis for your decision in part (a) and the main differences between the two options with respect to CGS plant operation.

Energy Northwest Response:

a. CGS plans to implement full ARTS/MELLLA.

b. Full ARTS/MELLLA includes the hardware for the ARTS Rod Block Monitoring (RBM) system, while partial ARTS/MELLLA does not include this hardware. The Power Range Neutron Monitoring (PRNM) System hardware includes the RBM system. For full ARTS plants, the RBM setpoint is the basis for the statistical analysis used to determine the 95/95 Rod Withdrawal Error (RWE) Operating Limit Minimum Critical Power Ratio (OLMCPR). Partial ARTS plants must use the maximum unblocked rod result for each control rod as the basis for the statistical analysis.

NRC Request:

2. Please provide the CGS core design for which the MELLLA analysis was performed and the following results of fuel-dependent portion of the analysis:

   a. Please describe core design for the current Cycle 20 for which the MELLLA analysis was performed, including number of GE14 and ATRIUM-10 fuels assumed to comprise the core, and their burnup history (i.e., number of cycles exposed in the core) for each fuel types.

   b. For a given core condition of a mixed core, the Specified Acceptable Fuel Design Limits (SAFDLs), such as minimum critical power ratio (MCPR), linear heat generation rate (LHGR), and peak cladding temperature (PCT), depend on the type of fuel design and their respective burnup history. Which fuel design (GE14 or

ATRIUM-10) produces the limiting SAFDLs for the transient and accident analyses performed at the MELLLA conditions? Please explain why.

Energy Northwest Response:

a. The core design for Cycle 20 is described in Tables 2-1a and 2-1b and Figure 2-1 which lists the bundles in the core and maps the core loading.

b GEH transient analyses performed for the implementation of ARTS/MELLLA are compared to bounding, burnup-independent, fuel specific screening criteria to protect against violation of the SAFDLs for MCPR, LHGR and cladding strain limits. The Safety Limit Minimum Critical Power Ratio (SLMCPR) is the same for both fuel types. The bounding cladding strain and LHGR screening limits were higher for ATRIUM-10 fuel. The PCT limit is the same for both fuel types. For the mixed ATRIUM-10 and GE14 core analyzed for the implementation of ARTS/MELLLA, the GE14 fuel was bounding with respect to approaching the SAFDLs (MCPR, LHGR, cladding strain) during a transient. The GE14 fuel was also bounding for PCT during an accident. This is because the analysis was performed based on the CGS Cycle 20 core that consisted of once and twice burned ATRIUM-10 fuel assemblies and fresh GE14 fuel assemblies. Fresh fuel generally produces more limiting results with respect to approaching the SAFDLs due to the fact that the fresh assemblies operate at higher power levels than the once or twice burned fuel. The following Tables 2-1a, 2-1b and Figure 2-1 define the number of GE14 and ATRIUM-10 fuel bundles in the CGS Cycle 20 core, the number of cycles exposed in the core, and the current core location:

Table 2-1a Bundle Type Inventory

Description	Type	Cycle Loaded
ATRIUM-10	3	18
ATRIUM-10	4	18
ATRIUM-10	5	19
ATRIUM-10	6	19
GE14	7	20
GE14	8	20
GE14	10	20
GE14	12	20

Table 2-1b Bundle and Batch Average Exposure Ranges (MWd/ST)

		Beginning of Cycle		End of Rated Target[1]	
Type	**Number in Core**	**Minimum Bundle**	**Batch Average**	**Batch Average**	**Maximum Bundle**
3	84	31931.	36913.	42008.	44559.
4	156	26505.	33805.	42781.	46033.
5	200	14408.	19998.	35033.	39032.
6	72	14725.	17548.	32827.	37865.
7	80	0.	0.	18572.	18929.
8	112	0.	0.	17453.	18544.
10	40	0.	0.	17911.	18992.
12	20	0.	0.	18083.	18417.

Note 1: Based on an end of rated (EOR) incremental exposure of 13695. MWd/ST.

Figure 2-1 Bundle Type Array

(Bundle types defined in Table 2-1a)

	1	3	5	7	9	11	13	15	17	19	21	23	25	27	29	31	33	35	37	39	41	43	45	47	49	51	53	55	57	59
60									4C	3	4	3	4C	4C	4	4	4C	4C	3	4	3	4C								
58								3C	4	3C	4	5C	3	5	3	4	5	3	5C	4	3C	4	3C							
56						3	4	3	4	5C	4	5	5	6	6	6	6	5	5	4	5C	4	3	4	3					
54						3	3C	5	5C	5	6	10	5	8	8	8	8	5	10	6	5	5C	5	3C	3					
52					3	4	4	5	4	6	7	10	8	7	5	5	7	8	10	7	6	4	5	4	4	3				
50			3	3	4	4C	6	6	8	5	7	4	10	5	12	12	5	10	4	7	5	8	6	6	4C	4	3	3		
48			4	3C	4	6	4	8	6	8	5	12	5	10	5	5	10	5	12	5	8	6	8	4	6	4	3C	4		
46		3C	3	5	5	6	8	12	8	5	8	6	7	5	8	8	5	7	6	8	5	8	12	8	6	5	5	3	3C	
44	4C	4	4	5C	4	8	6	8	6	8	4	8	5C	8	4	4	8	5C	8	4	8	6	8	6	8	4	5C	4	4	4C
42	3	3C	5	5	6	5	8	5	8	6	7	5	8	5	7	7	5	8	5	7	6	8	5	8	5	6	5	5	3C	3
40	4	4	4	6	7	7	5	8	4	7	5	10	4	7	5	5	7	4	10	5	7	4	8	5	7	7	6	4	4	4
38	3	5C	5	10	10	4	12	6	8	5C	10	4	7	5C	8	8	5C	7	4	10	5	8	6	12	4	10	10	5	5C	3
36	4C	3	5	5	8	10	5	7	5C	8	4	7	4	7	5	5	7	4	7	4	8	5C	7	5	10	8	5	5	3	4C
34	4	5	6	8	7	5	10	5C	8	5	7	5C	7	5	7	7	5	7	5C	7	5	8	5C	10	5	7	8	6	5	4
32	4	3	6	8	5	12	5C	8	4	7	5	8	5	7	4	4	7	5	8	5	7	4	8	5C	12	5	8	6	4	4
30	4	4	6	8	5	12	5C	8	4	7	5	8	5	7	4	4	7	5	8	5	7	4	8	5C	12	5	8	6	4	3
28	4	5	6	8	7	5	10	5C	8	5	7	5C	7	5	7	7	5	7	5C	7	5	8	5C	10	5	7	8	6	5	4
26	4C	3	5	5	8	10	5	7	5C	8	4	7	4	7	5	5	7	4	7	4	8	5C	7	5	10	8	5	5	3	4C
24	3	5C	5	10	10	4	12	6	8	5C	10	4	7	5C	8	8	5C	7	4	10	5C	8	6	12	4	10	10	5	5C	3
22	4	4	4	6	7	7	5	8	4	7	5	10	4	7	5	5	7	4	10	5	7	4	8	5	7	7	6	4	4	4
20	3	3	5	5	6	5	8	5	8	6	7	5	8	5	7	7	5	8	5	7	6	8	5	8	5	6	5	5C	4	3
18	4C	4	3	5C	4	8	6	8	6	8	4	8	5C	8	4	4	8	5C	8	4	8	6	8	6	8	4	5C	3	4	4C
16		3C	3	5	5	6	8	12	8	5	8	6	7	5	8	8	5	7	6	8	5	8	12	8	6	5	5	3	3C	
14			4	3C	4	6	4	8	6	8	5	12	5	10	5	5	10	5	12	5	8	6	8	4	6	4	3C	4		
12			3	3	4	4C	6	6	8	5	7	4	10	5	12	12	5	10	4	7	5	8	6	6	4C	4	3	3		
10					3	4	4	5	4	6	7	10	8	7	5	5	7	8	10	7	6	4	5	4	4	3				
8						3	4	5	5C	5	6	10	5	8	8	8	8	5	10	6	5	5C	5	4	3					
6						3	3	3	3	5C	4	5	5	6	6	6	6	5	5	4	5C	3	3	3	3					
4								3C	4	3	4	5C	3	5	4	4	5	3	5C	4	3C	4	3C							
2									4C	3	3C	3	4C	4C	4	3	4C	4C	3	4	3	4C								

[1] C indicates a previous cycle control cell bundle which was controlled in the previous cycle

NRC Request:

3. The fourth bullet from the bottom of page 7-2 of NEDC-33507P, Revision 1, states, "A full core of GE14 fuel is assumed to comprise the core."

   a. Please clarify which of the transient and accident analyses (i.e., anticipated operational occurrences (AOOs), loss-of-coolant accident (LOCA), American Society of Mechanical Engineers (ASME) Overpressure and ATWS) assumed a full core of GE14 fuel.

   b. Please explain why GE14 fuel was assumed to comprise the entire core when the core included ATRIUM-10 fuel, as well.

   c. If a full core of GE14 fuel was assumed to comprise the core, please explain how the analysis captures a situation when ATRIUM-10 fuel is more limiting in the transient and accident analyses, and provide reasonable assurance as to how the analysis of record is most conservative.

Energy Northwest Response:

a. A full core of GE14 fuel is assumed for the emergency core cooling system (ECCS)-LOCA analysis.

b, c The methodology of the SAFER/GESTR Evaluation Model (NEDE-30996P-A, Volume I, October 1987) is defined to analyze a single bundle to be representative of the "hot" bundle. The balance of the core is representative as an "average" bundle, or channel, insofar as the core thermal-hydraulics goes, and boundary conditions for fuel rod heating following a LOCA event. For the analysis for PCT, the most limiting bundle that can be construed from the core is defined for consideration as the "hot bundle." Upon that limiting bundle the several power distribution parameters are superimposed; mean average peak linear heat generation rate, radial peaking factor, axial power shape, all at a design, maximum allowed or achievable value, so as to arrive at an overall bounding condition for the hot spot, or location of maximum cladding heat-up. That result is used as the bounding temperature for the core (PCT) for use to demonstrate compliance (less than 10CFR50.46 Acceptance Criteria.)

For a core composed of bundles of different design, the same overall objective remains, to represent, or determine, the single most limiting hot spot in the core to define PCT. The time of exposure which produces the limiting cladding heating in terms of PCT following an accident is at the time of maximum fuel pellet densification. This occurs fairly early in exposure (~4000 MWd/MTU), during the first cycle of operation in which the fuel bundle is resident. The most recently inserted of the fuel bundle designs would demonstrate this limiting PCT behavior, generally, given similar power distribution limits as noted above. For CGS, that fuel would be GE14. Also, General Electric has determined both through internal studies as well as licensed applications involving several plants with mixed cores, given the maximum allowed operational parameters supported by the fuel bundle design, that the GE14 fuel bundle would be more limiting in PCT than the residual ATRIUM-10 bundle, assuming either as the hot bundle in the Evaluation Model. These demonstrations support the use of the single GE14 bundle as suitable for the PCT calculation in the SAFER/GESTR Evaluation Model.

NRC Request:

4. In Table 7-1, "ECCS-LOCA Analysis Bases for CGS ARTS/MELLLA," of NEDC-33507P, Revision 1, the emergency core cooling system-LOCA rated thermal power was assumed to be equal to 3629 megaWatts (MWt), which is 4.1 percent (143 MWt) higher than the current licensed thermal power (CLTP) of 3486 MWt. The NRC staff understands that Appendix K to 10 CFR Part 50 requires analyses to be performed at 2 percent higher than the CLTP to allow for instrumentation error. Please explain why the LOCA analysis was performed at a thermal power which is 4.1 percent higher than the CLTP.

Energy Northwest Response:

The power level of 3629 MWt has been consistently used as an analysis basis for the ECCS-LOCA analysis for CGS.

The original SAFER/GESTR analysis (NEDC-32115P, September 1992) assumed core power as 9.2% higher than the (then) current licensed thermal power of 3323 MWt. The intent was that the analysis would remain valid, conservative for operating conditions to which the plant had been licensed at the time, but would afford a valid licensing analysis basis for later potential changes. As higher power is always more conservative (greater PCT resulting), this represents attributable margin in the analysis as was a practice in the industry. The margin in initial power assumption would allow for a later action to take advantage of thermal power optimization or a stretch uprate as the plant might desire to pursue.

A subsequent stretch power uprate was studied for implementation, as documented in NEDC-32141P, June 1993, which had as its basis an uprated power of 3486 MWt, a reported 5% increase. It cited the results of the analysis above, as it was conservative and sufficient to show compliance.

A subsequent ECCS-LOCA analysis had been performed supporting operation assuming introduction of the GE14 fuel bundle in the core (May 2008). As the stretch uprate had been implemented, accounting for the power increase mentioned above, that analysis was performed consistently at 3629 MWt, but described as 4.1% higher than the (then and now) current licensed thermal power of 3486 MWt.

The PRNM/ARTS/MELLLA analysis to address ECCS-LOCA response, as part of the submittal currently under review, maintained the same basis for power level. Thus, this assumption constitutes attributable margin in the conservative analysis, maintained at the licensee's direction. In point of fact, the Appendix K assumption for the current analysis was applied by taking the 2% uncertainty value on power from this baseline; that is, 106.1% of CLTP (3702 MWt) as used in determining the reported Licensing Basis PCT, keeping clear the 4.1% attributable margin as conservatism in the analysis.

NRC Request:

5. Please describe your training program for the operators in preparation for implementing the ARTS/MELLLA operation at CGS. If applicable, please describe any additional changes to training or qualifications.

Energy Northwest Response:

Training impact for ARTS/MELLLA operation at CGS:

- Simulator training involving operations in the MELLLA domain will be conducted for Licensed Operators prior to operating in the MELLLA domain and is currently scheduled to occur in December 2014 after the simulator is updated to include the PRNM modifications. Operations in the MELLLA domain are not anticipated until the fall of 2015.

Training impact for installation of the PRNM System at CGS:

- Simulator training on the PRNM System will be conducted for Licensed Operators after the simulator is updated to include the PRNM modifications; this training will be completed prior to startup from the planned implementing outage (spring 2015). PRNM System hardware changes in the simulator are currently scheduled to be installed in the third quarter of calendar year 2014.

  Note: Since the scheduled on-going Initial Operator License Class (ILC) will be approaching their NRC exam date, the back panel components will be installed in a separate cabinet from the existing average power range monitoring (APRM) hardware. The front panel components will be installed on a roll-around cart, with software flags in simulation determining whether the legacy neutron monitoring system or the PRNM System is in operation. This will allow for continuance of ILC training on both systems. The final installation of PRNM components in their associated cabinets in the simulator will occur during the planned implementing outage (spring 2015).

NRC Request:

6. In many of the sections describing human factors changes, the submittal states that changes were made in accordance with CGS Design Specification 204 for Human Factors. Please provide a description of this Design Specification and its relevance to Human Factors. Also, please identify any regulatory precedents where this Design Specification was approved previously.

Energy Northwest Response:

Design Specification 204 is the Energy Northwest implementing procedure that ensures that modifications made at CGS address the appropriate Human Factors elements. The requirements delineated in this specification procedure:

- Ensure control room design reviews are performed in accordance with NUREG-0700, "Human-System Interface Design Review Guidelines"

- Reflect consideration of the NRC Action Plan developed as a result of the Three Mile Island (TMI)-2 accident (NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," and NUREG-0737, "Clarification of TMI Action Plan Requirements")

- Incorporate guidance of Regulatory Guide 1.97 "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants"

- Specify additional Energy Northwest Human Factors Engineering Standards

Energy Northwest is not requesting approval of this plant specific implementing procedure (Design Specification 204). This document is available for NRC audit and can be provided if desired.

NRC Request:

7. Please describe what, if any, operator actions are being changed, added or deleted.

Energy Northwest Response:

No Operator actions are being changed, added, or deleted as a result of the PRNM System modification or the change in the licensing basis for the number of required SLC pumps needed to mitigate an Anticipated Transient Without Scram (ATWS).

With the implementation of MELLLA, the Operators will be required to change the setpoints for the flow biased recirculation system inputs to the APRMs when transitioning from normal two-loop reactor recirculation operations to single-loop operations (SLO). This change will be captured in the transition to single loop procedure as described further below in the response to NRC Request 10. The setpoints will need to be changed because CGS is not utilizing the MELLLA limits while in SLO.

NRC Request:

8. Please describe any new task analyses results that were completed to identify functional requirements and to provide inputs for function allocation.

Energy Northwest Response:

Installation of a digital PRNM System replaces the existing APRM, RBM, and oscillation power range monitoring (OPRM) systems with updated equipment that perform the same function as the equipment being replaced. As identified in the response to NRC Request 7 above, the Operator actions remain essentially unchanged. Operations with the ARTS/MELLLA improvements also do not change the required Operator actions, with the exception of a procedurally controlled change in operating setpoints when transitioning to SLO.

As there have been no identified significant changes to Operator actions or functions, no new task analyses were performed.

NRC Request:

9. Please describe any changes to physical interfaces.

Energy Northwest Response:

In Reference 1, Enclosure 2, Attachment 1, section 2.3.4, Energy Northwest identified that the configuration option specified in the NRC-approved PRNM Licensing Topical Report (LTR) [NEDC-32410P-A, Volumes 1 & 2 and Supplement 1] for the Operator Panel interface matches that of LTR section 2.3.3.6.2.1.

A summary of the changes required to implement the PRNM replacement system can also be found in section 5.2 of the LTR.

For CGS, implementation of the PRNM System translates to the following changes of physical interfaces for the Control Room Operator:

- Replacing two existing APRM bypass switches on the main operator console with a single mechanical fiber optic bypass switch.

- Replacing the LPRM meters with four Operator Display Assembly (ODA) NUMAC units on the main operator console. An ODA is provided for each RBM (A and B). There is also an ODA dedicated for APRM 1 & 3 and a second ODA dedicated for APRM 2 & 4. APRM ODAs provide OPRM status information as well as conventional APRM and LPRM data.

- Removing two existing flow unit bypass switches on the main operator console. Switch functions have been moved to the fiber optic bypass switch and indication/status will be displayed at the ODAs.

- Removing eight existing intermediate range monitor (IRM)/APRM/RBM selector switches on the main operator console. Each IRM/APRM/RBM input will be wired to a dedicated channel on one of four recorders.

- Updating or deleting various annunciator window tiles, status lights, and computer points per the modification package.

- Installing a NUMAC PRNM System back-panel [containing Local Power Range Monitor (LPRM) chassis, APRM chassis, RBM chassis, RBM interface units, Two-out-of-Four Logic system (2-out-of-4 Voter) and power supply chassis], in the existing LPRM/APRM/Flow Unit cabinetry.

- Removing the existing OPRM interface computer from panel C91-P610.

NRC Request:

10. Please describe changes that will be required to the procedures for the new digital upgrade. Please provide a list of those changes.

Energy Northwest Response:

There are no new Operator actions required as a result of implementing the new PRNM System with the ARTS/MELLLA setpoints as regards emergency operating and abnormal operating procedures. The PRNM/ARTS/MELLLA changes will impact plant procedures as follows:

Reactor Recirculation procedures: Abnormal, Operating, and Surveillance procedure changes will include the new power to flow map (two loop operation) which reflects the MELLLA operating domain. The transition to recirculation system SLO will include requirements to reduce rod line as necessary to get below the Extended Load Line Limit Analysis (ELLLA) boundary by inserting control rods. This does not constitute a change in

Operator actions since the current transition to SLO already requires that rod line be reduced to avoid operation in the plant instability regions. A change in setpoints for the APRM system simulated thermal power trip will also be required since the plant will not be retaining the MELLLA setpoints while in SLO. This change in setpoints will be controlled via the SLO transition procedures and involves selecting new digital setpoints in the PRNM System for implementation. Establishing the new setpoints can be accomplished in just a few minutes and will not challenge the Technical Specification (TS) completion time requirement of 4 hours for TS 3.4.1 Condition B.

Alarm Response procedures are impacted by a reduction in the number of annunciators. This is because the APRMs were previously divided into two groups or divisions. The PRNM System has grouped the APRMs in one group, which is non-divisional. Therefore the same alarm (APRM TROUBLE, APRM UPSCALE OR INOP TRIP, OPRM TRIP, OPRM TRIP ENABLED, and OPRM INOP) is actuated from any of the four APRM channels or OPRM channels. The annunciator changes also affect other procedures which reference the annunciators. There are no new Operator actions required to support alarm responses.

Surveillance procedures will be updated to perform the TS required testing using to the new instruments, revised setpoints, and changes in frequency as applicable.

Administrative changes that are reflected in procedures include:

- Control of Plant Operating Keys - Removed keys that are no longer needed and added new keys related to bypassing the PRNM System.

- Administrative controls are being placed in the procedure for LPRM alarm response to reflect the revised LPRM requirements in TS Bases 3.3.1.1 including the minimum number of operable LPRMs per APRM, minimum number of operable LPRMs required per detector level, as well as the number of inputs that become inoperable since the last calibration.

NRC Request:

11. Please describe how CGS has applied insights from industry operating experience.

Energy Northwest Response:

Numerous industry issues related to digital systems and the PRNM System in particular were considered when reviewing this system for installation at CGS. The following are some of the relevant industry operating experience (OE) summaries and how CGS is applying the insights learned:

- SEN 271 - On November 28, 2007, Perry Nuclear Power Plant was at 100 percent power when failures in the redundant auctioneered power supplies for the digital feedwater control system resulted in an invalid high reactor water level trip signal. This caused the main turbine and both reactor feedwater pump turbines to trip and an automatic reactor scram. The digital feedwater control system primary power supply degraded under load for several months, but its status light indicated conditions were normal. As a result, plant personnel were unaware of the degradation. When the secondary power supply failed, the primary was unable to carry the load thereby initiating the reactor scram. The transient was caused by the loss of two power supplies. Both power supplies failed prematurely because of a manufacturing defect in a transformer on the 24 Volt DC output boards in each power supply. Both failed within three years of being installed.

CAUSE: The root cause of the scram was insufficient reliability of the digital feedwater control system power supplies. Both of the power supplies had premature failures because of a manufacturing defect with a subcomponent, the T2 transformer, on the 24 Volt DC output boards in each power supply. Testing indicated that the T2 transformers had high coil resistance. The failed power supplies were initially placed in service in May 2005. Two independent power supply failures caused the digital feedwater control system malfunction. The transient was caused by failures in the redundant auctioneered power supplies for the digital feedwater control system.

REVIEW/RESPONSE: The outputs of the redundant power supplies used in the new PRNM System are normally monitored by the associated chassis (APRM or RBM) for indications of failure or output drift that may exceed prescribed limits. Each unit is normally configured to work in parallel with another Low Voltage Power Supply (LVPS) module, but is designed to supply the required electrical load of both (paralleled) modules. A complete self-test, in the operate mode, is automatically performed once every few minutes. The loss of a single supply's output causes a self-test alarm. The loss of redundant outputs causes an INOP condition. Therefore, failure of redundant auctioneered power supplies will be very unlikely as the self-test feature will notify the Operator of any malfunction.

- OE11021 - PRNM System SLO Setpoint Deviation (Limerick): During modification acceptance testing of the recently installed GEH PRNM System (prior to placing the system into service), it was discovered that if the APRMs were in the single recirculation loop mode of operation, the APRMs would not calculate simulated thermal power scram setpoints in accordance with the TS equation.

  CAUSE: The cause of this problem was the internal programming of the APRM chassis. The PRNM System equation used to calculate the setpoint is as follows:

  RUN Setpoint = {Slope * (Flow – delta flow)} + Offset

  Where delta flow is the adjustment factor applied when in SLO

For Limerick, the normally entered parameters are Slope = .66, delta flow = 7.6 and Offset = 62.8. The problem was caused by programming logic which prohibits the term (Flow – delta flow) from going negative. If this term goes negative, the internal logic of the APRM sets the term to zero; thus, the lowest setpoint which can be calculated is the entered Offset value.

REVIEW/RESPONSE: This is software-related failure due to the PRNM System APRM programming which has been since addressed by GEH. The software has been modified to allow negative values for (flow-delta flow).

- OE14123 - PRNM System APRM Calibration Measurements Affected by Test Point Location: While performing PRNM System APRM calibration, as described in the APRM User's Manual Figure 3.4-3 and paragraph 3.4, the resistance of R65 for A5-CAL was found to be out of tolerance. Troubleshooting revealed that the resistance was satisfactory with power to the APRM turned off. Following discussion with GEH the resistance was re-measured at TP-60 and TP-61. This reading was also within tolerance. The error introduced during measurement at the BNC connector, A5-CAL, was found to be caused by a ground loop that exists due to the calibration jack being grounded.

  This ground loop does not affect the measurement when TP-60 and TP-61 are used or when the measurement is performed with the APRM de-energized. Initial analysis by GEH indicated the ground current to be approximately 1 micro-amp; however the magnitude may be system specific. The use of an ohmmeter that produces at least 1 milliamp of current will also prevent the problem for a system with a nominal 1 micro-amp of ground current.

  CAUSE: The APRM BNC calibration jacks, A5-CAL and A11-CAL are grounded resulting in a ground current which can affect the R65 measurement.

  REVIEW/RESPONSE: The design is modified to ensure that calibration jacks (A5-CAL and A11-CAL) used for testing are not grounded so that a ground loop doesn't affect measurements. CGS has verified that test points on the PRNM System are isolated from ground current.

- OE10132 - GEH PRNM System Relay Logic Card Module Failures (Browns Ferry): A PRNM System relay logic card module failed on Unit 3, resulting in a spurious rod block condition. Troubleshooting revealed that reactor manual control system (RMCS) relay K1 had failed in a shorted condition. The relay short, in conjunction with branch circuit fusing that was not appropriately sized, caused the relay logic card module to fail.

  CAUSE: The PRNM System relay logic card modules failed due to an over-current condition. The over-current condition occurred because of improperly sized branch circuit fusing. Engineering personnel were not made aware of a change in the contact rating capacity when using relay logic card modules to replace existing electro-mechanical relays. Consequently, relay logic card module over-current protection was not considered in the design change which installed the PRNM System modifications.

REVIEW/RESPONSE: GEH conducted an investigation of the event and determined that this appears to be an isolated event. GEH continues to monitor the performance of this component for an adverse trend. In addition, the RMCS design for CGS is not similar to the one discussed in this OE.

- OE4086 - LPRM Cable Reversal: LPRM detectors were found reading inconsistently when compared to other LPRMs in the same general area. Further investigation revealed possible LPRM configuration problems.

  CAUSE: Subsequent LPRM configuration verification testing by monitoring core flux changes with rod movements in the vicinity of the LPRMs substantiated that eight reversals in the LPRM configuration existed for three cycles and went unnoticed.

  REVIEW/RESPONSE: The installation work order will contain instructions for ensuring cables are connected to the correct connection. Each cable connection will contain a signoff for the performer and another for the verifier (Independent Verification). In addition, Reactor Engineering will confirm proper LPRM response to control rod movements.

- OE20368 - OPRM Trip Indication in Single Recirculation Loop Operation (Peach Bottom): With Unit 3 in SLO, Control Room Operators identified that the OPRM system had experienced a trip. There were no signs of thermal hydraulic instability and no reactor protection system (RPS) actuations took place since the trip functions of the OPRM system had not yet been officially activated. The spurious trip was attributed to overly conservative interim trip setpoints and elevated neutron noise while in SLO.

  CAUSE: The cause of the spurious OPRM trip was a combination of elevated, coherent background flux noise that was present in single loop conditions, coupled with excessively conservative interim trip setpoints that were nominally 85% lower than those analytically determined for Peach Bottom.

  REVIEW/RESPONSE: CGS is not impacted by this OE since the setpoint at CGS is higher than the 1.07 setpoint at Peach Bottom. CGS is increasing the setpoint to 1.15 to improve margin to trip and imposing a SLO administrative limit of 100% rod line to reduce noise

- OE23468 - Susquehanna Unit 1 Automatic Reactor Scram upon Routine Transfer of RPS Power Supply from Normal to Alternate: At 0300 hr on June 15, 2006, Control Room Operators were transferring "B" RPS power from normal to alternate source to prepare for maintenance, upon which the unit experienced a scram. All control rods fully inserted and the reactor system responded to the scram as expected.

CAUSE: The cause of the automatic reactor scram is associated with the Reactor Mode switch interface with the recently-installed GEH PRNM System.

REVIEW/RESPONSE: This issue does not impact CGS since the equivalent circuit is wired directly to the mode switch contacts instead of having interposing relay contacts.

- OE23560 - Non-conservative BWR OPRM Core Flow Setpoint (Perry): Based on information contained in GEH Safety Communication (SC) 06-10, Perry declared the OPRM system inoperable. The enabled region total drive flow setpoint was set low enough that following a trip to SLO from high core flows, the OPRM System may not have been enabled with core flow less than 60% rated core flow. The required stability protection may not have been provided.

  CAUSE: As identified in the SC, the setpoint methodology does not specifically identify how this setpoint is determined nor does it specify what conditions should be considered.

  REVIEW/RESPONSE: This is not applicable to CGS. GEH report 0000-0107-2061-R0, Task T0202, "Thermal-Hydraulic Stability," documents that SC 06-10 does not impact the PRNM System OPRM. SC 06-10 identifies a restriction in the OPRM enable setpoint. GEH recommends setting it no lower than 56% +/- 3% rated drive flow. The applicable CGS calculation requires consideration of SC 06-10 if the OPRM flow enable setpoint is ever reduced below 60%.

- OE24238 - OPRM Trip While in Single Recirculation Loop Operation (Brunswick): On December 25, 2006, while Unit 2 was at 64% reactor power in SLO, an automatic scram occurred. Safety related equipment operated as designed. The scram was due to an actuation of two channels of the OPRM from the growth rate based algorithm. At the time of the event, the reactor was being operated in the OPRM enabled region of the SLO Power/Flow map. This event shows the uncertainties associated with extended SLO at higher power levels.

  CAUSES: The OPRM trips were not due to thermal hydraulic instabilities event but from random APRM reactor noise that fell within the frequency range of the thermal hydraulic instabilities growth-rate based algorithm for the OPRM trip signal, and due to an actuation of two channels of the OPRM from the growth rate based algorithm.

  REVIEW/RESPONSE: Operation in SLO has been shown to introduce various thermal hydraulic effects, which may result in increased fluctuations in jet pump flows, reactor water level readings and APRM noise bandwidth. The GEH study of the Brunswick thermal hydraulic instabilities event showed that the growth rate algorithm scram was caused by noise induced oscillations on APRM flux. GEH was able to model and reproduce the event in the model from noise introduced due to core flow. CGS has successfully operated in SLO at the rated rod line with a 1.10 setpoint. CGS is increasing the setpoint to 1.15 to improve margin to trip and imposing a SLO administrative limit of 100% rod line to reduce noise.

- OE30500 - Digital EHC Engineering Work Station Software Infected By Computer Virus - (Seabrook Station): On November 13, 2009, an Information Security Analyst contacted one of the Digital EHC project team members because the analyst was receiving automated alerts that two viruses had been detected on both a laptop and desktop computer. Investigation revealed that a portable external storage device (thumb drive) had been used on both the laptop and desktop computer. Further investigation found that the infected external storage device had also been inserted into the Digital EHC turbine control System Engineering workstation. The workstation is a device within the Level 4 network as defined by NEI 04-04, "Cyber Security Program for Power Reactors." Although the workstation had antivirus software installed, the definitions were not up to date and the workstation was subsequently infected by the virus. Since this Level 4 critical digital asset was potentially compromised, the cyber security incident handling and response procedure was initiated and executed around the clock for over 48 hours. The process quantified the impact and defined appropriate actions to both eliminate and minimize impact of the virus. The virus was subsequently removed from the workstation.

  CAUSES: An infected portable external storage device infected the workstation.

  REVIEW/RESPONSE: The current configuration for the Level 4 equipment at CGS does not permit the use of portable storage drives. Therefore, the system will not allow infected data to transfer onto the system.

References:

1. Letter, GO2-12-017, dated January 31, 2012, BJ Sawatzke (Energy Northwest) to NRC, "License Amendment Request to Change Technical Specifications in Support of PRNM / ARTS / MELLLA Implementation"

List of Commitments

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

Commitment	Scheduled Completion Date
Conduct simulator training for all Licensed Operators for operations in the MELLLA domain.	Prior to operating in the MELLLA domain
Conduct simulator training for all Licensed Operators on the PRNM System.	Prior to startup from refueling outage 22
Perform Independent Verification of LPRM cable connections to the PRNM System components. Work instructions will ensure each LPRM cable connection will contain a signoff for the performer and another for the verifier.	Prior to startup from refueling outage 22