LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM / ARTS / MELLLA IMPLEMENTATION

Attachment 1

Columbia Generating Station Plant-Specific Responses
Required by NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A)

GE Hitachi Nuclear Energy
3901 Castle Hayne Road
Wilmington, NC 28401

0000-0101-7647-R2
GEH DRF 0000-0098-4294
Class III
April 2010

Columbia Generating Station

Plant-Specific Responses Required By NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report (NEDC-32410P-A)

Prepared by: S. Rudy

Verified by: S. Gupta

Approved by: E. Schrull

Table of Contents

Columbia Specific Responses Required by NUMAC PRNM Retrofit Topical Report	**1-19**
Appendix A, Columbia NUMAC PRNM LTR Deviations	**A1-A5**

0000-0101-7647-R2
Columbia Specific Responses Required by NUMAC PRNM Retrofit Topical Report

The section numbers and Utility Actions Required listed below are from the NUMAC PRNM Retrofit Plus Option III Stability Trip Function Topical Report NEDC-32410P-A including Supplement 1.

Section No.	Utility Action Required	Response
2.3.2	<u>Option III Stability Implementation</u> Not a required specific LTR response Confirm that the actual plant Option III configuration is included in the variations covered in the Power Range Neutron Monitor (PRNM) Licensing Topical Report (LTR) [NEDC-32410P-A, Volumes 1 & 2 and Supplement 1].	The CGS Option III implementation is in accordance with the LTR Requirements of section 2.3.2.
2.3.4	<u>Plant Unique or Plant-Specific Aspects</u> Confirm that the actual plant configuration is included in the variations covered in the Power Range Neutron Monitor (PRNM) Licensing Topical Report (LTR) [NEDC-32410P-A, Volumes 1 & 2 and Supplement 1], and the configuration alternative(s) being applied for the replacement PRNM are covered by the PRNM LTR. Document in the *plant-specific licensing submittal* for the PRNM project the actual, current plant configuration of the replacement PRNM, and document confirmation that the PRNM LTR covers those. For any changes to the plant operator's panel, document in the submittal the human factors review actions that were taken to confirm compatibility with existing plant commitments and procedures.	The actual, current plant configuration and the proposed replacement PRNM are included in the PRNM LTR as follows: (Applicable LTR sections are listed.) — <u>Current</u> — <u>Proposed</u> APRM — 2.3.3.1.1.2 — 2.3.3.1.2.1 RBM — 2.3.3.2.1.1 — 2.3.3.2.2.1 Flow Unit — 2.3.3.3.1.2 — 2.3.3.3.2.2 Rod Control — 2.3.3.4.1.2 — 2.3.3.4.2.2 Panel Interface — 2.3.3.6.1.1 — 2.3.3.6.2.1 Human Factors Engineering is part of the CGS design process. The actual PRNMS System to be installed at CGS contains 1 deviation from the system design as described in the LTR. Justification for these deviations is provided as Appendix A.
3.4	<u>System Functions</u> As part of the *plant-specific licensing submittal*, the utility should document the following: 1) The pre-modification flow channel configuration, and any changes planned (normally changes will be either adding two channels to reach four or no change planned) NOTE: If transmitters are added, the requirements on the added transmitters should be:	1) The current flow channel configuration consists eight flow transmitters (LTR Section 3.2.3.1.1). Thus, the current configuration meets the requirements described in LTR Section 3.2.3.2.2.

Section No.	Utility Action Required	Response
	• Non-safety related, but qualified environmentally and seismically to operate in the application environment. • Mounted with structures equivalent or better than those for the currently installed channels. • Cabling routed to achieve separation to the extent feasible using existing cableways and routes.	
	2) Document the APRM trips currently applied at the plant. If different from those documented in the PRNM LTR, document plans to change to those in the LTR.	2) APRM trips currently applied at the plant are listed below along with changes planned. The "post-modification" trips will be the same as those identified in the LTR. • "Inop" Retained, except the logic is modified slightly (same as described in LTR paragraph 3.2.10). • "Fixed Neutron Flux-High" is modified to "Neutron Flux-High" (as described in LTR paragraph 3.2.5). • "Flow Biased Simulated Thermal Power-High" is modified to Simulated Thermal Power-High" (as described in LTR paragraph 3.2.5). • "APRM Neutron Flux – High (Setdown) is retained as described in LTR paragraphs 3.2.4 and 8.3.1.4. • Add 2-Out-of-4 Voter as described in LTR paragraphs 3.2.2 and 8.3.2.4.
	3) Document the current status related to ARTS and the planned post modification status as: • ARTS currently implemented, and retained in the PRNM • ARTS will be implemented concurrently with the PRNM (reference ARTS submittal) • ARTS not implemented and will not be implemented with the PRNM • ARTS not applicable	3) ARTS will be implemented concurrently with the PRNM.
4.4.1.11	Regulatory Requirements of the Replacement System – System Design This section identifies requirements that are expected to encompass most specific plant commitments relative to the PRNM replacement project, but may not be complete and some may not apply to all plants. Therefore, the utility must confirm that the requirements identified	A review of the CGS requirements confirms that the regulatory requirements addressed in the LTR encompass the related CGS requirements. The design change process will confirm that the regulatory and licensing requirements are met. CGS commitments will be reviewed against the LTR. Upon initial review, the LTR meets CGS commitments.

Section No.	Utility Action Required	Response
	here address all of those identified in the plant commitments. The plant-specific licensing submittal should identify the specific requirements applicable for the plant, confirm that any clarifications included here apply to the plant, and document the specific requirements that the replacement PRNM is intended to meet for the plant.	
4.4.2.2.1.4	Regulatory Requirements for the Replacement System -Equipment Qualification - Temperature and Humidity Plant-specific action will confirm that the maximum control room temperatures plus mounting panel temperature rise, allowing for heat load of the PRNM equipment, does not exceed the temperatures presented in the PRNM LTR, and that control room humidity is maintained within the limits stated in the PRNM LTR. This evaluation will normally be accomplished by determining the operating temperature of the current equipment which will be used as a bounding value because the heat load of the replacement system is less than the current system while the panel structure, and thus cooling, remains essentially the same. Documentation of the above action, including the specific method used for the required confirmation should be included in *plant-specific licensing submittals.*	The PRNM control room electronics are qualified for continuous operation under the following temperature conditions: 5 to 50 °C [41 to 122 °F]. The CGS control room temperature range is: 40 - 104 °F (72-78 °F normal). The design process includes actions to confirm that the PRNM equipment, as installed in the plant, is qualified for the environmental limits, including temperature rise measurements. The PRNM control room electronics are qualified for continuous operation under the following relative humidity conditions: 10 to 90% (non-condensing). The CGS relative humidity requirement for control room equipment is: 10 - 60%, which is within the range for which the PRNM equipment is qualified. The qualification results will be documented in a plant unique "Qualification Summary."
4.4.2.2.2.4	Regulatory Requirements for the Replacement System -Equipment Qualification - Pressure Plant-specific action will confirm that the maximum control room pressure does not exceed the limits presented in the PRNM LTR. Any pressure differential from inside to outside the mounting panel assumed to be negligible since the panels are not sealed and there is no forced cooling or ventilation. Documentation of this action and the required confirmation should be included in *plant-specific licensing submittals.*	The PRNM control room electronics are qualified for continuous operation under the following pressure conditions: 13 - 16 psia. The CGS normal ambient atmospheric pressure is approximately 14.43 psia (nominally 14.0-15.0 psia), and is within these limits. The qualification results will be documented in a plant unique "Qualification Summary."
4.4.2.2.3.4	Regulatory Requirements for the Replacement System -Equipment Qualification -Radiation Plant-specific action will confirm that the maximum control room radiation levels do not exceed the limits presented in the PRNM LTR. Documentation of this action and the required confirmation should be included in *plant-specific licensing submittals.*	The PRNM control room electronics are qualified for continuous operation under the following conditions: Dose Rate $\leq$ 0.001 Rads (carbon)/hr and Total Integrated Dose (TID) $\leq$ 1000 Rads (carbon). The CGS control room dose rates (350 Rads (Carbon, gamma) over 40 years) and TID are within the qualified ranges. The qualification results will be documented in a plant unique "Qualification Summary."

Section No.	Utility Action Required	Response
4.4.2.3.4	Regulatory Requirements for the Replacement System -Seismic Qualification Plant-specific action or analysis will confirm that the maximum seismic accelerations at the mounting locations of the equipment (control room floor acceleration plus panel amplification) for both OBE and SSE spectrums do not exceed the limits stated in the PRNM LTR. Documentation of this action and the required confirmation should be included in *plant-specific licensing submittals.*	Evaluations to confirm that the maximum seismic accelerations at the mounting locations of the equipment do not exceed qualification limits of the equipment are completed as part of the CGS normal design change process. The seismic qualification results will be documented in "Qualification Summary
4.4.2.4.4	Regulatory Requirements for the Replacement System -EMI Qualification The utility should establish or document practices to control emission sources, maintain good grounding practices and maintain equipment and cable separation. 1) Controlling Emissions a) Portable Transceivers (walkie-talkies): Establish practices to prevent operation of portable transceivers in close proximity of equipment sensitive to such emissions. (NOTE: The qualification levels used for the NUMAC PRNM exceed those expected to result from portable transceivers, even if such transceivers are operated immediately adjacent to the NUMAC equipment.) b) ARC Welding: Establish practices to assure that ARC welding activities do not occur in the vicinity of equipment sensitive to such emissions, particularly during times when the potentially sensitive equipment is required to be operational for plant safety. (NOTE: The qualification levels used for NUMAC PRNM minimize the likelihood of detrimental effects due to ARC welding as long as reasonable ARC welding control and shielding practices are used.) c) Limit Emissions from New Equipment: Establish practices for new equipment and plant modifications to assure that they either do not produce unacceptable levels of emissions, or installation shielding, filters, grounding or other	1.) Controlling Emissions a) The qualification levels used for the NUMAC PRNM system exceed those expected to result from portable transceivers, even if such transceivers are operated immediately adjacent to NUMAC equipment. CGS generally prohibits operation of portable transceivers near sensitive equipment, and if warranted, requires positioning of warning signs at critical locations throughout the plant. Placement of warning signs will be evaluated as part of the modification process. b) The qualification levels used for the NUMAC PRNM system minimize the likelihood of detrimental effects due to ARC welding as long as reasonable ARC welding control and shielding practices are used. ARC welding is only performed at CGS with specific work orders and directions, and is known to have the potential to affect operation of I&C equipment at a number of locations in the plant. Therefore, ARC welding activity is only performed when any potential effect on I&C equipment is tolerable relative to plant operation. c) EMI emissions from new equipment installed at CGS are evaluated as part of the normal design modification process described in CGS procedures.

Section No.	Utility Action Required	Response
	methods prevent such emissions from reaching other potentially sensitive equipment. These practices should address both radiated emissions and conducted emissions, particularly conducted emissions on power lines and power distribution systems. Related to power distribution, both the effects of new equipment injecting noise on the power system and the power system conducting noise to the connected equipment should be addressed. (NOTE: For the qualification of the PRNM equipment includes emissions testing.)	
	2) Grounding Practices Existing Grounding System: The specific details and effectiveness of the original grounding system in BWRs varied significantly. As part of the modification process, identify any known or likely problem areas based on previous experience and include in the modification program either an evaluation step to determine if problems actually exist, or include corrective action as part of the modification. (NOTE: The PRNM equipment is being installed in place of existing PRM electronics which is generally more sensitive to EMI than the NUMAC equipment. As long as the plant has experienced no significant problems with the PRM, no problems are anticipated with the PRNM provided grounding is done in a comparable manner.) Grounding Practices for New Modifications: New plant modifications process should include a specific evaluation of grounding methods to be used to assure both that the new equipment is installed in a way equivalent to the conditions used in the qualification. (NOTE: NUMAC PRNM equipment qualification is performed in a panel assembly comparable to that used in the plant.)	2) The PRNM system equipment is being installed in place of existing Power Range Monitor (PRM) system electronics. The replacement system will interface with the same cables and wiring at the panel interfaces as the current system, including ground bus connections. No problems have been identified with the current PRM system related to grounding or grounding practices. The original installation included specific grounding practices designed to minimize performance problems. The replacement PRNM system is less sensitive to grounding issues than is the current system and includes specific actions in the wiring inside the panel to maximize shielding and grounding effectiveness.
	3) Equipment and Cable Separation a. Cabling: Establish cabling practices to assure that signal cables with the potential to be "receivers" are kept	3) Equipment and Cable Separation The original PRM system cable installation requirements met this objective. The replacement PRNM system uses the same

Section No.	Utility Action Required	Response
	separate from cables that are sources of noise. (NOTE: The original PRM cable installation requirements met this objective. The replacement PRNM uses the same cable routes and paths, so unless some specific problem has been identified in the current system, no special action should be necessary for the PRNM modification.) b. Equipment: Establish equipment separation and shielding practices for the installation of new equipment to simulate that equipment's qualification condition, both relative to susceptibility and emissions. (NOTE: The original PRM cabinet design met this objective. The replacement PRNM uses the same mounting cabinet, and used an equivalent mounting assembly for qualification. No special action should be necessary for the PRNM modification.) The *plant-specific licensing submittals* should identify the practices that are in place or will be applied for the PRNM modification to address each of the above items.	cable routes and paths at comparable energy levels where feasible. Because no specific problem has been identified in the current system, no special action is necessary for the PRNM modification. The existing system cabling complies with applicable CGS cable routing and separation requirements. Additionally, the modification process is performed in accordance with the existing separation criteria.
6.6	System Failure Analysis The utility must confirm applicability of the failure analysis conclusions contained in the PRNM LTR by the following actions: 1. Confirm that the events defined in EPRI Report No. NP-2230 or in Appendices F and G of Reference 11 of the PRNM LTR, encompass the events that are analyzed for the plant; 2. Confirm that the configuration implemented by the plant is within the limits described in the LTR; and	1. The CGS Technical Specification (TS) Surveillance Requirements for the Reactor Protection System (RPS) are based on Reference 11 of the PRNM LTR as discussed in the CGS Technical Specification Bases (Section 3.3.1.1, Reactor Protection System Instrumentation). Therefore, the Reference 11 failure analysis is applicable to CGS. The overall redundancy and diversity of sensors available to provide trip signals in the RPS meets NRC-approved licensing basis requirements. 2. The proposed PRNM configuration is included among the configurations described in the PRNM LTR, as itemized under Section 2.3.4 above. The proposed configuration is being designed by GEH and is within the limits described in the LTR.

Section No.	Utility Action Required	Response
	3. Prepare a plant-specific 10CFR50.59 evaluation of the modification per the applicable plant procedures. These confirmations and conclusions should be documented in the *plant-specific licensing submittals* for the PRNM modification. [Reference 11 of the LTR is NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System", Licensing Topical Report, GE Nuclear Energy, Class III (proprietary), dated March 1988.]	3. The requirements of 10CFR50.59 will be applied to the PRNMS modification in accordance with applicable plant procedures.
7.6	<u>Impact on UFSAR</u> The plant-specific action required for FSAR updates will vary between plants. In all cases, however, existing FSAR documents should be reviewed to identify areas that have descriptions specific to the current PRNM using the general guidance of Sections 7.2 through 7.5 of the PRNM LTR to identify potential areas impacted. The utility should include in the *plant-specific licensing submittal* a statement of the plans for updating the plant FSAR for the PRNM project.	Applicable sections of the FSAR are reviewed and appropriate revisions of those sections are prepared and approved as part of the normal design process. Following implementation of the design modification and closure of the design package, the FSAR revisions are included in the updated FSAR as part of the periodic 10 CFR 50.71(e) FSAR update submittal.
8.3.1.4	<u>APRM-Related RPS Trip Functions - Functions Covered by Technical Specifications</u> 1. Delete the APRM Downscale function, if currently used, from the RPS Instrumentation "function" table, the related surveillance requirements, and, if applicable, the related setpoint, and related descriptions in the bases sections. 2. Delete the APRM Flow-biased Neutron Flux Upscale function, if currently used, from the RPS Instrumentation "function" table, the related surveillance requirements, and, if applicable, the related setpoint, and related descriptions in the bases sections. Replace these with the corresponding entries for the APRM Simulated Thermal Power - High and the APRM Neutron Flux - High functions. Perform analysis necessary to establish setpoints for added trips. 3. Add the APRM Neutron Flux - High (Setdown) function, if not currently used, to the RPS Instrumentation "function" table, add the related surveillance requirements, and, if applicable, the related setpoints, and related descriptions in the bases sections.	1. CGS does not have an "APRM Downscale" RPS Trip Function Technical Specifications. 2. CGS currently uses a "Flow Biased Simulated Thermal Power-High," RPS trip function. This function is renamed to "Simulated Thermal Power-High". The "Fixed Neutron Flux-High" is retained as "Neutron Flux-High". 3. The current APRM Neutron Flux - High (Setdown) function is retained

Section No.	Utility Action Required	Response
	Perform analysis necessary to establish setpoints for added trips.	
8.3.2.4	APRM-Related RPS Trip Functions - Minimum Number of Operable APRM Channels 1. For the 4-APRM channel replacement configuration, revise the RPS Instrumentation "function" table to show 3 APRM channels, shared by both trip systems for each APRM function shown (after any additions or deletions per PRNM LTR Paragraph 8.3.1.4). Add a "2-out-of-4 Voter" function with two channels under the "minimum operable channels". For plants with Technical Specifications that include a footnote calling for removing shorting links, remove the references to the footnote related to APRM (retain references for SRM and IRM) and delete any references to APRM channels in the footnote. For smaller core plants, delete the notes for and references to special conditions related to loss of all LPRMs from the "other" APRM. 2. Review action statements to see if changes are required. If the improvements documented in Reference 11 have not been implemented, then changes will likely be required to implement the 12-hour and 6-hour operation times discussed above for fewer than the minimum required channels. If Improved Technical Specifications (IST) are applied to the plant, action statements remain unchanged. 3. Revise the Bases section as needed to replace the descriptions of the current 6- or 8-APRM channel systems and bypass capability with a corresponding description of the 4-APRM system, 2-out-of-4 Voter channels (2 per RPS system), and allowed one APRM bypass total.	1. The PRNM modification and the proposed Technical Specifications and Bases change implement the changes as described in the PRNM LTR for a "larger core" plant. CGS Technical Specifications do not include notes related to APRMs that call for removal of shorting links or references to special conditions related to loss of all LPRMs from the "other" APRM. Therefore, no related note changes are required. Revised RPS functions to indicate that the required number channels per a trip system is 3 APRM channels. "2-out-of-4 Voter" function with two channels under the "minimum operable channels" has been added as Function 2.e. 2. Action statement changes in the proposed Technical Specifications change are consistent with the PRNM LTR described changes for plants with Improved Technical Specifications. CGS had previously switched to the IST format. 3. The proposed Technical Specifications Bases changes include revisions to the descriptions of the architecture, consistent with the PRNM LTR.
8.3.3.4	APRM-Related RPS Trip Functions - Applicable Modes of Operation 1) APRM Neutron Flux - High (Setdown) Change Technical Specifications "applicable modes" entry, if required, to be Mode 2 (startup). Delete references to actions and surveillance requirements associated with other modes. Delete any references to notes associated with "non-coincidence" mode and correct notes as required. Revise Bases descriptions as required.	1) Technical Specifications and Bases changes are consistent with the PRNM LTR.

Section No.	Utility Action Required	Response
	2) APRM Simulated Thermal Power - High Retain as is unless this function is being added to replace the APRM Flow-biased Neutron Flux Trip. In that case, add requirement for operation in Mode 1 (RUN) and add or modify Bases descriptions as required.	2) The Flow Biased Simulated Thermal Power - High Technical Specification is retained however, the name is changed to "APRM Simulated Thermal Power – High," consistent with the PRNM LTR.
	3) APRM Neutron Flux - High Retain as is unless this function is being added to replace the APRM Flow-biased Neutron Flux Trip. In that case, add requirement for operation in Mode 1 (RUN) and add or modify Bases descriptions as required.	3) The Fixed Neutron Flux - High Technical Specification and Bases is retained however, the name is changed to "APRM Neutron Flux – High," consistent with the PRNM LTR.
	4) APRM Inop Trip Delete any requirements for operation in modes other than Mode 1 and Mode 2 (RUN and STARTUP). Revise the Bases descriptions as needed.	4) The current CGS Technical Specifications require this Function, only in Modes 1 and 2.
8.3.4.1.4	APRM-Related RPS Trip Functions - Channel Checks/ Instrument Checks a) For plants without Channel Check requirements, add once per 12 hour or once per day Channel Check or Instrument Check requirement for the three APRM flux based functions. No Channel Check requirements are added for APRM Inop function. Plants with once per 12 hour or once per shift requirements may change them to once per day.	 a) The CGS Technical Specifications currently include a once-per-shift Channel Check requirement for the APRM Functions (except for Inop). The APRM Function Channel Check requirements are maintained at once per 12 hours, consistent with the remainder of the RPS Technical Specifications. The proposed Technical Specification and Bases changes for the Channel Check SR are consistent with the PRNM LTR.
	b) For plants with 4 full recirculation flow channels and with Technical Specifications that call for daily or other channel check requirements for flow comparisons under APRM Flow Biased Simulated Thermal Power Trip, delete those requirements. Move any note reference related to verification of flow signals to Channel Functional Test entry.	b) CGS currently has 8 flow transmitters. Associated surveillances have been included in those for the APRM Simulated Thermal Power – High, and the OPRM Upscale Functions (the latter because of the OPRM trip enable function). The proposed Technical Specification and Bases changes for the recirculation flow related SRs are consistent with the PRNM LTR but with some expansion to clarify that the recirculation flow functions also support the OPRM Upscale Function trip enable.

Section No.	Utility Action Required	Response
8.3.4.2.4	<u>APRM-Related RPS Trip Functions - Channel Functional Tests</u> a) Delete existing channel functional test requirements and replace with a requirement for a Channel Functional Test frequency of each 184 days (6 months) [delete any specific requirement related to startup or shutdown except for the APRM Neutron Flux - High (Setdown) function as noted in Paragraph 8.3.4.2.2(1) of the PRNM LTR. Add a notation that both the APRM channels and the 2-out-of-4 Voter channels are to be included in the Channel Functional Test. b) Add a notation for the APRM Simulated Thermal Power - High function that the test shall include the recirculation flow input processing, excluding the flow transmitters. CAUTION: Plants that have not implemented the APRM surveillance improvements of Reference 11 of the PRNM LTR, or those that have continued to use a weekly surveillance of scram contactors, may need to implement or modify surveillance actions to continue to provide a once per week functional test of scram contactors. (Prior to changes defined in Reference 11, the weekly APRM functional test also provides a weekly test of all automatic scram contactors.)	a) The proposed Technical Specification and Bases changes related to Channel Functional Tests are consistent with the PRNM LTR. b) The proposed Technical Specification and Bases changes to Channel Functional Tests for the APRM Functions include a notation, applicable to the Simulated Thermal Power – High (Function 2.b) and the OPRM Upscale (Function 2.f), consistent with the PRNM LTR requirements, that the SR includes the recirculation flow input processing, excluding the flow transmitters. However, the PRNM LTR includes this notation only in the Bases. For the CGS Technical Specification, the Channel Functional Test has been added as SR 3.3.1.1.16 and has been expanded from that in the LTR to also apply to the OPRM Upscale Function (to cover OPRM Upscale trip enable). The functional test procedure will be established to test all of the hardware required to produce the trip functions, but not to directly re-test software-only (firmware-only) logic. The APRM automatic self-test function monitors the integrity of the EPROMs storing all of the firmware so that if a hardware fault results in a "change" to the firmware (software), that fault will be detected by the self-test logic. The continued operation of the self-test procedures is monitored by the built-in "watch-dog timer" function, so if for some unforeseen reason the self-test function (lowest priority in the instrument logic) stops running, that failure also will be detected automatically. To provide further assurance that the self-test function continues to operate, a step will be included in the APRM Channel Check surveillance to confirm that self-test is still running.

Section No.	Utility Action Required	Response
8.3.4.3.4	<u>APRM-Related RPS Trip Functions - Channel Calibrations</u> a) Replace current calibration interval with either 18 or 24 months except for APRM Inop. Retain Inop requirement as is (i.e., no requirement for calibration). b) Delete any requirement for flow calibration and calibration of the 6 second time constant separate from overall calibration of the APRM Simulated Thermal Power Upscale Trip. c) Replace every 3 day frequency for calibration of APRM power against thermal power with a 7 day frequency if applicable. d) Revise Bases text as required.	a) The proposed Technical Specification and Bases changes related to Channel Calibration for the APRM Functions include a 24-month interval, with no calibration required for the Inop Function, consistent with the PRNM LTR. b) Consistent with the PRNM LTR requirements, the proposed Technical Specification and Bases changes add a notation applicable to the Channel Calibration for the APRM Simulated Thermal Power – High and OPRM Upscale Functions to include requirements for calibration of the recirculation flow transmitter and flow processing function. However, the PRNM LTR includes this notation only in the Bases. For the CGS Technical Specification, the notation has been included in the Channel Calibration SR (Table 3.3.1.1-1), and has been expanded from that in the LTR to also apply to the OPRM Upscale Function (to cover OPRM Upscale trip enable). c) The current CGS Technical Specifications include a "weekly" frequency for the verification of APRM power versus calculated plant thermal power so no change in that frequency is required to be consistent with the PRNM LTR. d) The proposed Technical Specification Bases changes related to Channel Calibrations are consistent with the PRNM LTR.
8.3.4.4.4	<u>APRM-Related RPS Trip Functions - Response Time Testing</u> Delete response time testing requirement from Technical Specifications or plant procedures, as applicable, for the APRM functions. Replace it with a response time testing requirement for the 2-out-of-4 Voter "pseudo" function, to include the output solid-state relays of the voter channel through the final RPS trip channel contactors. Frequency of response time testing shall be determined using four 2-out-of-4 Voter channels, but tests may alternate use of 2-out-of-4 Voter outputs provided each APRM/RPS interfacing relay is tested at least once per eight refueling cycles (based on a maximum 24 month cycle), and each RPS scram contactor is tested at	The proposed Technical Specification and Bases changes related to Response Time Testing (3.3.1.1.15 and Table 3.3.1.1-1) are consistent with the justification in the PRNM LTR Supplement 1. Consistent with the PRNM LTRs, the only APRM Function to which the SR will apply is Function 2.e (voter). However, while the PRNM LTRs justified reduced response time testing frequency for Function 2.e, no technical specification markups were included to implement an "n" greater than 4 (the total number of voter channels). Therefore, a note is added to the CGS SR 3.3.1.1.15 to define that "n=8" for Function 2.e. The PRNM LTR Supplement 1 justified response time testing at a rate that tested one

Section No.	Utility Action Required	Response
	least once per four refueling cycles. Each 2-out-of-4 Voter output shall be tested at no less than half the frequency of the tests of the APRM/RPS interface relays. Tests shall alternate such that one logic train for each RPS trip system is tested every two cycles.	RPS Interface relay every plant operating cycle, with tests using the APRM output for one cycle and the OPRM output for the next cycle. This yields a testing rate once per 8 operating cycles for each RPS interface relay and once per every 16 operating cycles for the APRM or OPRM output. The PRNM modification includes redundant APRM trip and redundant OPRM trip outputs from each 2-Out-Of-4 Voter channel. One of the OPRM outputs and one of the APRM outputs are connected in series to the coil of one RPS interface relay. The second OPRM output and the second APRM output from the 2-Out-Of-4 Voter channel are connected in series with the coil to a second RPS interface relay. There are 8 total RPS interface relays.
8.3.5.4	<u>APRM-Related RPS Trip Functions - Logic System Functional Testing (LSFT)</u> Revise Technical Specifications to change the interval for LSFT from 18 months to 24 months unless the utility elects to retain the 18-month interval for plant scheduling purposes. Delete any LSFT requirements associated with the APRM channels and move it to the 2-out-of-4 Voter channel. Include testing of the 2-out-of-4 voting logic and any existing LSFTs covering RPS relays.	The CGS Technical Specifications include a SR for LSFTs for the APRM related functions. These will be deleted, except for the new 2-Out-Of-4 Voter, Function 2.e, the LSFT will be added. The LSFT requirement for that Function is at a 24-month interval.
8.3.6.1	<u>APRM-Related RPS Trip Functions - Setpoints</u> Add to or delete from the appropriate document any changed RPS setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Technical Specifications submittal information with the PRNM information in the plant-specific submittal, or reference the ARTS submittal in the PRNM submittal. In the *plant-specific licensing submittal,* identify what changes, if any, are being implemented and identify the basis or method used for the calculation of setpoints and where the setpoint information or changes will be recorded.	ARTS will be applicable at CGS. CGS will perform setpoint calculations for the ARTS submittal. The results of the ARTS calculations will be used for the PRNM modification. PRNM setpoints and Allowable Values are re-calculated or confirmed using approved setpoint methodology. The Allowable Values for the APRM RPS Functions are included in the Technical Specifications or the COLR, comparable to what is currently in the CGS Technical Specifications and consistent with the PRNM LTR.
8.4.1.4	<u>OPRM-Related RPS Trip Functions - Functions Covered by Technical Specifications</u> Add the OPRM Upscale function as an "APRM function" in the RPS Instrumentation "function" table. Also add the related surveillance requirements and, if applicable, the related setpoint, and the related descriptions in the bases sections. Perform analysis necessary to	An OPRM Upscale Function is added to the CGS Technical Specification as an "APRM Function" (Function 2.f) consistent with PRNM LTR Supplement 1, Appendix H. Additions to the Technical Specification Bases for Function 2.f have also been incorporated consistent with the PRNM LTR. The PRNM LTR Supplement 1 included some additional wording for Function 2.e (voter) to

Section No.	Utility Action Required	Response
	establish setpoints for the OPRM Upscale trip. Add discussions related to the OPRM function in the Bases for the APRM Inop and 2-out-of-4 Voter functions. NOTE: The markups in Appendix H of Supplement 1 to the PRNM LTR show the OPRM Upscale as an APRM sub-function. However, individual plants may determine that for their particular situation, addition of the OPRM to the RPS Instrumentation table separate from the APRM, or as a separate Technical Specification, better meets their needs. In those cases, the basis elements of the Technical Specifications as shown in this Supplement would remain, but the specific implementation would be different.	address independent voting of the OPRM and APRM signals
8.4.2.4	<u>OPRM-Related RPS Trip Functions - Minimum Number of Operable OPRM Channels</u> For the OPRM functions added (Section 8.4.1), include in the OPRM Technical Specifications a "minimum operable channels" requirement for three OPRM channels, shared by both trip systems. Add the same action statements as for the APRM Neutron Flux - High function for OPRM Upscale function. In addition, add a new action statement for OPRM Upscale function unavailable per Paragraph 8.4.2.2 of the PRNM LTR. Revise the Bases section as needed to add descriptions of the 4-OPRM system with 2-out-of-4 output Voter channels (2 per RPS Trip System), and allowed one OPRM bypass total. The NRC SER states the OPRM function be monitored during the first fuel cycle to ensure the OPRM algorithms perform according to the design specifications. During this monitoring period the OPRM trip capabilities would be disabled. Upon completion of this initial monitoring period, the OPRM trip function would be enabled.	A minimum operable channels requirement of three, shared by both trip systems has been included in the Technical Specification for the OPRM Upscale Function (Function 2.f). This addition, as well as addition of Required Action statements and Bases descriptions, is consistent with the PRNM LTR and LTR Supplement 1. Regarding the initial monitoring period, the GEH NUMAC OPRM system can be installed and activated immediately without an initial monitoring period because: 1) The operating experience of the GEH NUMAC OPRM system in general is sufficient, 2) The GEH NUMAC OPRM system is replacing the current Option III OPRM system, and 3) The data received during the initial monitoring period for the currently "armed" and operating digital OPRM system demonstrated that the algorithm was robust and not sensitive to system settings within the range of values described in NEDO-32465-A.

Section No.	Utility Action Required	Response
8.4.3.4	<u>OPRM-Related RPS Trip Functions - Applicable Modes of Operation</u> Add the requirement for operation of the OPRM Upscale function in Mode 1 (RUN) when Thermal Power is $\geq$ 25% RTP, and add Bases descriptions as required.	A Modes of Operation requirement of $\geq$ 20% RTP, consistent with the PRNM LTR Supplement 1 has been included in the Technical Specification along with associated Bases descriptions. The change from the 25% RTP operability value shown in the LTR to 20% RTP has been done to provide the same margin (5%) between the OPERABILITY requirement and the auto-enable setpoint (25% -- see response to 8.4.6.1 below) as the margin included in the PRNM LTR. The OPRM operable and enabled values are core specific, the actual values used can be the ones specified in the cycle specific COLR.
8.4.4.1.4	<u>OPRM-Related RPS Trip Functions - Channel Check</u> Add once per 12 hour or once per day Channel Check or Instrument Check requirements for the OPRM Upscale function.	A Channel Check requirement of once per 12 hours is included for the OPRM Upscale Function, consistent with the PRNM LTR Supplement 1.
8.4.4.2.4	<u>OPRM-Related RPS Trip Functions - Channel Functional Test</u> Add Channel Functional Test requirements with a requirement for a test frequency of every 184 days (6 months), including the 2-out-of-4 Voter function. Add a "confirm auto-enable region" surveillance on a once per outage basis up to 24 month intervals.	A Channel Functional Test requirement with a test frequency of every 184 days (Table 3.3.1.1-1) has been added as SR 3.3.1.1.16 for the OPRM Upscale and 2-Out-Of 4 Voter Functions consistent with the PRNM LTR, Supplement 1. Note, SR 3.3.1.3 has been removed, including the previously existing OPRM instrumentation section. A second note to SR 3.3.1.1.16 (not included in the PRNM LTR) is included to clarify that the SR also applies to the flow input function, except the transmitters. A "confirm auto-enable region" surveillance requirement 3.3.1.1.17, Table 3.3.1.1-1, is added to require confirmation that the OPRM Upscale trip output auto-enable (not bypassed) setpoints remain correct. The SR Bases wording is consistent with the LTR. The sample Technical Specifications in the LTR include the generic 30% power, and 60% flow values for the auto-enable setpoints as well as the 25% OPRM operable value. The reload stability analysis includes a confirmation that the auto-enable region bounds the part of the power flow map where the plant may be susceptible to an instability event. If this confirmation is not

Section No.	Utility Action Required	Response
		successful, the region boundaries will have to be expanded for that cycle. Similar to the other cycle specific values referenced in the Technical Specifications, the flow and power values defining the auto-enable region boundaries, and the OPRM operable region are specified in COLR. These values are referenced in the Technical Specifications. Note that the cycle specific auto-enable region boundaries are at least as large as the generic auto-enable region boundaries.
8.4.4.3.4	OPRM-Related RPS Trip Functions - Channel Calibration Add calibration interval requirement of every 24 months for the OPRM Upscale function. Revise Bases text as required.	A Channel Calibration requirement for the OPRM Upscale Function references the existing 24-month frequency of SR 3.3.1.1.10 PRNM LTR Supplement 1.
8.4.4.4.4	OPRM-Related RPS Trip Functions - Response Time Testing Modify as necessary the response time testing procedure for the 2-out-of-4 Voter function to include the Voter OPRM output solid-state relays as part of the response time tests, alternating testing of the Voter OPRM output with the Voter APRM output.	See response to 8.3.4.4.4. That response also addresses OPRM. Current CGS Technical Specification SR 3.3.1.1.15 is modified to address response time testing for APRM and OPRM.
8.4.5.4	OPRM-Related RPS Trip Functions - Logic System Functional Testing (LSFT) Add requirement for LSFT every refueling cycle, 18 or 24 months at the utility's option based on which best fits plant scheduling.	The LSFT (Table 3.3.1.1-1) for the OPRM Upscale Function is the same as for the APRM, a test of the 2-Out-Of-4 Voter only. Consistent with the PRNM LTR Supplement 1, the only change required to implement the OPRM "LSFT" is the addition of "and OPRM" in the Technical Specification Bases and revision of the related plant procedures to include testing of the OPRM Upscale trip outputs from the 2-Out-Of-4 Voter. The procedure changes will be made as part of the normal modification process.
8.4.6.1	OPRM-Related RPS Trip Functions - Setpoints Add setpoint information to the appropriate document and identify in the plant-specific submittal the basis or method used for the calculation and where the setpoint information will be recorded.	There are four "sets" of OPRM related setpoints and adjustable parameters: a) OPRM trip auto-enable (not bypassed) setpoints for STP and drive flow; b) period based detection algorithm (PBDA) confirmation count and amplitude setpoints; c) period based detection algorithm tuning parameters; and d) growth rate algorithm

Section No.	Utility Action Required	Response
		(GRA) and amplitude based algorithm (ABA) setpoints. The first set, the setpoints for the "auto-enable" region for OPRM, discussed in the Bases for Function 2.f, will be treated as nominal setpoints with no additional margins added. The deadband for these setpoints is established so that it increases the enabled region once the enabled region is entered. The settings are specified in the COLR from the plant procedures. The second set, the PBDA trip setpoints, will be established in accordance with the BWROG LTR 32465-A methodology, previously reviewed and approved by the NRC, and will be documented in the COLR. The third set, the PBDA "tuning" parameter values, will be established in accordance with and controlled by CGS procedures, within the limits established in the BWROG LTRs, or as documented in this submittal. The fourth set, the GRA and ABA setpoints, consistent with the BWROG submittals, will be established as nominal values only, and controlled by CGS procedures.
8.5.1.4	APRM-Related Control Rod Block Functions - <u>Functions Covered by Technical Specifications</u> If ARTS will be implemented concurrently with the PRNM modification, include or reference those changes in the *plant-specific PRNM submittal.* Implement the applicable portion of the above described changes via modifications to the Technical Specifications and related procedures and documents. In the *plant-specific submittal,* identify functions currently in the plant Technical Specifications and which, if any, changes are being implemented. For any functions deleted from Technical Specifications, identify where setpoint and surveillance requirements will be documented. NOTE: A utility may choose not to delete some or all of the items identified in the PRNM LTR from the plant Technical Specifications.	ARTS will be implemented concurrently with the PRNM modification at CGS. CGS Technical Specifications currently do not contain any APRM rod block functions. These have been moved to the CGS LCS. RBM functions within the Technical Specification, required channels will remain at two.
8.5.2.4	<u>APRM-Related Control Rod Block Functions - Minimum Number of Operable Control Rod Block Channels</u> Change the minimum number of APRM channels to three, if APRM functions are retained in Technical Specifications. No additional action is required relative to minimum operable channels beyond that required by	See 8.5.1.4 above. No additional confirmation of action required relative to minimum operable channels as shown in the Technical Specifications beyond that required by 8.5.1.4 above. The APRM rod block functions are listed in the LCS. In the LCS, the minimum number of

Section No.	Utility Action Required	Response
	Paragraph 8.5.1.4 of the PRNM LTR.	APRM channels is four and will be changed to three. RBM functions, within the Technical Specification, required channels will remain at two.
8.5.3.4	<u>APRM-Related Control Rod Block Functions - Applicable Modes of Operation</u> No action required relative to modes during which the function must be available beyond that required by Paragraph 8.5.1.4 of the PRNM LTR unless APRM functions are retained in Technical Specifications and include operability requirements for Mode 5. In that case, delete such requirements.	See 8.5.1.4 above. No additional confirmation of action required relative to applicable modes of operation as shown in the Technical Specifications beyond that required by 8.5.1.4 above. The APRM rod block functions are listed in the LCS. There are no operability requirements in Mode 5 for the APRM rod block functions in the LCS, consistent with the PRNM LTRs.
8.5.4.1.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Channel Check</u> Delete any requirements for instrument or channel checks related to RBM and, where applicable, recirculation flow rod block functions (non-ARTS plants), and APRM functions. Identify in the plant-specific PRNM submittals if any checks are currently included in Technical Specifications, and confirm that they are being deleted.	CGS Technical Specifications currently do not contain any APRM rod block functions, or any Channel Check requirements for the RBM rod block functions. Therefore, no change to CGS Technical Specifications is required to implement the PRNM LTR requirements. The RBM is not applicable to CGS. The LCS currently includes no Channel Check requirements for the APRM rod block functions.
8.5.4.2.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Channel Functional Test</u> Change Channel Functional Test requirements to identify a frequency of every 184-days (6 months). In the *plant-specific licensing submittal,* identify current Technical Specification test frequencies that will be changed to 184 days (6 months).	CGS Technical Specifications currently do not contain any APRM rod block functions. The Channel Functional Test frequency for the APRM rod block functions is changed to once per 184 days in the LCS.
8.5.4.3.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Channel Calibrations</u> Change channel calibration requirements to identify a frequency of every 24 months. In the *plant-specific licensing submittal,* identify current Technical Specification test frequencies that will be changed to 24 months.	CGS Technical Specifications currently do not contain any APRM rod block functions. The Channel Calibration frequency for the APRM rod block functions is once per 184 days, which is changed to 18 months per the LTR. RBM Channel Calibration check, within the Technical Specifications, was changed from 184 days to 24 months consistent with the NUMAC PRNM LTR.
8.5.4.4.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Response Time Testing</u> None.	CGS Technical Specifications currently do not contain any APRM rod block functions. Response time testing is not required for these functions per the CGS licensing basis.
8.5.5.4	<u>APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Logic System Functional Testing (LSFT)</u> None.	CGS Technical Specifications currently do not contain any APRM rod block functions. Logic System Functional testing is currently included in the LCS at a frequency of 24 months.

Section No.	Utility Action Required	Response
8.5.6.1	APRM-Related Control Rod Block Functions - Required Surveillances and Calibration - Setpoints Add to or delete from the appropriate document any changed control rod block setpoint information. If ARTS is being implemented concurrently with the PRNM modification, either include the related Technical Specifications submittal information with the PRNM information in the *plant-specific submittal,* or reference the ARTS submittal in the PRNM submittal. In the *plant-specific submittal,* identify what changes, if any, are being implemented and identify the basis or method used for calculation of setpoints and where the setpoint information or changes will be recorded.	ARTS will be implemented concurrently with the PRNM modification at CGS. APRM rod block setpoints are based on setpoint calculations using approved setpoint methodology. The actual Allowable Values and setpoints are defined in the LCS. Setpoint changes (if any) that will be implemented due to ARTS/PRNM submittal will be identified.
8.6.2	Shutdown Margin Testing - Refueling As applicable, revise the Shutdown Margin Testing - Refueling (or equivalent Technical Specifications) LCO(s), action statements, surveillance requirements and Bases as required to be consistent with the APRM Technical Specifcation changes implemented for PRNM.	Added function 2.e to SR 3.10.8.1. Technical Specification or Technical Specification Bases changes to Specification 3.10.8, Shutdown Margin (SDM) Test – Refueling and its associated bases were modified consistent with the APRM Technical Specification changes implemented for PRNM.
None	Core Operating Limits Report Reporting requirements Section 5.6.3 does not currently address the OPRM.	Specification 5.6.3 has been modified to require that the Period Based Detection Algorithm (PBDA) setpoints be included in the COLR to support LCO 3.3.1.1.
9.1.3	Utility Quality Assurance Program As part of the *plant-specific licensing submittal,* the utility should document the established program that is applicable to the project modification. The submittal should also document for the project what scope is being performed by the utility and what scope is being supplied by others. For scope supplied by others, document the utility actions taken or planned to define or establish requirements for the project, to assure those requirements are compatible with the plant-specific configuration. Actions taken or planned by the utility to assure compatibility of the GEH-I quality program with the utility program should also be documented. Utility planned level of participation in the overall V&V process for the project should be documented, along with utility plans for software configuration management and provision to support any required changes after	Quality assurance requirements for work performed at CGS are defined and described in Energy Northwest "Operational Quality Assurance Program Description (EN-QA-004)" (OQAPD), LDN-OQAPD-OQAPD-01. Note: The OQAPD applies to all activities associated with structures, systems, and components, which are safety related or controlled by 10 CFR 72. The OQAPD also applies to transportation packages controlled by 10 CFR 71. The OQAPD implements 10 CFR 50 Appendix B, 10 CFR 71 Subpart H, and 10 CFR 72 Subpart G. It is implemented by site procedures and instructions. For the PRNM modification, CGS has contracted with GEH to include the following PRNM scope: 1) design, 2) hardware/ software, 3) licensing support, 4) training, 5) O&M manuals and design documentation, 6) EMI/RFI qualification of equipment, and 7) PRNMS

Section No.	Utility Action Required	Response
	delivery should be documented.	setpoint calculations. On-site engineering work to incorporate the GEH-provided design information into an Engineering Change (EC) or to provide any supporting, interface design changes will be performed per requirements of applicable CGS procedures. Modification work to implement the design change will be performed per CGS procedures or CGS-approved contractor procedures. CGS has participated and will continue to participate in appropriate reviews of GEH's design and V&V program for the PRNM modification. For software delivered in the form of hardware (EPROMs), CGS intends to have GEH maintain post delivery configuration control of the actual source code and handle any changes. CGS will handle any changes in the EPROMs as hardware changes under its applicable hardware modification procedures.

Appendix A

Columbia NUMAC PRNM LTR Deviations

Columbia NUMAC PRNM LTR Deviation

Energy Northwest will be submitting a license application for the implementation of Power Range Neutron Monitor (PRNM) using the Long Term Stability Solution Option III at the Columbia Generating Station. The bases for the license application are the referenced documents in the relevant licensing topical reports (Reference 1-3).

The PRNM developed for Columbia has one deviation from the referenced documents. It is summarized in Table 1 and discussed in detail below. The licensing topical reports explicitly allow for plant-to-plant variation of some features. These are not addressed herein.

Table 1. Columbia NUMAC PRNM LTR Deviations

	Function/ Equipment	PRNM Licensing Basis	Columbia Design	Justification
a.	APRM Upscale / OPRM Upscale / APRM Inop Function Logic	OPRM Upscale function voted separately from the APRM Inop function	OPRM Upscale function voted with the APRM Inop function	Improved operating flexibility

Technical Justifications

a. APRM Upscale / OPRM Upscale / APRM Inop Function Logic

Licensing Topical Report NEDC-32410P-A Supplement 1 (Reference 3) Section 8.4.1.3 describes the logic wherein the OPRM Upscale function is voted separately from the APRM Inop function. That is, an APRM Inop in one APRM channel and an OPRM Upscale in another will result in two half-trips in each of the 2-out-of-4 voter channels, but no RPS trips.

Designed this way, when an APRM chassis keylock switch is placed in the "INOP" position, the APRM upscale trip signal sent to the 2-out-of-4 voter channels is set to trip. However, the OPRM trip output from that chassis continues to be processed normally. Typically this logic is of no consequence because if an APRM chassis (affecting both the APRM and OPRM channels) is declared inoperable, the APRM bypass can be used to bypass both the APRM and OPRM trips from that channel, which in turn modifies the logic in the 2-out-of-4 voter to be a 2-out-of-3 vote of both the APRM and OPRM trips from the remaining 3 channels. However, if the need to declare a second APRM/OPRM channel inoperable arises when another APRM/OPRM channel is already bypassed (and cannot be returned to service within the allowed out of service time), it is necessary to place the APRM and OPRM outputs from the second channel in the tripped condition to satisfy Technical Specification requirements. If the APRM channel is still sufficiently functional to process trip outputs, placing the keylock switch in the INOP position will force a trip for the APRM channel, but not for the OPRM channel. Other action, such as disconnecting a fiber-optic cable to the 2-out-of-4 voters or removing power from the APRM chassis, is necessary to satisfy the requirement to place the OPRM channel in the tripped condition.

The automatic APRM Inop trip is intended to provide a trip when the APRM channel is known to be incapable of providing a trip based on normal functions. This trip occurs immediately even though the Technical Specification requirements allow a period of time for action. The automatic trip is provided to assure that conditions that may disable the APRM trip function do not go undetected. Since the OPRM trip function is implemented in the same equipment as the APRM trip function, conditions that could disable the APRM trip function would likely disable the OPRM trip function as well.

For the Columbia PRNM, the OPRM Upscale function is combined with the APRM Inop function as the OPRM channel input to be voted. That is, an APRM Inop in one APRM channel and an OPRM Upscale in another will result in RPS trip outputs from all four 2-out-of-4 voter channels. Again this logic is typically of no consequence because if an APRM chassis (affecting both the APRM and OPRM channels) is declared inoperable, the APRM bypass can be used to bypass both the APRM and OPRM trips from that channel, which in turn modifies the logic in the 2-out-of-4 voter to be a 2-out-of-3 vote of both the APRM and OPRM trips from the remaining 3 channels. This design allows using the APRM chassis keylock switch to place APRM and OPRM outputs from a second channel in the tripped condition when another APRM/OPRM channel is already bypassed (and cannot be returned to service within the allowed out of service time) without having to resort to other actions such as disconnecting a fiber-optic cable to the 2-out-of-4 voters or removing power from the APRM chassis.

For the Columbia PRNM, the Supplement 1 (Reference 3) Bases are changed as follows.

1. Page H-12: change the second paragraph as shown below.

    The APRM System is divided into four APRM channels and four 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channels, to be bypassed. A trip from any one unbypassed APRM will result in a "half-trip" in all four of the voter channels, but no trip inputs to either RPS trip system. ~~APRM trip Functions 2.a, 2.b, 2.c, and 2.d are voted independently from OPRM Upscale Function 2.f. Therefore, any Function 2.a, 2.b, 2.c, or 2.d trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1, and B2). Similarly, a Function 2.f trip from any two unbypassed APRM channels will result in a full trip from each of the four voter channels.~~ Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM Functions 2.a, 2.b, and 2.c, at least [20] LPRM inputs, with at least [three] LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel. For the OPRM Upscale, Function 2.f, LPRMs are assigned to "cells" of [4] detectors. A minimum of [later] cells, each with a minimum of [2] LPRMs, must be OPERABLE for the OPRM Upscale Function 2.f to be OPERABLE.

    Replaced deleted text with the following:

    Since APRM trip Functions 2.a, 2.b, 2.c and 2.f are implemented in the same hardware, these trip Functions are combined with APRM Inop trip Function 2.d. Any Function 2.a, 2.b, 2.c or 2.d trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1, and B2). Similarly, any Function 2.d or 2.f trip from any two unbypassed APRM channels will result in a full trip from each of the four voter channels.

2. Page H-13: For Function 2.e, change the 1st sentence of the 3rd paragraph to the following. "The 2-Out-Of-4 Voter Function votes APRM Functions 2.a, 2.b, and 2.c independently of Function 2.f."

References

1. NEDC-32410P-A Volume 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October, 1995.
2. NEDC-32410P-A Volume 2 -- Appendices, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," October, 1995.
3. NEDC-32410P-A Supplement 1, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," November, 1997.

Technical Specifications page Markups

TABLE OF CONTENTS

1.0	USE AND APPLICATION	
1.1	Definitions	1.1-1
1.2	Logical Connectors	1.2-1
1.3	Completion Times	1.3-1
1.4	Frequency	1.4-1
2.0	SAFETY LIMITS (SLs)	
2.1	SLs	2.0-1
2.2	SL Violations	2.0-1
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	3.0-1
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	3.0-4
3.1	REACTIVITY CONTROL SYSTEMS	
3.1.1	SHUTDOWN MARGIN (SDM)	3.1.1-1
3.1.2	Reactivity Anomalies	3.1.2-1
3.1.3	Control Rod OPERABILITY	3.1.3-1
3.1.4	Control Rod Scram Times	3.1.4-1
3.1.5	Control Rod Scram Accumulators	3.1.5-1
3.1.6	Rod Pattern Control	3.1.6-1
3.1.7	Standby Liquid Control (SLC) System	3.1.7-1
3.1.8	Scram Discharge Volume (SDV) Vent and Drain Valves	3.1.8-1
3.2	POWER DISTRIBUTION LIMITS	
3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	3.2.1-1
3.2.2	MINIMUM CRITICAL POWER RATIO (MCPR)	3.2.2-1
3.2.3	LINEAR HEAT GENERATION RATE (LHGR)	3.2.3-1
~~3.2.4~~	~~Average Power Range Monitor (APRM) Gain and Setpoint~~	~~3.2.4-1~~
3.3	INSTRUMENTATION	
3.3.1.1	Reactor Protection System (RPS) Instrumentation	3.3.1.1-1
3.3.1.2	Source Range Monitor (SRM) Instrumentation	3.3.1.2-1
~~3.3.1.3~~	~~Oscillation Power Range Monitor (OPRM) Instrumentation~~	~~3.3.1.3-1~~
3.3.2.1	Control Rod Block Instrumentation	3.3.2.1-1
3.3.2.2	Feedwater and Main Turbine High Water Level Trip Instrumentation	3.3.2.2-1
3.3.3.1	Post Accident Monitoring (PAM) Instrumentation	3.3.3.1-1
3.3.3.2	Remote Shutdown System	3.3.3.2-1
3.3.4.1	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	3.3.4.1-1
3.3.4.2	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	3.3.4.2-1
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation	3.3.5.1-1

(continued)

1.1 Definitions (continued)

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

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.1.7.6	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1220 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.8	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months AND Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1

SR 3.1.7.9	Verify sodium pentaborate enrichment is ≥ 44.0 atom percent B-10.	Prior to addition to SLC Tank

Deleted

~~3.2 POWER DISTRIBUTION LIMITS~~

~~3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint~~

~~LCO 3.2.4~~

~~a.~~ ~~MFLPD shall be less than or equal to Fraction of RTP (FRTP); or~~

~~b.~~ ~~Each required APRM Flow Biased Simulated Thermal Power—High Function Allowable Value shall be modified by greater than or equal to the ratio of FRTP and the MFLPD; or~~

~~c.~~ ~~Each required APRM gain shall be adjusted such that the APRM readings are ≥ 100% times MFLPD.~~

~~APPLICABILITY:~~ ~~THERMAL POWER ≥ 25% RTP.~~

~~ACTIONS~~

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

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.4.1~~	~~-------------------NOTE-------------------~~ ~~Not required to be met if SR 3.2.4.2 is satisfied for LCO 3.2.4.b or LCO 3.2.4.c requirements.~~ ~~--~~ ~~Verify MFLPD is within limits.~~	~~Once within 12 hours after ≥ 25% RTP~~ ~~AND~~ ~~24 hours thereafter~~
~~SR 3.2.4.2~~	~~-------------------NOTE-------------------~~ ~~Not required to be met if SR 3.2.4.1 is satisfied for LCO 3.2.4.a requirements.~~ ~~--~~ ~~Verify each required:~~ ~~a. APRM Flow Biased Simulated Thermal Power - High Function Allowable Value is modified by greater than or equal to the ratio of FRTP and the MFLPD; or~~ ~~b. APRM gain is adjusted such that the APRM reading is ≥ 100% times MFLPD.~~	~~12 hours~~

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable. Add note before A.2 & B ---------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -------------------------	A.1 Place channel in trip. <u>OR</u> A.2 Place associated trip system in trip.	12 hours 12 hours
B. One or more Functions with one or more required channels inoperable in both trip systems.	B.1 Place channel in one trip system in trip. <u>OR</u> B.2 Place one trip system in trip.	6 hours 6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

Insert A

Insert A

I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations. <u>AND</u> I.2 --------------------NOTE---------------- LCO 3.0.4 is not applicable -- Restore required channels to OPERABLE.	12 hours 120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1 Reduce THERMAL POWER to ~~≤ 25% RTP~~. less than the value specified in the COLR.	4 hours

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

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

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP ~~plus any gain adjustment required by LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint,"~~ while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. --------------------------------------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1130 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	~~------------------NOTES------------------~~ ~~1. Neutron detectors are excluded.~~ ~~2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2.~~ ~~---------------------------------------~~ ~~Perform CHANNEL CALIBRATION.~~ Deleted	~~184 days~~

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. -- Perform CHANNEL CALIBRATION.	18 months for Functions 1, 3, ~~through~~ 4, 6, 7, and 9 through 11 AND 24 months for Functions 2, 5, ~~5~~ and 8
SR 3.3.1.1.11	~~Verify the APRM Flow Biased Simulated Thermal Power - High Function time constant is ≤ 7 seconds.~~ Deleted	~~18 months~~
SR 3.3.1.1.12	Verify Turbine Throttle Valve – Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 30% RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.15	-------------------NOTES------------------ 1. Neutron detectors are excluded. 2. Channel sensors for Functions 3 and 4 are excluded. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

Insert B

4. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and OPRM outputs shall alternate.

Insert B

SR 3.3.1.1.16	-------------------------NOTE------------------------- 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. --- Perform CHANNEL FUNCTIONAL TEST.	184 days
SR 3.3.1.1.17	Verify the Oscillation Power Range Monitor (OPRM) is not bypassed when APRM Simulated Thermal Power is greater than or equal to the value specified in the COLR <u>AND</u> recirculation drive flow is less than the value specified in the COLR.	24 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – High~~,~~ ~~Setdown~~ (Setdown)	2	~~2~~ 3(b)	G	SR 3.3.1.1.1 ~~SR 3.3.1.1.3~~ SR 3.3.1.1.6 SR 3.3.1.1.7 ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.14~~ SR 3.3.1.1.10 SR 3.3.1.1.16	≤ 20% RTP
b. ~~Flow Biased~~ Simulated Thermal Power – High	1	~~2~~ 3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.11~~ ~~SR 3.3.1.1.14~~ SR 3.3.1.1.10 SR 3.3.1.1.16	~~≤ 0.58W + 62% RTP and ≤ 114.9% RTP~~ ≤ 0.63W + 64.0% RTP and ≤ 114.9% RTP (c)
c. ~~Fixed~~ Neutron Flux – High	1	~~2~~ 3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.9~~ ~~SR 3.3.1.1.14~~ ~~SR 3.3.1.1.15~~ SR 3.3.1.1.10 (d) (e) SR 3.3.1.1.16	≤ 120% RTP
Insert C d. Inop	1,2	~~2~~ 3(b)	G	~~SR 3.3.1.1.7~~ ~~SR 3.3.1.1.8~~ ~~SR 3.3.1.1.14~~ SR 3.3.1.1.16	NA
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 1079 psig

(continued)

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

Insert C1

INSERT C

e. 2-Out-of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16	NA
f. OPRM Upscale	1(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	NA (g)

INSERT C1

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(c) ≤ 0.63W + 60.8% RTP and ≤ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(d) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, and the methodology used to determine this value, is specified in the Licensee Controlled Specifications.

(f) When greater than the RTP value specified in the COLR.

(g) The OPRM Upscale does not have an Allowable Value. The Period Based Detection Algorithm (PBDA) trip setpoints are specified in the COLR.

~~3.3 INSTRUMENTATION~~ Deleted

~~3.3.1.3 Oscillation Power Range Monitor (OPRM) Instrumentation~~

~~LCO 3.3.1.3~~ ~~Four channels of the OPRM instrumentation shall be OPERABLE within the limits as specified in the COLR.~~

~~APPLICABILITY:~~ ~~THERMAL POWER ≥ 25% RTP.~~

~~ACTIONS~~

~~NOTE~~
~~Separate Condition entry is allowed for each channel.~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A.~~ ~~One or more required channels inoperable.~~	~~A.1~~ ~~Place channel in trip.~~	~~30 days~~
	~~OR~~	
	~~A.2~~ ~~Place associated RPS trip system in trip.~~	~~30 days~~
	~~OR~~	
	~~A.3~~ ~~Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.~~	~~30 days~~

~~(continued)~~

~~ACTIONS~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. OPRM trip capability not maintained.~~	~~B.1 Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.~~	~~12 hours~~
~~C. Required Action and associated Completion Time not met.~~	~~C.1 Reduce THERMAL POWER < 25% RTP.~~	~~4 hours~~

~~SURVEILLANCE REQUIREMENTS~~

~~NOTE~~

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

~~SURVEILLANCE~~	~~FREQUENCY~~
~~SR 3.3.1.3.1 Perform CHANNEL FUNCTIONAL TEST.~~	~~184 days~~

~~(continued)~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.3.1.3.2~~	~~Calibrate the local power range monitors.~~	~~1130 MWD/T average core exposure~~
~~SR 3.3.1.3.3~~	~~-----NOTE-----~~ ~~Neutron detectors are excluded.~~ ~~----------~~ ~~Perform CHANNEL CALIBRATION.~~	~~24 months~~
~~SR 3.3.1.3.4~~	~~Perform LOGIC SYSTEM FUNCTIONAL TEST.~~	~~24 months~~
~~SR 3.3.1.3.5~~	~~Verify OPRM is not bypassed when THERMAL POWER is ≥ 30% RTP and core flow ≤ 60% rated core flow.~~	~~24 months~~
~~SR 3.3.1.3.6~~	~~-----NOTE-----~~ ~~Neutron detectors are excluded.~~ ~~----------~~ ~~Verify the RPS RESPONSE TIME is within limits.~~	~~24 months on a STAGGERED TEST BASIS~~

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	<u>AND</u>		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTES-----

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

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

	SURVEILLANCE	FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	~~92~~ $\boxed{184}$ days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-------------------NOTE------------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	-------------------NOTE------------------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4 Insert C2	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RBM is not bypassed: ~~a. When THERMAL POWER is ≥ 30% RTP; and~~ ~~b. When a peripheral control rod is not selected.~~	24 months ~~92 days~~
SR 3.3.2.1.5	-------------------NOTE------------------- Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	24 months ~~92 days~~

(continued)

INSERT C2

a. Low Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and a peripheral control rod is not selected.

b. Intermediate Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and a peripheral control rod is not selected.

c. High Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 83% RTP and a peripheral control rod is not selected.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
~~a. Upscale~~	~~(a)~~	~~2~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.4~~ ~~SR 3.3.2.1.5~~	~~≤ 0.58W + 51% RTP~~
Insert C3				
d. ~~b.~~ Inop	(a),(b),(c) ~~(a)~~	2	SR 3.3.2.1.1 ~~SR 3.3.2.1.4~~	NA
~~c. Downscale~~	~~(a)~~	~~2~~	~~SR 3.3.2.1.1~~ ~~SR 3.3.2.1.4~~ ~~SR 3.3.2.1.5~~	~~≥ 3% RTP~~
2. Rod Worth Minimizer	1(d) ~~(b)~~, 2(d) ~~(b)~~	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA
3. Reactor Mode Switch — Shutdown Position	~~(c)~~ (e)	2	SR 3.3.2.1.7	NA

Insert C4

~~(a) THERMAL POWER ≥ 30% RTP and no peripheral control rod selected.~~

(d) ~~(b)~~ With THERMAL POWER ≤ 10% RTP.

(e) ~~(c)~~ Reactor mode switch in the shutdown position.

Insert C5

INSERT C3

a. Low Power Range – Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5 (g)(h)	(f)
b. Intermediate Power Range – Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5 (g)(h)	(f)
c. High Power Range – Upscale	(c)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5 (g)(h)	(f)

INSERT C4

(a) APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and MCPR is less than the limit specified in the COLR and no peripheral control rod selected.

(b) APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and MCPR is less than the limit specified in the COLR and no peripheral control rod selected.

(c) APRM Simulated Thermal Power is ≥ 83% and MCPR is less than the limit specified in the COLR and no peripheral control rod selected.

INSERT C5

(f) Allowable Value specified in the COLR.

(g) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(h) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, and the methodology used to determine this value, is specified in the Licensee Controlled Specifications.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation provided that the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR; ~~and~~

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR~~.~~ ; and

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors, Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop flow mismatch not within limits.	A.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours
B. Requirements of the LCO not met for reasons other than Condition A.	B.1 Satisfy the requirements of the LCO.	4 hours

(continued)

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test — Refueling

LCO 3.10.8

The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Function 2.a, ~~and~~ 2.d, of Table 3.3.1.1-1; and 2.e

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with banked position withdrawal sequence requirements of SR 3.3.2.1.8 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated control rod drive (CRD);

d. All control rod withdrawals during out of sequence control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, ~~and~~ 2.d, and 2.e of Table 3.3.1.1-1.	According to the applicable SRs
SR 3.10.8.2	------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. --- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. --- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

5.6 Reporting Requirements (continued)

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

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

1. The APLHGR for Specification 3.2.1;

2. The MCPR for Specification 3.2.2;

3. The LHGR for Specification 3.2.3; ~~and~~

~~4. LCO 3.3.1.3, "Oscillation Power Range Monitor (OPRM) Instrumentation."~~

4. The Oscillation Power Range Monitor (OPRM) Instrumentation for Specification 3.3.1.1; and
5. The Rod Block Monitor Instrumentation for Specification 3.3.2.1.

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

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company

3. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation

4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation

5. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company

(continued)

LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM / ARTS / MELLLA IMPLEMENTATION
Attachment 3

Technical Specifications Bases page markups (for information only)

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Standby Liquid Control (SLC) System

BASES

BACKGROUND

The SLC System is designed to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon free state without taking credit for control rod movement. The SLC System satisfies the requirements of 10 CFR 50.62 (Ref. 1) on anticipated transient without scram (ATWS).

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water (Ref. 4).

The SLC System consists of a boron solution storage tank, two positive displacement pumps, two explosive valves, which are provided in parallel for redundancy, and associated piping and valves used to transfer borated water from the storage tank to the reactor pressure vessel (RPV). The borated solution is discharged through the high pressure core spray system sparger.

APPLICABLE SAFETY ANALYSES

CHANGE B 3.1.7-1

The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures, if the operator believes the reactor cannot be shut down, or kept shut down, with the control rods. The SLC System is used in the event that not enough control rods can be inserted to accomplish shutdown and cooldown in the normal manner. The SLC System injects borated water into the reactor core to compensate for all of the various reactivity effects that could occur during plant operation. To meet this objective, it is necessary to inject, using ~~both~~ one SLC pump~~s~~, a quantity of boron ~~that produces a concentration of 660~~ equivalent in Boron-10 to a concentration of 780 ppm of natural boron in the reactor core, including recirculation loops, at 70°F and normal reactor water level.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

To allow for potential leakage and imperfect mixing in the reactor system, an additional amount of boron equal to 25% of the amount cited above is added (Ref. 2). ~~An additional 275 ppm is provided to accommodate dilution in the RPV by the residual heat removal shutdown cooling piping.~~ ~~The temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved.~~ This quantity of borated solution is the amount that is above the pump suction shutoff level in the boron solution storage tank. No credit is taken for the portion of the tank volume that cannot be injected.

The volume limit in SR 3.1.7.1 and the temperature versus concentration limits in Figure 3.1.7-1 are calculated such that the required concentration is achieved accounting for dilution in the RPV with normal water level and including the water volume in the residual heat removal shutdown cooling piping and in the recirculation loop piping.

CHANGE 3.1.7-2

Following a LOCA, offsite doses from the accident will remain within 10 CFR 50.67, "Accident Source Term," limits (Ref. 5) provided sufficient iodine activity is retained in the suppression pool. Credit for iodine deposition in the suppression pool is allowed (Ref. 4) as long as suppression pool pH is maintained at or above 7. Alternative Source Term analyses credit the use of the SLC System for maintaining the pH of the suppression pool at or above 7.

The SLC System satisfies Criteria 3 and 4 of Reference 3.

The OPERABILITY of the SLC System provides backup capability for reactivity control, independent of normal reactivity control provisions provided by the control rods. Additionally, an OPERABLE SLC System has the ability to inject boron under post LOCA conditions to maintain the suppression pool pH above 7. The OPERABILITY of the SLC System is based on the conditions of the borated solution in the storage tank and the availability of a flow path to the RPV, including the OPERABILITY of the pumps and valves. Two SLC subsystems are required to be OPERABLE, each containing an OPERABLE pump, an explosive valve and associated piping, valves, and instruments and controls to ensure an OPERABLE flow path.

APPLICABILITY

In MODES 1 and 2, shutdown capability is required. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate controls to ensure the reactor remains subcritical. In MODE 5, only a

(continued)

BASES

APPLICABILITY (continued)

single control rod can be withdrawn from a core cell containing fuel assemblies. Demonstration of adequate SDM (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") ensures that the reactor will not become critical. Therefore, the SLC System is not required to perform its ATWS function during MODES 3, 4, or 5.

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 5) limits following a LOCA involving significant fission product releases. The SLC System is used to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water (Ref. 4).

ACTIONS

A.1

If one SLC System subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this condition, the remaining OPERABLE subsystem is adequate to perform the original licensing basis shutdown function. ~~However, the overall capability is reduced since the remaining OPERABLE subsystem cannot meet the requirements of Reference 1.~~ However, the overall reliability is reduced because a single failure in the remaining OPERABLE subsystem could result in reduced SLC System shutdown capability. The 7 day Completion Time is based on the availability of an OPERABLE subsystem capable of performing the original licensing basis SLC System function and the low probability of a Design Basis Accident (DBA) or severe transient occurring concurrent with the failure of the Control Rod Drive System to shut down the plant.

CHANGE 3.1.7-3

B.1

If both SLC subsystems are inoperable, at least one subsystem must be restored to OPERABLE status within 8 hours. The allowed Completion Time of 8 hours is considered acceptable, given the low probability of a DBA or transient occurring concurrent with the failure of the control rods to shut down the reactor.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.7.7 and SR 3.1.7.8 (continued)

acceptable method for verifying that the suction piping up to the suction valve is unblocked is to pump from the storage tank to the test tank. Upon completion of this verification, the pump suction piping must be drained and flushed with demineralized water since the suction piping between the pump suction valve and pump suction is not heat traced. The 24 month Frequency is acceptable since there is a low probability that the subject piping will be blocked due to precipitation of the boron from solution in the heat traced piping. However, if, in performing SR 3.1.7.1, it is determined that the temperature of the solution in the storage tank has fallen below the specified minimum, SR 3.1.7.8 must be performed once within 24 hours after the solution temperature is restored within the limits of Figure 3.1.7-1.

SR 3.1.7.9
Enriched sodium pentaborate solution is made by mixing granular, enriched sodium pentaborate with water. Isotopic tests on the granular sodium pentaborate to verify the actual B-10 enrichment must be performed prior to addition to the SLC tank in order to ensure that the proper B-10 atom percentage is being used.

CHANGE SR 3.1.7.9

REFERENCES

1. 10 CFR 50.62.
2. FSAR, Section 9.3.5.3.
3. 10 CFR 50.36(c)(2)(ii).
4. Regulatory Guide 1.183, July 2000,
5. 10 CFR 50.67, "Accident Source Term."

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 3, 4, 5, 6, and 7. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 50.67. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Ref. 8).

1

INSERT A

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

INSERT A:

LHGR limits are developed as a function of exposure, core flow and power to ensure adherence to fuel design limits during the limiting AOOs (Ref. 10). The exposure dependent LHGR limits are reduced by power-dependent ($LHGRFAC_p$) and core flow-dependent ($LHGRFAC_f$) multipliers for operation below rated power and flow. This will result in a lower LHGR thermal limit at reduced power and flow. A step change in the LHGR limit occurs when scram trips are bypassed for turbine throttle valve closure and turbine governor valve fast closure.

$LHGRFAC_f$ multipliers are determined using the three dimensional BWR simulator code (Ref. 11) to analyze slow flow runout transients. $LHGRFAC_f$, is dependent on the maximum core flow runout capability. $LHGRFAC_p$ multipliers are determined based on analyses of limiting plant transients (other than core flow increases) over a range of power and flow conditions. The transient response is sensitive to initial core flow at power levels below those at which turbine throttle valve closure and turbine governor valve fast closure scram trips are bypassed (P_{bypass}). Both high and low core flow $LHGRFAC_p$ multipliers are provided for operation at power levels between 25% RTP and P_{bypass}. A complete discussion of the analysis code is provided in Reference 12. The exposure, core flow and power dependent LHGR limits ensure that all fuel design limits are met for normal operation and AOOs.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

1. FSAR, Chapter 4.

2. FSAR, Chapter 15.

3. NEDC-32868P, "GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR)," Revision 2, September 2007.

4. NEDC-33241P, "GE14 Fuel Rod Thermal-Mechanical Design Report," Revision 1, January 2006.

5. NEDC-33236P, "GE14 Fuel Assembly Mechanical Design Report," November 2005.

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

7. EMF-85-74(P) Revision 0 Supplement 1 (P)(A) and Supplement 2 (P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Nuclear Power Corporation, February 1998.

8. NUREG-0800, Section II A.2(g), Revision 2, July 1981.

9. 10 CFR 50.36(c)(2)(ii).

2

INSERT A1

INSERT A1:

10. NEDC-33507P, Revision 0, “Energy Northwest Columbia Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA),” March 2010.

11. NEDO-30130-A, "Steady State Nuclear Methods," May 1985.

12. NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978.

3 ~~B 3.2 POWER DISTRIBUTION LIMITS~~

~~B 3.2.4 Average Power Range Monitor (APRM) Gain and Setpoint~~

~~BASES~~

~~BACKGROUND~~

~~The OPERABILITY of the APRMs and their setpoints is an initial condition of all safety analyses that assume rod insertion upon reactor scram. Applicable GDCs are GDC 10, "Reactor Design"; GDC 13, "Instrumentation and Control"; GDC 20, "Protection System Functions"; and GDC 29, "Protection against Anticipated Operation Occurrences" (Ref. 1). This LCO is provided to require the APRM gain or APRM flow biased scram setpoints to be adjusted when operating under conditions of excessive power peaking to maintain acceptable margin to the fuel cladding integrity Safety Limit (SL) and the fuel cladding 1% plastic strain limit.~~

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

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

~~indicating that MFPLD is not decreasing proportionately to the overall power reduction, or conversely, that power peaking is increasing. To maintain margins similar to those at RTP conditions, the excessive power peaking is~~ of ~~compensated by gain adjustment on the APRMs or adjustment of the APRM Flow Biased Simulated Thermal Power - High Function Allowable Value (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b). Either of these adjustments has effectively the same result as maintaining MFLPD less than or equal to FRTP and thus maintains RTP margins for APLHGR, MCPR, and LHGR.~~

~~The normally selected APRM Flow Biased Simulated Thermal Power - High Function Allowable Value positions the scram above the upper bound of the normal power/flow operating region that has been considered in the design of the fuel rods. The Allowable Value is flow biased with a slope that approximates the upper flow control line. The normally~~

~~(continued)~~

~~BASES~~

~~BACKGROUND (continued)~~

~~selected APRM Allowable Value is supported by the analyses presented in References 1 and 2 that concentrate on events initiated from rated conditions. Design experience has shown that minimum deviations occur within expected margins to operating limits (APLHGR, MCPR, and LHGR), at rated conditions for normal power distributions. However, at other than rated conditions, control rod patterns can be established that significantly reduce the margin to thermal limits. Therefore, the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value may be reduced during operation when the combination of THERMAL POWER and MFLPD indicates an excessive power peaking distribution.~~

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

~~APPLICABLE SAFETY ANALYSES~~

~~The acceptance criteria for the APRM gain or setpoint adjustments are that acceptable margins (to APLHGR, MCPR, and LHGR) be maintained to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.~~

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

~~(continued)~~

~~BASES~~

~~APPLICABLE SAFETY ANALYSES (continued)~~

~~(APLHGR)," LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR) limit the initial margins to these operating limits at rated conditions so that specified acceptable fuel design limits are met during transients initiated from rated conditions. At initial power levels less than rated levels, the margin degradation of either the APLHGR, the MCPR, or the LHGR during a transient can be greater than at the rated condition event. This greater margin degradation during the transient is primarily offset by the larger initial margin to limits at the lower than rated power levels. However, power distributions can be hypothesized that would result in reduced margins to the pre-transient operating limit. When combined with the increased severity of certain transients at other than rated conditions, the SLs could be approached. At substantially reduced power levels, highly peaked power distributions could be obtained that could reduce thermal margins to the minimum levels required for transient events. To prevent or mitigate such situations, either the APRM gain is adjusted upward by the ratio of the core limiting MFLPD to the FRTP, or the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value is required to be reduced by the ratio of FRTP to the core limiting MFLPD. Either of these adjustments effectively counters the increased severity of some events at other than rated conditions by proportionally increasing the APRM gain or proportionally lowering the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value dependent on the increased peaking that may be encountered.~~

~~The APRM gain and setpoint satisfy Criteria 2 and 3 of Reference 3.~~

~~LCO~~

~~Meeting any one of the following conditions ensures acceptable operating margins for events described above:~~

~~a. Limiting excess power peaking;~~

~~b. Reducing the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value by multiplying the APRM Flow Biased Simulated Thermal Power—High Function Allowable Value by the ratio of FRTP and the core limiting value of MFLPD; or~~

~~(continued)~~

~~BASES~~

~~LCO (continued)~~

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

~~MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Framatome ANP fuel, MFDLRX is the equivalent of MFLPD. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Simulated Thermal Power - High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Simulated Thermal Power - High Function Allowable Value. Adjusting the APRM gain or modifying the Flow Biased Simulated Thermal Power - High Function Allowable Value is equivalent to maintaining MFLPD less than or equal to FRTP, as stated in the LCO.~~

~~For compliance with LCO Item b (APRM Flow Biased Simulated Thermal Power - High Function Allowable Value modification) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b, are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain or Allowable Value adjusted or modified independently of other APRMs that are having their gain or Allowable Value adjusted or modified.~~

~~APPLICABILITY~~

~~The MFLPD limit, APRM gain adjustment, or APRM Flow Biased Simulated Thermal Power - High Function Allowable Value modification is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the plant is operating at ≥ 25% RTP.~~

~~(continued)~~

~~BASES (continued)~~

~~ACTIONS~~

~~A.1~~

~~If the APRM gain or Flow Biased Simulated Thermal Power — High Function Allowable Value is not within limits while the MFLPD has exceeded FRTP, the margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit may be reduced. Therefore, prompt action should be taken to restore the MFLPD to within its required limit or make acceptable APRM adjustments such that the plant is operating within the assumed margin of the safety analyses.~~

~~The 6 hour Completion Time is normally sufficient to restore either the MFLPD to within limits or the APRM gain or Flow Biased Simulated Thermal Power — High Function Allowable Value to within limits and is acceptable based on the low probability of a transient or Design Basis Accident occurring simultaneously with the LCO not met.~~

~~B.1~~

~~If the APRM gain or Flow Biased Simulated Thermal Power — High Function Allowable Value cannot be restored to within their required limits within the associated Completion Time, the plant must be brought to a MODE or other specified condition in which the LCO does not apply. To achieve this status, THERMAL POWER must be reduced to < 25% RTP within 4 hours. The allowed Completion Time is reasonable, based on operating experience, to reduce THERMAL POWER to < 25% RTP in an orderly manner and without challenging plant systems.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.2.4.1 and SR 3.2.4.2~~

~~The MFLPD is required to be calculated and compared to FRTP or APRM gain or Flow Biased Simulated Thermal Power — High Function Allowable Value to ensure that the reactor is operating within the assumptions of the safety analysis. These SRs are required only to determine the MFLPD and, assuming MFLPD is greater than FRTP, the appropriate APRM gain or Flow Biased Simulated Thermal Power — High Function Allowable Value, and is not intended to be a CHANNEL FUNCTIONAL TEST for the APRM gain or APRM Flow Biased Simulated Thermal Power — High Function circuitry. The 24 hour Frequency of SR 3.2.4.1 is chosen to coincide with~~

~~(continued)~~

~~BASES~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.2.4.1 and SR 3.2.4.2 (continued)~~

~~the determination of other thermal limits, specifically those for the APLHGR and LHGR (LCO 3.2.1 and LCO 3.2.3, respectively). The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.~~

~~The 12 hour Frequency of SR 3.2.4.2 is required when MFLPD is greater than FRTP, because more rapid changes in power distribution are typically expected.~~

~~REFERENCES~~

1. ~~10 CFR 50, Appendix A, GDC 10, GDC 13, GDC 20, and GDC 29.~~
2. ~~FSAR, Chapters 15 and 15.F.~~
3. ~~10 CFR 50.36(c)(2)(ii).~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

1.b. Intermediate Range Monitor—Inop

This trip signal provides assurance that a minimum number of IRMs are OPERABLE. Anytime an IRM mode switch is moved to any position other than "Operate," the detector voltage drops below a preset level, loss of the negative DC voltage, or a module is not plugged in, an inoperative trip signal will be received by the RPS unless the IRM is bypassed. Since only one IRM in each trip system may be bypassed, only one IRM in each RPS trip system may be inoperable without resulting in an RPS trip signal.

This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

Six channels of Intermediate Range Monitor—Inop with three channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal.

Since this Function is not assumed in the safety analysis, there is no Allowable Value for this Function.

This Function is required to be OPERABLE when the Intermediate Range Monitor Neutron Flux—High Function is required.

Insert D

2.a. Average Power Range Monitor Neutron Flux—High, ~~Setdown~~ (Setdown)

~~The APRM channels receive input signals from the local power range monitors (LPRM) within the reactor core, which provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP.~~ For operation at low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High (Setdown) ~~Setdown~~ Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High (Setdown) ~~Setdown~~ Function will provide a secondary scram to the Intermediate Range Monitor Neutron

(continued)

Insert D

2. <u>Average Power Range Monitor</u>

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The APRM channels receive input signals from the local power range monitors (LPRMs) within the reactor to provide an indication of the power distribution and local power changes. The APRM channels average these LPRM signals to provide a continuous indication of average reactor power from a few percent to greater than RTP. Each ARPM also includes an Oscillation Power Range Monitor (OPRM) Upscale Function which monitors small groups of LPRM signals to detect thermal-hydraulic instabilities.

The APRM System is divided into four APRM channels and 2-out-of-4 voter channels. Each APRM channel provides inputs to each of the four voter channels. The four voter channels are divided into two groups of two each, with each group of two providing inputs to one RPS trip system. The system is designed to allow one APRM channel, but no voter channel, to be bypassed. A trip from any one unbypassed APRM will result in a "single vote" in all of the voter channels, but no trip inputs to either RPS trip system. Since APRM trip Functions 2.a, 2.b, 2.c, and 2.f are implemented in the same hardware, these trip Functions are combined with APRM Inop trip Function 2.d. Any Function 2.a, 2.b, 2.c or 2.d trip from any two unbypassed APRM channels will result in a full trip in each of the four voter channels, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1, and B2). Similarly, any Function 2.d or 2.f trip from any two unbypassed APRM channels will result in a full trip from each of the four voter channels. Three of the four APRM channels and all four of the voter channels are required to be OPERABLE to ensure that no single failure will preclude a scram on a valid signal. In addition, to provide adequate coverage of the entire core, consistent with the design bases for the APRM Functions 2.a, 2.b, and 2.c, at least 20 APRM inputs, with at least three LPRM inputs from each of the four axial levels at which the LPRMs are located, must be operable for each APRM channel. To ensure that the plant is operated within analyzed conditions, a maximum of 9 LPRMs can be bypassed between APRM adjustments. For the OPRM Upscale, Function 2.f, LPRMs are assigned to "cells" of 4 detectors. A minimum of 25 cells, each with a minimum of 2 LPRMs, must be OPERABLE for the OPRM Upscale Function 2.f to be OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

27

2.a. Average Power Range Monitor Neutron Flux - High, ~~Setdown~~ (continued)

(Setdown)

(Setdown) 28

Flux - High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux - High, ~~Setdown~~ Function will provide the primary trip signal for a core-wide increase in power.

29 (Setdown)

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux - High, ~~Setdown~~ Function. However, this Function indirectly ensures that, before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

30

~~The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux - High, Setdown, with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

In accordance with the NRC Safety Evaluation for Amendment TBD (Ref. 20), the Average Power Range Monitor Neutron Flux - High (Setdown) Function is not LSSS SL-related.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

31 (Setdown)

The Average Power Range Monitor Neutron Flux - High, ~~Setdown~~ Function must be OPERABLE during MODE 2 when control rods may be withdrawn. In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and Rod Block Monitor protect against control rod withdrawal error events.

32

(continued)

See Section 2, Average Power Range Monitor, for additional information.

THE APRM CALCULATES THE SIMULATED THERMAL POWER (STP) LEVEL OF THE REACTOR CORE BY APPLYING A SINGLE-POLE INFINITE IMPULSE RESPONSE (IIR) FILTER W/ A FIXED 6.0 SECOND TIME CONSTANT TO THE AVERAGE NEUTRON FLUX LEVEL.

BASES

33

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.b. Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High

34

349

The Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. ~~The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor.~~ The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows the setpoint is reduced proportional to the reduction in power experienced as core flow is reduced with a fixed control rod pattern) but is clamped at an upper limit that is always lower than the Average Power Range Monitor ~~Fixed~~ Neutron Flux — High Function Allowable Value.

36

INSERT C

35

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power — High Function~~ provides protection against transients where THERMAL POWER increases slowly (such as the loss of feedwater heating event) and protects the fuel cladding integrity by ensuring that the MCPR SL is not exceeded. During these events, the THERMAL POWER increase does not significantly lag the neutron flux response and, because of a lower trip setpoint, will initiate a scram before the high neutron flux scram. For rapid neutron flux increase events, the THERMAL POWER lags the neutron flux and the Average Power Range Monitor ~~Fixed~~ Neutron Flux — High Function will provide a scram signal before the Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High Function setpoint is exceeded.

37

38

~~The APRM System is divided into two groups of channels with three APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one Average Power Range Monitor channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Flow Biased Simulated Thermal Power — High, with two channels in each trip system arranged in one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

39

(continued)

INSERT C:

¶

A note is included, applicable when the plant is in single recirculation loop operation per LCO 3.4.1, which requires a change in the Allowable Value equation. The Allowable Value is established to conservatively bound the inaccuracy created in the core flow/drive flow relationship due to back flow in the jet pumps associated with the inactive recirculation loop. This adjusted Allowable Value thus maintains thermal margins essentially unchanged from those for two-loop operation.

~~limit~~

~~No specific safety analyses take credit for the Average Power Range Monitor Simulated Thermal Power—High Function; however, it~~

The Average power Range Monitor Simulated Thermal Power High Function

40

uses one total drive flow signal representative of

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High (continued)

core 41

Each APRM channel ~~receives two independent, redundant flow signals representative of~~ total ~~recirculation driving~~ flow. The total ~~recirculation driving flow signals are generated by four flow units, two of which supply signals to the trip system A APRMs, while the other two supply signals to the trip system B APRMs. Each flow unit signal is provided by summing the flow signals from the two recirculation loops. These redundant flow signals are sensed from four pairs of elbow taps, two in each recirculation loop. To obtain the most conservative reference signals under single failure conditions, the total flow signals from the two flow units (associated with a trip system as described above) are routed to a low auction circuit associated with each APRM. Each APRM's circuit selects the lower of the two flow unit signals for use as the reference for that particular APRM. Each required Average Power Range Monitor Flow Biased Simulated Thermal Power — High channel only requires an input from one OPERABLE flow unit, since the individual APRM channel will perform the intended function with only one OPERABLE flow unit input. However, in order to maintain single failure criteria as described above for the Function, at least one required Average Power Range Monitor Flow Biased Simulated Thermal Power — High channel in each trip system must be capable of maintaining an OPERABLE flow unit signal in the event of a failure of an auction circuit, or a flow unit, in the associated trip system (e.g., if a flow unit is inoperable, one of the two required Average Power Range Monitor Flow Biased Simulated Thermal Power — High channels in the associated trip system must be considered inoperable).~~ 43

42 drive flow signal is generated by the flow processing logic, part of the APRM channel, by summing up the flow calculated from two flow transmitter signal inputs, one from each of the two recirculation loop flows. The flow processing logic OPERABILITY is part of the APRM channel OPERABILITY requirements for this Function.

44

No specific safety analyses take direct credit for the Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High Function. Originally, the clamped Allowable Value was based on analyses that took credit for the Average 45 Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power — High Function for the mitigation of the loss of feedwater heater event. However, the current methodology for this event is based on a steady state analysis that allows power to increase beyond the clamped Allowable Value. Therefore,

(continued)

BASES

46

6 seconds is based on the fuel heat transfer dynamics and provides a signal that is proportional to the THERMAL POWER. The THERMAL POWER time constant is stored as a digital value in the firmware of the Average Power Range Monitor System.

See Section 2, Average Power Range Monitor, for additional information.

In accordance with the NRC Safety Evaluation for Amendment TBD (Ref. 20), the Average Power Range Monitor Simulated Thermal Power – High Function is not LSSS SL-related.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power – High (continued) 45a

applying a clamp is conservative. The THERMAL POWER time constant ~~of ≤ 7 seconds is based on the fuel heat transfer dynamics and provides a signal that is proportional to the THERMAL POWER.~~

The Average Power Range Monitor ~~Flow Biased~~ Simulated Thermal Power – High Function is required to be OPERABLE in MODE 1 when there is the possibility of generating excessive THERMAL POWER and potentially exceeding the SL applicable to high pressure and core flow conditions (MCPR SL). During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

47

2.c. Average Power Range Monitor ~~Fixed~~ Neutron Flux – High

~~The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases.~~ 48 The Average Power Range Monitor ~~Fixed~~ Neutron Flux – High Function is capable of generating a trip signal to prevent fuel damage or excessive Reactor Coolant System (RCS) pressure. For the overpressurization protection analyses of References 2 and 3, the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High Function is assumed 49 to terminate the main steam isolation valve (MSIV) closure event and, along with the safety/relief valves (SRVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 9) takes credit for the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High 50 Function to terminate the CRDA.

~~The APRM System is divided into two groups of channels with three APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Fixed Neutron Flux – High with two channels in each trip system arranged in a one-out of two logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide~~ 51

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor ~~Fixed~~ Neutron Flux – High (continued)

~~adequate coverage of the entire core, at least 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.~~

51a

In accordance with the guidance of Regulatory Issue Summary 2006-17 (Ref. 19) and the NRC Safety Evaluation for Amendment TBD (Ref. 20), the Average Power Range Monitor Neutron Flux - High Function is LSSS SL-related.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

52

~~The Average Power Range Monitor Fixed Neutron Flux – High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. The Average Power Range Monitor Fixed Neutron Flux – High Function is assumed in the CRDA analysis (Ref. 9) that is applicable in MODE 2. However, the Average Power Range Monitor Neutron Flux – High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Monitor Fixed Neutron Flux – High Function is not required in MODE 2.~~

53 54 (Setdown)

55b

See Section 2, Average Power Range Monitor, for additional information.

55c

Range 55a

2.d. Average Power Range Monitor – Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than "Operate," an APRM module is unplugged, or ~~the APRM has too few LPRM inputs (< 14),~~ an ~~inoperative trip signal will be received by the RPS, unless the APRM is bypassed. Since only one APRM in each trip system may be bypassed, only one APRM in each trip system may be inoperable without resulting in an RPS trip signal.~~ This Function was not specifically credited in the accident analysis, but it is retained for the overall redundancy and diversity of the RPS as required by the NRC approved licensing basis.

57

the automatic self-test system detects a critical fault with the APRM channel, an Inop trip is sent to all four voter channels. Inop trips from two or more non-bypassed APRM channels result in a trip output from all four voter channels to their associated trip system.

58

~~Four channels of Average Power Range Monitor – Inop with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal.~~

(continued)

(59)

See Section 2, Average Power Range Monitor, for additional information

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

2.d. Average Power Range Monitor – Inop (continued)

There is no Allowable Value for this Function.

This Function is required to be OPERABLE in the MODES where the APRM Functions are required.

Insert E

(60)

3. Reactor Vessel Steam Dome Pressure – High

An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes the neutron flux and THERMAL POWER transferred to the reactor coolant to increase, which could challenge the integrity of the fuel cladding and the RCPB. No specific safety analysis takes direct credit for this Function. However, the Reactor Vessel Steam Dome Pressure – High Function initiates a scram for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power. For the overpressurization protection analyses of References 2 and 3, the reactor scram (the analyses conservatively assume scram on the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High signal, not the Reactor Vessel Steam Dome Pressure – High signal), along with the SRVs, limits the peak RPV pressure to less than the ASME Section III Code limits.

(59)

High reactor pressure signals are initiated from four pressure switches that sense reactor pressure. The Reactor Vessel Steam Dome Pressure – High Allowable Value is chosen to provide a sufficient margin to the ASME Section III Code limits during the event.

Four channels of Reactor Vessel Steam Dome Pressure – High Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. The Function is required to be OPERABLE in MODES 1 and 2 since the RCS is pressurized and the potential for pressure increase exists.

(continued)

Insert E

2.e. 2-Out- Of-4 Voter

The 2-Out-Of-4 Voter Function provides the interface between the APRM Functions, including the OPRM Upscale Function, and the final RPS trip system logic. As such, it is required to be OPERABLE in the MODES where the APRM Functions are required and is necessary to support the safety analysis applicable to each of those Functions. Therefore, the 2-Out-Of-4 Voter Function needs to be OPERABLE in MODES 1 and 2.

All four-voter channels are required to be OPERABLE. Each voter channel includes self-diagnostic functions. If any voter channel detects a critical fault in its own processing, a trip is issued from that voter channel to the associated trip system.

The 2-Out-Of-4 Voter Function votes APRM Functions 2.a, 2.b, and 2.c independently of Function 2.f. The voter also includes separate outputs to RPS for the two independently voted sets of Functions, each of which is redundant (four total outputs). The voter Function 2.e must be declared inoperable if any of its functionally is inoperable. ~~However, due to the independent voting of APRM trips, and the redundancy of outputs there may be conditions where the voter Function 2.e is inoperable, but trip capability for one or more of the other APRM Functions through that voter is still maintained. This may be considered when determining the condition of other APRM Functions resulting from partial inoperability of the Voter Function 2.e.~~

There is no Allowable Value for this Function.

2.f. Oscillation Power Range Monitor (OPRM) Upscale

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit (SL) due to anticipated thermal-hydraulic power oscillations.

References 15, 16, and 17 describe three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. All three are implemented in the OPRM Upscale Function, but the safety analysis takes credit only for the period based detection algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. OPRM Upscale Function OPERABILITY for Technical Specification purposes is based only on the period based detection algorithm.

The OPRM Upscale Function receives input signals from the local power range monitors (LPRMs) within the reactor core, which are combined into “cells” for evaluation by the OPRM algorithms.

Automatic trip is enabled when THERMAL POWER, as indicated by the APRM Simulated Thermal Power, is greater than or equal to the value specified in the COLR and core flow, as indicated by recirculation drive flow, is less than the value specified in the COLR. Within this operating region actual thermal-hydraulic oscillations may occur. The OPRM Upscale Function is required to be OPERABLE when the power is greater than or equal to the OPRM OPERABLE value specified in the COLR. This is the region of power-flow operation where anticipated events could lead to thermal-hydraulic instability and related neutron flux oscillations. The lower bound, as noted in the COLR, is chosen to provide margin in the unlikely event of loss of feedwater heating while the plant is operating below the automatic OPRM Upscale trip enable point. Loss of feedwater heating is the only identified event that could cause reactor power to increase into the region of concern without operator action.

specified in the COLR

An OPRM Upscale trip is issued from an APRM channel when the period based detection algorithm in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with period confirmations and relative cell amplitude exceeding ~~specified~~ setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel trip. An OPRM Upscale trip is also issued from the channel if either the growth rate or amplitude based algorithms detect growing oscillatory changes in the neutron flux for one or more cells in that channel.

Three of the four channels are required to be operable. Each channel is capable of detecting thermal-hydraulic instabilities, by detecting the related neutron flux oscillations, and issuing a trip signal before the MCPR SL is exceeded. There is no allowable value for this function.

The cycle-specific thermal-hydraulic detection algorithms trip settings are nominal settings determined applying the stability analysis licensing methodology (Refs. 15, 16 and 17) developed by the BWR Owners Group and General Electric. There is no Allowable Value for this Function. The settings are not traditional instrumentation setpoints determined under an instrument setpoint methodology. Since the settings may vary cycle-to-cycle, a note indicates the OPRM Upscale Function trip settings, i.e., the period based detection algorithm, are specified in the COLR. In accordance with the NRC Safety Evaluation for Amendment TBD (Ref. 20), the OPRM Upscale Function is not LSSS SL-related.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

C2

5. Main Steam Isolation Valve – Closure (continued)

are completely closed in anticipation of the complete loss of the normal heat sink and subsequent overpressurization transient. However, for the overpressurization protection analyses of References 2 and 3, the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High Function, along with the SRVs, limits the peak RPV pressure to less than the ASME Code limits. That is, the direct scram on position switches for MSIV closure events is not assumed in the overpressurization analysis. Additionally, MSIV closure is assumed in the transients analyzed in Reference 5 (e.g., low steam line pressure, manual closure of MSIVs, high steam line flow). The reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the ECCS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

MSIV closure signals are initiated from position switches located on each of the eight MSIVs. Each MSIV has two position switches; one inputs to RPS trip system A while the other inputs to RPS trip system B. Thus, each RPS trip system receives an input from eight Main Steam Isolation Valve – Closure channels, each consisting of one position switch. The logic for the Main Steam Isolation Valve – Closure Function is arranged such that either the inboard or outboard valve on three or more of the main steam lines (MSLs) must close in order for a scram to occur. In addition, certain combinations of valves closed in two lines will result in a half-scram.

The Main Steam Isolation Valve – Closure Allowable Value is specified to ensure that a scram occurs prior to a significant reduction in steam flow, thereby reducing the severity of the subsequent pressure transient.

Sixteen channels of the Main Steam Isolation Valve – Closure Function with eight channels in each trip system are required to be OPERABLE to ensure that no single instrument failure will preclude the scram from this Function on a valid signal. This Function is only required in MODE 1 since, with the MSIVs open and the heat generation rate high, a pressurization transient can occur if the MSIVs close. In MODE 2, the heat generation rate is low enough so that the other diverse RPS functions provide sufficient protection.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Turbine Throttle Valve – Closure (continued)

RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Throttle Valve – Closure channels, each consisting of one valve stem position switch. The logic for the Turbine Throttle Valve – Closure Function is such that three or more TTVs must close to produce a scram. In addition, certain combinations of two valves closed will result in a half-scram.

This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function.

The Turbine Throttle Valve – Closure Allowable Value is selected to detect imminent TTV closure thereby reducing the severity of the subsequent pressure transient.

Eight channels of Turbine Throttle Valve – Closure Function, with four channels in each trip system, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function if any three TTVs should close. This Function is required, consistent with analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure – High and the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High Functions are adequate to maintain the necessary safety margins.

9. Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low

Fast closure of the TGVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated on TGV fast closure in anticipation of the transients that would result from the closure of these valves. The Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low Function is the primary scram signal for the generator load rejection event analyzed in Reference 5. For

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

9. Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low (continued)

this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the EOC-RPT System, ensures that the MCPR SL is not exceeded.

Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low signals are initiated by the digital-electro hydraulic fluid pressure at each governor valve. There is one pressure switch associated with each governor valve, the signal from each switch being assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening the turbine bypass valves may affect this Function. The basis for the setpoint of this automatic bypass is identical to that described for the Turbine Throttle Valve – Closure Function.

The Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low Allowable Value is selected high enough to detect imminent TGV fast closure.

Four channels of Turbine Governor Valve Fast Closure, Trip Oil Pressure – Low Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. This Function is not required when THERMAL POWER is < 30% RTP since the Reactor Vessel Steam Dome Pressure – High and the Average Power Range Monitor ~~Fixed~~ Neutron Flux – High Functions are adequate to maintain the necessary safety margins.

10. Reactor Mode Switch – Shutdown Position

The Reactor Mode Switch – Shutdown Position Function provides signals, via the manual scram logic channels, that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

11. Manual Scram (continued)

Four channels of Manual Scram with two channels in each trip system arranged in a one-out-of-two logic, are available and required to be OPERABLE in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn.

ACTIONS

A Note has been provided to modify the ACTIONS related to RPS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable RPS instrumentation channels provide appropriate compensatory measures for separate, inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable RPS instrumentation channel.

A.1 and A.2

Because of the diversity of sensors available to provide trip signals and the redundancy of the RPS design, an allowable out of service time of 12 hours has been shown to be acceptable (Ref. 11) to permit restoration of any inoperable channel to OPERABLE status. However, this out of service time is only acceptable provided the associated Functions inoperable channel is in one trip system and the Function still maintains RPS trip capability (refer to Required Actions B.1, B.2, and C.1 Bases.) If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel or the associated trip system must be placed in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable channel in trip (or the associated trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not

and 14

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

desired to place the channel (or trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), Condition D must be entered and its Required Action taken.

66 Insert F

B.1 and B.2

Condition B exists when, for any one or more Functions, at least one required channel is inoperable in each trip system. In this condition, provided at least one channel per trip system is OPERABLE, the RPS still maintains trip capability for that Function, but cannot accommodate a single failure in either trip system.

Required Actions B.1 and B.2 limit the time the RPS scram logic for any Function would not accommodate single failure in both trip systems (e.g., one-out-of-one and one-out-of-one arrangement for a typical four channel Function). The reduced reliability of this logic arrangement was not evaluated in Reference 11 for the 12 hour Completion Time. Within the 6 hour allowance, the associated Function will have all required channels either OPERABLE or in trip (or in any combination) in one trip system.

Completing one of these Required Actions restores RPS to an equivalent reliability level as that evaluated in Reference ~~11~~ and 14, which justified a 12 hour allowable out of service time as presented in Condition A. The trip system in the more degraded state should be placed in trip or, alternatively, all the inoperable channels in that trip system should be placed in trip (e.g., a trip system with two inoperable channels could be in a more degraded state than a trip system with four inoperable channels, if the two inoperable channels are in the same Function while the four inoperable channels are all in different Functions). The decision as to which trip system is in the more degraded state should be based on prudent judgment and current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or recirculation pump trip, it is permissible to place the other trip system or its inoperable channels in trip.

(continued)

Insert F

As noted, Action A.2 is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Inoperability of one required APRM channel affects both trip systems. For that condition, Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of more than one required APRM channel of the same trip function results in loss of trip capability and entry into Condition C, as well as entry into Condition A for each channel.

BASES

ACTIONS

B.1 and B.2 (continued)

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

08

Insert G

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve–Closure), this would require both trip systems to have each channel associated with the MSIVs in three MSLs (not necessarily the same MSLs for both trip systems), OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Throttle Valve–Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

(continued)

channels for which condition B applies. For an inoperable APRM channel,

Insert G

As noted, Condition B is not applicable for APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f. Inoperability of an APRM channel affects both trip systems and is not associated with a specific trip system as are the APRM 2-out-of-4 voter and other non-APRM ~~channel,~~ Required Action A.1 must be satisfied, and is the only action (other than restoring OPERABILITY) that will restore capability to accommodate a single failure. Inoperability of a Function in more than one required APRM channel results in loss of trip capability for that Function and entry into Condition C, as well as entry into Condition A for each channel. Because Conditions A and C provide Required Actions that are appropriate for the inoperability of APRM Functions 2.a, 2.b, 2.c, 2.d, or 2.f, and these functions are not associated with specific trip systems as are the APRM 2-out-of-4 voter and other non-APRM channels, Condition B does not apply.

BASES

ACTIONS (continued)

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C, and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

69

E.1, F.1, ~~and~~ G.1, and J.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition, the Completion ~~Time~~ Times of Required ~~Action E.1 is~~ Actions E.1 and J.1 are consistent with the Completion Time provided in LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)."

70 71

H.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by immediately initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are, therefore, not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

72

Insert H

(continued)

Insert H

I.1

If OPRM Upscale trip capability is not maintained, Condition I exists. Reference 14 justified use of alternate methods to detect and suppress oscillations for a limited period of time. The alternate methods are procedurally established consistent with the guidelines indentified in Reference 18 requiring manual operator action to scram the plant if certain predefined events occur. The 12 hour allowed action time is based on engineering judgment to allow orderly transition to the alternate methods while limiting the period of time during which no automatic or alternate detect and suppress trip capability is formally in place. Based on the small probability of an instability event occurring at all, the 12 hours is judged to be reasonable.

I.2

The alternate method to detect and suppress oscillations implemented in accordance with I.1 was evaluated (Reference 14) based on use up to 120 days only. The evaluation, based on engineering judgment, concluded that the likelihood of an instability event that could not be adequately handled by the alternate methods during this 120 day period was negligibly small. The 120 day period is intended to be an outside limit to allow for the case where design changes or extensive analysis might be required to understand or correct some unanticipated characteristic of the instability detection algorithms or equipment. This action is not intended and was not evaluated as a routine alternative to returning failed or inoperable equipment failure or inoperability is expected to normally be accomplished with the completion times allowed for Actions for Conditions A and B.

A note is provided to indicate that LCO 3.0.4 is not applicable. The intent of that note is to allow plant startup while operating within the 120-day completion time for action I.2. The primary purpose of this exclusion is to allow an orderly completion of design and verification activities, in the event of a required design change, without undue impact on plant operation.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.1 (continued)

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

SR 3.3.1.1.2

73a

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. ~~LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate within 2% RTP of calculated MFLPD.~~ The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.7.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.3

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.3 is not required to be performed when entering MODE 2 from MODE 1 since testing of the MODE 2 required IRM ~~and APRM~~ Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average unavailability over the Frequency interval and is based on reliability analysis (Ref. 11).

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended Function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 11. (The Manual Scram Functions CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions Frequencies.)

SR 3.3.1.1.5 and SR 3.3.1.1.6

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status.

The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a region without adequate neutron flux

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.8 and SR 3.3.1.1.13

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

74

~~For Function 2.b, the CHANNEL FUNCTIONAL TEST includes the adjustment of the APRM channel to conform to a calibrated flow signal. This ensures that the total loop drive flow signals from the flow unit used to vary the setpoint are appropriately compared to an injection test flow signal to verify the flow signal trip setpoint and, therefore, the APRM Function accurately reflects the required setpoint as a function of flow. If the flow signal trip setpoint is not within the appropriate limit, the APRMs that receive an input from the inoperable flow unit must be declared inoperable.~~

75

The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 11. The 24 month Frequency of SR 3.3.1.1.13 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

76

~~SR 3.3.1.1.9 and~~ SR 3.3.1.1.10

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

77

For the APRM Simulated Thermal Power - High Function, this SR also includes calibrating the associated recirculation loop flow channel.

Note 1 states that neutron detectors are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day

(continued)

BASES

SURVEILLANCE REQUIREMENTS

78 ~~SR 3.3.1.1.9 and~~ SR 3.3.1.1.10 (continued)

calorimetric calibration (SR 3.3.1.1.2) and the 1130 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note is provided that requires the ~~APRM and~~ IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. 79 Testing of the MODE 2 ~~APRM and~~ IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or moveable links. 80 This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. ~~The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.~~ 81a The Frequency of SR 3.3.1.1.10 is based on the assumption of an 18 month calibration interval for Functions 1, 3, 4, ~~1 through 4,~~ 6, 7, and 9 through 11 in the determination of the magnitude of equipment drift in the setpoint analysis. 81b

A Frequency of 24 months is assumed for Functions 2, 5 and 8 because the position switches that perform these Functions are not susceptible to instrument drift. 81c

81d

Note (d) requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is not the NTSP but is conservative with respect to the Allowable Value. For digital channel components, no as-found tolerance or as-left tolerance can be specified. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. Any nonconformance will be entered into the Corrective Action Program which will ensure required review and documentation of the condition for continued OPERABILITY.

Note (e) requires that the as-left setting for the instrument be returned to the NTSP. If the as-left instrument setting cannot be returned to the NTSP, then the instrument channel shall be declared inoperable.

SR 3.3.1.1.11

~~The Average Power Range Monitor Flow Biased Simulated Thermal Power High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant must be verified to ensure that the channel is accurately reflecting the desired parameter.~~ Not used. 82

~~The Frequency of 18 months is based on engineering judgment and reliability of the components.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.14 (continued)

Surveillance was performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

83

Insert K

SR 3.3.1.1.15

This test may be performed in one measurement or in overlapping segments, with verification that all components are tested.

84

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 12.

~~As noted (Note 1), neutron detectors for Function 2 are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. In addition, Note 2 states that channel sensors for Functions 3 and 4 are excluded and therefore, it is not required to quantitatively measure the sensor response time to satisfy the requirement to verify RPS RESPONSE TIME. This is acceptable since the sensor response time can be qualitatively verified by other methods (Ref. 13). If the response time of the sensor is not quantitatively measured, the acceptance criteria must be reduced by the time assumed for sensor response in the design analyses, as verified by statistical analyses or vendor data.~~

85

80

Insert L

RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. Note 2 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. Therefore, staggered testing results in response time verification of these devices every 24 months. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

86

Insert M

occurrences

(continued)

86

Insert K

The LOGIC SYSTEM FUNCTIONAL TEST for APRM Function 2.e simulates APRM and OPRM trip conditions at the 2-out-of-4 voter channel inputs to check all combinations of two tripped inputs to the 2-out-of-4 logic in the voter channels and APRM related redundant RPS relays.

The initiation of the input to the RPS logic commences in the 2 out of 4 voter as a vote either for the APRM UPSC/Inop or OPRM UPSC/Inop. The APRM modules are not divisional and do not provide a direct input to RPS.

Insert L

RPS RESPONSE TIME for the APRM 2-out-of-4 Voter Function (2.e) includes the output relays of the voter and the associated RPS relays and contactors. (The digital portion of the APRM and 2-out-of-4 voter channels are excluded from RPS RESPONSE TIME testing because self-testing and calibration checks the time base ~~is adequate to assure required response times are met.~~ Confirmation of the time base is adequate to assure required response times are met. Neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.)

of the digital electronics.

The staggered test basis will test both the APRM and the OPRM outputs of the 2-out-of-4 voter during each iteration of the surveillance. Each iteration will also test both the "X" and "Y" outputs of the voter. Each successive test will alternate the RPS divisions. Each successive test on the specific voter, every 4th test, will test the opposite "X" and "Y" output from the voter. This will accomplish alternating APRM and OPRM and "X" and "Y" outputs of the voter in a specific test while alternating RPS divisions during subsequent tests.

Insert M

SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. For the APRM Functions, this test supplements the automatic self-test functions that operate continuously in the APRM and voter channels. The APRM CHANNEL FUNCTIONAL TEST covers the APRM channels (including recirculation flow processing – applicable to Function 2.b only), the 2-out-of-4 voter channels, and the interface connections into the RPS trip system from the voter channels. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 184 day Frequency of SR 3.3.1.1.17 is based on the reliability analysis of Reference 14. (NOTE: The actual voting logic of the 2-out-of 4 Voter Function is tested as part of SR 3.3.1.1.14.)

A Note is provided for Function 2.a that requires this SR to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM Function cannot be performed in MODE 1 without utilizing jumpers or lifted leads. This Note Frequency is not met per SR 3.02.

SR 3.3.1.1.17

This SR ensures that scrams initiated from OPRM Upscale Function (Function 2.f) will not be inadvertently bypassed when APRM Simulated Thermal Power is greater than or equal to the value specified in the COLR and recirculation drive flow is less than the value specified in the COLR. This normally involves confirming the bypass setpoints, which are considered to be nominal values as discussed in Reference 21. The actual surveillance ensures that the OPRM Upscale Function is enabled (not bypassed) for the correct values of APRM Simulated Thermal Power and recirculation drive flow. Other surveillances ensure that the APRM Simulated Thermal Power and recirculation flow properly correlate with THERMAL POWER (SR 3.3.1.1.2) and core flow (SR 3.3.1.1.10), respectively.

If any bypass setpoint is nonconservative (i.e., the OPRM Upscale Function is bypassed when APRM Simulated Thermal Power is greater than or equal to and recirculation drive flow is less than the values in the COLR), then the affected channel is considered inoperable for the OPRM Upscale Function. Alternatively, the bypass setpoint may be adjusted to place the channel in a conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

BASES (continued)

REFERENCES

1. FSAR, Section 7.2.
2. FSAR, Section 5.2.2.
3. Columbia Generating Station Calculation NE-02-94-66, Revision 0, November 13, 1995.
4. FSAR, Section 6.3.3.
5. FSAR, Chapter 15.
6. 10 CFR 50.36(c)(2)(ii).
7. FSAR, Section 15.4.1.
8. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.
9. FSAR, Section 15.4.9.
10. Letter, P. Check (NRC) to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980.
11. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.
12. Licensee Controlled Specifications Manual.
13. NEDO 32291-A, "System Analyses for Elimination of Selected Response Time Testing Requirements, October 1995.

INSERT C1 89

INSERT C1:

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

15. NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.

16. NEDO-31960-A, Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.

17. NEDO-32465-A, "BWR Owners' Group Long-Term Stability Detect and Suppress Solutions Licensing Basis Methodology And Reload Applications," March 1996.

18. Letter, LA England (BWROG) to MJ Virgilio, "BWR Owners' Group Guidelines for Stability Interim Corrective Action", June 6, 1994.

19. U.S. NRC Regulatory Issue Summary 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006.

20. Amendment No. TBD, "Issuance of Amendment Re: License Amendment Request In Support of PRNM / ARTS / MELLLA," dated TBD. (ADAMS Accession No. TBD)

21. BWROG Letter 96113, K. P. Donovan (BWROG) to L.E. Phillips (NRC), "Guidelines for Stability Option III 'Enable Region' (TAC M92882)," dated September 17, 1996

BASES

APPLICABLE SAFETY ANALYSES (continued)

System (RPS) Instrumentation," Intermediate Range Monitor (IRM) Neutron Flux High and Average Power Range Monitor (Setdown) (APRM) Neutron Flux – High, ~~Setdown~~ Functions; and LCO 3.3.2.1, "Control Rod Block Instrumentation." 90

The SRMs have no safety function and are not assumed to function during any design basis accident or transient analysis. However, the SRMs provide the only on scale monitoring of neutron flux levels during startup and refueling. Therefore, they are being retained in the Technical Specifications.

LCO

During startup in MODE 2, three of the four SRM channels are required to be OPERABLE to monitor the reactor flux level prior to and during control rod withdrawal, to monitor subcritical multiplication and reactor criticality, and to monitor neutron flux level and reactor period until the flux level is sufficient to maintain the IRMs on Range 3 or above. All channels but one are required in order to provide a representation of the overall core response during those periods when reactivity changes are occurring throughout the core.

In MODES 3 and 4, with the reactor shut down, two SRM channels provide redundant monitoring of flux levels in the core.

In MODE 5, during a spiral offload or reload, an SRM outside the fueled region is not required to be OPERABLE, since it is not capable of monitoring neutron flux in the fueled region of the core. Thus, CORE ALTERATIONS are allowed in a quadrant with no OPERABLE SRM in an adjacent quadrant, provided the Table 3.3.1.2-1, footnote (b), requirement that the bundles being spiral reloaded or spiral offloaded are all in a single fueled region containing at least one OPERABLE SRM is met. Spiral reloading and offloading encompass reloading or offloading a cell on the edges of a continuous fueled region (the cell can be reloaded or offloaded in any sequence).

In nonspiral routine operations, two SRMs are required to be OPERABLE to provide redundant monitoring of reactivity changes occurring in the reactor core. Because of the local nature of reactivity changes during refueling, adequate coverage is provided by requiring one SRM to be OPERABLE in

(continued)

B 3.3 ~~INSTRUMENTATION~~ Deleted (91)

~~B 3.3.1.3 Oscillation Power Range Monitor (OPRM)~~

~~BASES~~

~~BACKGROUND~~

~~General Design Criterion 10 (GDC 10) requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the affects of anticipated operational occurrences. Additionally, GDC 12 requires the reactor core and associated coolant, control, and protection systems to be designed to assure that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The OPRM System provides compliance with GDC 10 and GDC 12, thereby providing protection from exceeding the fuel MCPR safety limit.~~

~~References 1, 2, and 3 describe three separate algorithms for detecting stability related oscillations: the period based detection algorithm, the amplitude based algorithm, and the growth rate algorithm. The OPRM System hardware implements these algorithms in microprocessor based modules. These modules execute the algorithms based on LPRM inputs and generate alarms and trips based on these calculations. These trips result in tripping the Reactor Protection System (RPS) when the appropriate RPS trip logic is satisfied, as described in the Bases for LCO 3.3.1.1, "RPS Instrumentation." Only the period based detection algorithm is used in the safety analysis. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations.~~

~~The period based detection algorithm detects a stability related oscillation based on the occurrence of a fixed number of consecutive LPRM signal period confirmations followed by the LPRM signal amplitude exceeding a specified setpoint. Upon detection of a stability related oscillation, a trip is generated for that OPRM channel.~~

~~(continued)~~

(9)

BASES

~~BACKGROUND (continued)~~

~~The OPRM System consits of 4 OPRM trip channels, each channel consisting of two OPRM modules. Each OPRM module receives input from LPRMs. Each OPRM module also receives input from the Neutron Monitoring System (NMS) average power range monitor (APRM) power and flow signals to automatically enable the trip function of the OPRM module.~~

~~Each OPRM module is continuously tested by a self-test function. On detection of any OPRM module failure, either a trouble alarm or INOP alarm is activated. The OPRM module provides an INOP alarm when the self-test feature indicates that the OPRM module may not be capable of meeting its functional requirements.~~

~~APPLICABLE SAFETY ANALYSES~~

~~It has been shown that BWR cores may exhibit thermal-hydraulic reactor instabilities in high power and low flow portions of the core power to flow operating domain. GDC 10 requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the affects of anticipated operational occurrences. GDC 12 requires assurance that power oscillations which can result in conditions exceeding acceptable fuel design limits are either not possible or can be reliably and readily detected and suppressed. The OPRM System provides compliance with GDC 10 and GDC 12 by detecting the onset of oscillations and suppressing them by initiating a reactor scram. This assures that the MCPR safety limit will not be violated for anticipated oscillations.~~

~~The OPRM Instrumentation satisfies Criteria 3 of the NRC Policy Statement.~~

~~LCO~~

~~Four channels of the OPRM System are required to be OPERABLE to ensure that stability related oscillations are detected and suppressed prior to exceeding the MCPR safety limit. Only one of the two OPRM modules' period based detection algorithm is required for OPRM channel OPERABILITY. The minimum number of LPRMs required OPERABLE to maintain an OPRM channel OPERABLE is consistent with the minimum number of LPRMs required to maintain the APRM system OPERABLE per~~

~~(continued)~~

---

(91)

~~BASES~~

~~LCO (continued)~~

~~LCO 3.3.1.1. The Allowable Value for the OPRM Period Based Algorithm setpoint (Sp) is derived from Analytic Limit corrected for the instrument and calibration errors.~~

~~APPLICABILITY~~

~~The OPRM instrumentation is required to be OPERABLE in order to detect and suppress neutron flux oscillations in the event of thermal-hydraulic instability. As described in References 1, 2, and 3, the power/core flow region protected against anticipated oscillations is defined by THERMAL POWER ≥ 30% RTP and core flow ≤ 60% rated core flow. The OPRM trip is required to be enabled in this region and the OPRM must be capable of enabling the trip function as a result of anticipated transients. Therefore, the OPRM is required to be OPERABLE with THERMAL POWER ≥ 25% RTP. It is not necessary for the OPRM to be OPERABLE with THERMAL POWER < 25% RTP because transients from below this THERMAL POWER are not anticipated to result in power that exceeds 30% RTP.~~

~~ACTIONS~~

~~A Note has been provided to modify the ACTIONS related to the OPRM instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times base on initial entry into the Condition. However, the Required Actions for inoperable OPRM instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable OPRM instrumentation channel.~~

~~A.1, A.2, and A.3~~

~~Because of the reliability and on-line self-testing of the OPRM instrumentation and the redundancy of the RPS design, an allowable out of service time of 30 days has been shown to be acceptable (Reference 7) to permit restoration of any inoperable channel to OPERABLE status. However, this out of~~

~~(continued)~~

(a1)

~~BASES~~

~~ACTIONS~~

~~A.1, A.2, and A.3~~ ~~(continued)~~

~~service time is only acceptable provided the OPRM Instrumentation still maintains OPRM trip capability (refer to Required Actions B.1 and B.2). The remaining OPERABLE OPRM channels continue to provide trip capability (see Condition B) and provide operator information relative to stability activity. The remaining OPRM modules have high reliability. With this high reliability, there is a low probability of a subsequent channel failure within the allowable out of service time. In addition, the OPRM modules continue to perform on-line self-testing and alert the operator if any further system degradation occurs.~~

~~If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the OPRM channel or associated RPS trip system must be place in the tripped condition per Required Actions A.1 and A.2. Placing the inoperable OPRM channel in trip (or the associated RPS trip system in trip) would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the OPRM channel (or RPS trip system) in trip (e.g., as in the case where placing the inoperable channel in trip would result in a full scram), the alternate method of detecting and suppressing thermal hydraulic instability oscillations is required (Required Action A.3). This alternate method is described in Reference 5. It consists of increased operator awareness and monitoring for neutron flux oscillations when operating in the region where oscillations are possible. If indications of oscillation, as described in Reference 5, are observed by the operator, the operator will take the actions described by procedures, which include initiating a manual scram of the reactor.~~

~~B.1~~

~~Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped OPRM channels within the same RPS trip system result in not maintaining OPRM trip capability. OPRM trip capability is considered to be maintained when sufficient OPRM channels are OPERABLE or in trip (or the associated RPS trip system is in trip), such that a valid OPRM signal will generate a~~

~~(continued)~~

(a1)

~~BASES~~

~~ACTIONS~~

~~B.1 (continued)~~

~~trip signal in both RPS trip systems. This would require both RPS trip systems to have one OPRM channel OPERABLE or in trip (or the associated RPS trip system in trip).~~

~~Because of the low probability of the occurrence of an instability, 12 hours is an acceptable time to initiate the alternate method of detecting and suppressing thermal hydraulic instability oscillations described in Action A.3 above. The alternate method of detecting and suppressing thermal hydraulic instability oscillations would adequately address detection and mitigation in the event of instability oscillations. Based on industry operating experience with actual instability oscillation, the operator would be able to recognize instabilities during this time and take action to suppress them through a manual scram. In addition, the OPRM System may still be available to provide alarms to the operator if the onset of oscillations were to occur. Since plant operation is minimized in areas where oscillations may occur, operation without OPRM trip capability is considered acceptable with implementation of the alternate method of detecting and suppressing thermal hydraulic instability oscillations during the period when corrective actions are underway to resolve the inoperability that led to entry into Condition B. One reason this Condition may be used is to provide time to implement a software upgrade in the plant if a common cause software problem is identified.~~

~~C.1~~

~~With any Required Action and associated Completion Time not met, THERMAL POWER must be reduced to < 25% RTP within 4 hours. Reducing THERMAL POWER to < 25% RTP places the plant in a region where instabilities cannot occur. The 4 hours is reasonable, based on operating experience, to reduce THERMAL POWER < 25% RTP from full power conditions in an orderly manner and without challenging plant systems.~~

~~(continued)~~

(a1)

~~BASES~~ ~~(continued)~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.3.1.3.1~~

~~A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 184 days provides an acceptable level of system average availability over the Frequency and is based on the reliability of the channel (Reference 7).~~

~~SR 3.3.1.3.2~~

~~LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the OPRM System. The 1130 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.~~

~~SR 3.3.1.3.3~~

~~The CHANNEL CALIBRATION is a complete check of the instrument loop. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations, consistent with the plant specific setpoint methodology. Calibration of the channel provides a check of the internal reference voltage and the internal processor clock frequency. It also compares the desired trip setpoints with those in processor memory. Since the OPRM is a digital system, the internal reference voltage and processor clock frequency are, in turn, used to automatically calibrate the internal analog to digital converters. The Allowable Values are specified in the (COLR). As noted, neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 1130 MWD/T LPRM calibration using the TIPs (SR 3.3.1.3.2).~~

~~The Frequency of 24 months is based upon the assumption of the magnitude of equipment drift provided by the equipment supplier (Reference 7).~~

~~(continued)~~

(a1)

~~BASES~~

~~SURVEILLANCE REQUIREMENTS (continued)~~

~~SR 3.3.1.3.4~~

~~The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod Operability," and in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function. The OPRM self-test function may be utilized to perform this testing for those components that it is designed to monitor.~~

~~The 24 month Frequency is based on engineering judgment and reliability of the components and Operating experience.~~

~~SR 3.3.1.3.5~~

~~This SR ensures that trips initiated from the OPRM System will not be inadvertently bypassed when THERMAL POWER is ≥ 30% RTP and core flow is ≤ 60% rated core flow. This normally involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the actual setpoints (Reference 7).~~

~~If any bypass channel setpoint is nonconservative (i.e., the OPRM module is bypassed at ≥ 30% RTP and core flow ≤ 60% rated core flow),then the affected OPRM module is considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (Manual Enable). If placed in the Manual Enable condition, this SR is met and the module is considered OPERABLE.~~

~~The Frequency of 24 months is based on engineering judgment and reliability of the components.~~

~~SR 3.3.1.3.6~~

~~This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis (Reference 6). The OPRM self-test function may be utilized to perform this testing for those components it is designed to monitor. The LPRM amplifier~~

~~(continued)~~

~~BASES~~

(91)

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.3.1.3.6 (continued)~~

~~cards inputting to the OPRM are excluded from the OPRM response time testing. The RPS RESPONSE TIME acceptance criteria are included in Reference 8.~~

~~As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time. RPS RESPONSE TIME tests are conducted on a 24 month STAGGERED TEST BASIS. This Frequency is based upon operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.~~

~~REFERENCES~~

~~1. NEDO-31960-A, "BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995 (Sus) June 1991.~~

~~2. NEDO-31960-A, Supplement 1" BWR Owners Group Long-Term Stability Solutions Licensing Methodology," November 1995 (Sus) March 1992.~~

~~3. NRC Letter, A. Thadani to L.A. England, "Acceptance for Referencing of Topical Reports NEDO-31960, Supplement 1, 'BWR Owners Group Long-Term Stability Solutions Licensing Methodology,'" July 12, 1994.~~

~~4. Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors," July 11, 1994.~~

~~5. BWROG Letter BWROG-94079, "Guidelines for Stability Interim Corrective Action," June 6, 1994.~~

~~6. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solution Licensing Basis Methodology and Reload Application," August 1996 & May 1995.~~

~~7. CENPD-400-P, Rev 01, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM)," May 1995.~~

~~8. Licensee Controlled Specification Table 1.3.1.1-1~~

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

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

92

setpoint or an RBM inop condition exists.

INSERT I:

92a

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations (Ref. 1). It is assumed to function to block further control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block ~~setpoint.~~ One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. ~~The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals. One RBM channel averages the signals from LPRM detectors at the A and C positions in the assigned LPRM assemblies, while the other RBM channel averages the signals from LPRM detectors at the B and D positions. Alignment of LPRM assemblies to be used in RBM averaging is controlled by the selection of control rods. The RBM is automatically bypassed and the output set to zero if a peripheral rod is selected or the APRM used to normalize the RBM reading is < 30% RTP. If any LPRM detector assigned to an RBM is bypassed, the computed average signal is automatically adjusted to compensate for the number of LPRM input signals. The minimum number of LPRM inputs required for each RBM~~

(continued)

BASES

BACKGROUND (continued)

~~channel to prevent an instrument inoperative alarm is four when using four LPRM assemblies, three when using three LPRM assemblies, and two when using two LPRM assemblies. Each RBM also receives a recirculation loop flow signal from the APRM flow converters.~~

~~When a control rod is selected, the gain of each RBM channel output is normalized to an assigned APRM channel. The assigned APRM channel is on the same RPS trip system as the RBM channel. The gain setting is held constant during the movement of that particular control rod to provide an indication of the change in the relative local power level. If the indicated power increases above the preset limit, a rod block will occur. In addition, to preclude rod movement with an inoperable RBM, a downscale trip and an inoperable trip are provided.~~

The purpose of the RWM is to control rod patterns during startup and shutdown, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. A prescribed control rod sequence is stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 2). The RWM is a single channel system that provides input into one RMCS rod block circuit.

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

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 3. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. ~~Based on the specified Allowable Values, operating limits are established~~.

92B INSERT I.1

INSERT I.2 92C

The RBM Function satisfies Criterion 3 of Reference 4.

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

INSERT I.3 92E

~~Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for process and all instrument uncertainties, except drift and calibration. The trip setpoints are derived from the analytic limits, corrected for process and all instrument uncertainties, including drift and calibration. The trip setpoints derived in this manner provide adequate protection because all instrumentation uncertainties and process effects are taken into account.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~1. Rod Block Monitor (continued)~~

~~The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 30% RTP and a peripheral control rod is not selected. Below this power level, or if a peripheral control rod is selected, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3)~~.

2. Rod Worth Minimizer

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

The RWM Function satisfies Criterion 3 of Reference 4.

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

Compliance with the BPWS, and therefore OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 10% RTP. When THERMAL POWER is > 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA (Ref. 5). In MODES 3 and 4, all control rods are required to be inserted into the core; therefore, a CRDA cannot occur. In MODE 5, since only a single control rod

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

SURVEILLANCE REQUIREMENTS

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

The Surveillances are modified by a second Note to indicate that when an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 7) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The Frequency of ~~92~~ 184 days is based on reliability analyses (Ref. ~~8~~ 9). 92F

93A

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.2 and SR 3.3.2.1.3

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

SR 3.3.2.1.4

INSERT I.4
93B

~~The RBM is automatically bypassed when power is below a specified value or if a peripheral control rod is selected. The power level is determined from the APRM signals input to each RBM channel. The automatic bypass setpoint must be verified periodically to be < 30% RTP. In addition, it must also be verified that the RBM is not bypassed when a control rod that is not a peripheral control rod is selected (only one non-peripheral control rod is required to be verified). If any bypass setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the APRM channel can be placed in the conservative condition (non-bypass). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.4 (continued)

~~they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 92 day Frequency is based on the actual trip setpoint methodology utilized for these channels.~~

SR 3.3.2.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7.

24 month

The Frequency is based upon the assumption of a ~~92 day~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. (Ref. 9)

94

INSERT I.5:

SR 3.3.2.1.6

The RWM is automatically bypassed when power is above a specified value. The power level is determined from a steam flow signal. The automatic bypass setpoint must be verified periodically to be > 10% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. The Frequency is based on instrument drift analysis and the trip setpoint methodology utilized for the low power setpoint channel.

(continued)

INSERT I:

Each RBM uses all available C level LPRM inputs and half of the available B and D level LPRM inputs for the rod selected. RBM channel A uses the opposite B and D level LPRM inputs from RBM channel B.

The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A simulated thermal power signal from one of the four redundant average power range monitor (APRM) channels supplies a reference signal for one of the RBM channels and a simulated thermal power signal from another of the APRM channels supplies the reference signal to the second RBM channel. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM simulated thermal power is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 1).

When a control rod is selected the initial RBM LPRM averaged value is set and held constant until another rod is selected. Each subsequent RBM LPRM averaged value is normalized to the initial RBM LPRM averaged value (RBM flux). The RBM flux, in percent, is used to provide indication and actuation of automatic functions based on the change in the relative local power level.

INSERT I.1:

The Allowable Values and nominal trip setpoints are established in the COLR because they are confirmed or modified on a cycle-specific basis to support the operating limits established in the COLR.

INSERT I.2:

The Rod Block Monitor Low, Intermediate and High Power Range – Upscale functions (Functions 1a, 1b and 1c, respectively) are Limiting Safety System Settings (LSSS), SL-related, as determined in the NRC Safety Evaluation for Amendment TBD (Ref. 10).

INSERT I.3:

~~Nominal trip setpoints (NTSPs) are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter, the calculated RBM flux (RBM channel signal). When the normalized RBM flux value~~

The Analytic Limits are derived from the limiting values determined from the safety analysis. The Allowable Values are derived from the analytic limits correcting for all process and instrument uncertainties, excluding drift and calibration, using the setpoint methodology specified in the Licensee Controlled Specifications. The Nominal Trip Setpoints (NTSPs) are derived from the Analytic Limits in the same way as the Allowable Values except the NTSPs include drift and calibration uncertainties. The NTSPs thus ensure the setpoints don't exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the NTSP but within its Allowable Value, is acceptable.

For the RBM system there is no drift characteristic since it only performs digital calculations on the digitized input signals from the APRMs. Therefore the NTSP is the LSSS. The RBM also undergoes a normalization process after each rod selection. Therefore the only instrument signal variation that is considered in the setpoint methodology which determines the NTSP for the RBM is the actual process changes that take place between rod selection and rod movement.

~~with no as-left or as-found tolerances since it only performs digital calculations on the digitized input signals provided by the APRMs. For the Rod Block Monitor, which is a digital system with a zero as-found tolerance, the Limiting Trip Setpoint is the NTSP.~~

~~The NTSP (or Limiting Trip Setpoint) is the LSSS since the RBM has no drift characteristic. The RBM Allowable Value demonstrates that the analytic limit would not be exceeded, thereby protecting the safety limit. The trip setpoints and Allowable Values~~

the MCPR limit greater than or equal to the RBM MCPR Limits specified in the COLR

~~determined in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and environment errors are accounted for and appropriately applied for the RBM. There are no margins applied to the RBM nominal trip setpoint calculations which could mask RBM degradation.~~

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 28% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating with ~~MCPR ≥ the cycle and power dependent limits specified in the COLR (RBM MCPR Limit)~~, analyses have shown that no RWE event will result in exceeding the MCPR SL. Therefore, under these conditions, the RBM is also not required to be OPERABLE.

INSERT I.4:
The RBM setpoints are automatically varied as a function of power. The RBM Allowable Values required in Table 3.3.2.1-1, each within a specific power range, are specified in the COLR. The power at which the control rod block Allowable Values automatically change are based on the APRM simulated thermal power input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These control rod block bypass setpoints must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.7. The 24 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

INSERT I.5:
SR 3.3.2.1.5 for RBM Functions 1.a, 1.b and 1.c is modified by two Notes as identified in Table 3.3.2.1-1. These functions, in accordance with the guidance of Regulatory Issue Summary 2006-17 (Ref. 11) and as determined in the NRC Safety Evaluation for Amendment TBD (Ref. 10), are LSSS SL-related.

~~Note (g) requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is not the NTSP but is conservative with respect to the Allowable Value. For digital channel components, no as-found tolerance or as-left tolerance can be specified. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. This nonconformance will be entered into the Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition for continued OPERABILITY.~~

~~Note (h) requires that the as-left setting for the instrument be returned to the NTSP. If the as-left instrument setting cannot be returned to the NTSP, then the instrument channel shall be declared inoperable. The NTSPs and Allowable Values for Rod Block Monitor Functions 1.a, 1.b and 1.c are specified in the COLR. The methodology used to determine the NTSPs is specified in the Licensee Controlled Specifications, a document controlled under 10 CFR 50.59.~~

Note (g) requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is not the NTSP but is conservative with respect to the Allowable Value. For digital channel components, no as-found tolerance or as-left tolerance can be specified. Evaluation of instrument performance will verify that the instrument will continue to behave in accordance with design basis assumptions. The purpose of the assessment is to ensure confidence in the instrument performance prior to returning the instrument to service. Any nonconformance will be entered into the Corrective Action Program which will ensure required review and documentation of the condition for continued OPERABILITY.

Note (h) requires that the as-left setting for the instrument be returned to the NTSP. If the as-left instrument setting cannot be returned to the NTSP, then the instrument channel shall be declared inoperable. The NTSPs and Allowable Values for Rod Block Monitor Functions 1.a, 1.b and 1.c are specified in the COLR.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.7

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

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the 24 month Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.2.1.8

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

REFERENCES

1. FSAR, Section 7.7.1.8.

2. FSAR, Section 7.7.1.10.

3. ~~FSAR, Sections 15.4.1 and 15.4.2~~. 95A

NEDC-33507P, Revision 0, “Energy Northwest Columbia Generating Station APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA),” March 2010.

BASES

REFERENCES (continued)

4. 10 CFR 50.36(c)(2)(ii).

5. FSAR, Section 15.4.9.

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

7. GENE-770-06-1-A, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," December 1992.

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

95B

9. NEDC-32410P, "Nuclear Measurement Analysis and control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function", October 1995.
10 Amendment No. TBD, "Issuance of Amendment Re: License Amendment Request In Support of PRNM / ARTS / MELLLA," dated TBD. (ADAMS Accession No. TBD)
11 U.S. NRC Regulatory Issue Summary 2006 17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Throttle Valve – Closure (continued)

Closure of the TTVs is determined by measuring the position of each throttle valve. While there are two separate position switches associated with each throttle valve, only the signal from one switch for each TTV is used, with each of the four channels being assigned to a separate trip channel. The logic for the TTV – Closure Function is such that two or more TTVs must be closed to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TTV – Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TTV – Closure Allowable Value is selected to detect imminent TTV closure.

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure – High and the Average Power Range Monitor (APRM) ~~Fixed~~ Neutron Flux – High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

TGV Fast Closure, Trip Oil Pressure – Low

Fast closure of the TGVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TGV Fast Closure, Trip Oil Pressure – Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

Fast closure of the TGVs is determined by measuring the DEH fluid pressure at each control valve. There is one pressure switch associated with each control valve, and the signal from each switch is assigned to a separate trip channel. The logic for the TGV Fast Closure, Trip Oil Pressure – Low Function is such that two or more TGVs must be closed

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

TGV Fast Closure, Trip Oil Pressure — Low (continued)

(pressure switch trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 30% RTP. This is normally accomplished automatically by pressure switches sensing turbine first stage pressure; therefore, opening of the turbine bypass valves may affect this Function. Four channels of TGV Fast Closure, Trip Oil Pressure — Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TGV Fast Closure, Trip Oil Pressure — Low Allowable Value is selected high enough to detect imminent TGV fast closure.

This protection is required consistent with the analysis, whenever the THERMAL POWER is ≥ 30% RTP. Below 30% RTP, the Reactor Vessel Steam Dome Pressure — High and the APRM ~~Fixed~~ Neutron Flux — High Functions of the RPS are adequate to maintain the necessary safety margins. The turbine first stage pressure/reactor power relationship for the setpoint of the automatic enable is identical to that described for TTV closure.

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable EOC-RPT instrumentation channel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable (Ref. 2). The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 3), which are analyzed in Chapter 15 of the FSAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 4).

The transient analyses in Chapter 15 of the FSAR have also been performed for single recirculation loop operation (Ref. 4) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. INSERT J 9BA ~~The APLHGR and MCPR setpoints for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."~~

Recirculation loops operating satisfies Criterion 2 of Reference 5.

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied.

(continued)

BASES

LCO (continued)

Alternately, with only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), ~~and~~ MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO ~~(MCPR)")~~ must be applied to allow continued operation.

98B

INSERT J.1

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1 and B.1

With both recirculation loops operating but the flows not matched, the recirculation loops must be restored to operation within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation," as required by Required Action A.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing pump speeds to re-establish forward flow or by tripping the pump.

(continued)

INSERT J:

~~During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Simulated Thermal Power — High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."~~

INSERT J.1:

(MCPR)"), and APRM Simulated Thermal Power - Upscale Allowable Value (LCO 3.3.1.1) ~~must be applied to allow continued operation.~~

The APLHGR and MCPR limits for single loop operation are specified in the COLR. During single recirculation loop operation, modification to the Reactor Protection System(RPS) average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APRM Simulated Thermal Power - High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

BASES

ACTIONS

A.1 and B.1 (continued)

With the requirements of the LCO not met for reasons other than Condition A (e.g., one loop is "not in operation"), the recirculation loops must be restored to operation with matched flows within 4 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e., Required Action A.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status. 98C

Alternatively, if the single loop requirements of the LCO are applied to operating ~~limits,~~ limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 and 4 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

C.1

With the Required Action and associated Completion Time of Condition A or B not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow, 75.95 x 10^6 lbm/hr), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow.

The mismatch is measured in terms of percent of rated recirculation loop drive flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1. FSAR, Sections 6.3 and 15.6.

2. FSAR, Section 6.3.3.7.2.

3. FSAR, Section 5.4.1.

98D

4. FSAR, Section ~~6.A.~~ 6.3.3.8 and 15.0.2.1

5. 10 CFR 50.36(c)(2)(ii).

BASES

APPLICABLE SAFETY ANALYSES (continued)

CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. For SDM tests performed within these defined sequences, the analysis of Reference 1 is applicable. However, for some sequences developed for the SDM testing, the control rod patterns assumed in the safety analysis of Reference 1 may not be met. Therefore, special CRDA analyses, performed in accordance with an NRC approved methodology, are required to demonstrate that the SDM test sequence will not result in unacceptable consequences should a CRDA occur during the testing. For the purpose of this test, the protection provided by the normally required MODE 5 applicable LCOs, in addition to the requirements of this LCO, will maintain normal test operations as well as postulated accidents within the bounds of the appropriate safety analysis (Ref. 1). In addition to the added requirements for the Rod Worth Minimizer (RWM), APRM, and control rod coupling, the notch out mode is specified for out of sequence withdrawals. Requiring the notch out mode limits withdrawal steps to a single notch, which limits inserted reactivity, and allows adequate monitoring of changes in neutron flux, which may occur during the test.

As described in LCO 3.0.7, compliance with Special Operations LCOs is optional, and therefore, no criteria of Reference 2 apply. Special Operations LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective Bases.

LCO

a9

As described in LCO 3.0.7, compliance with this Special Operations LCO is optional. SDM tests may be performed while in MODE 2, in accordance with Table 1.1-1, without meeting this Special Operations LCO or its ACTIONS. For SDM tests performed while in MODE 5, additional requirements must be met to ensure that adequate protection against potential reactivity excursions is available. To provide additional scram protection, beyond the normally required IRMs, the APRMs are also required to be OPERABLE (LCO 3.3.1.1, Functions ~~2.a and 2.d~~ 2.a, 2.d, and 2.e,) as though the reactor were in MODE 2. Because multiple control rods will be withdrawn and the reactor will potentially become critical, the approved control rod withdrawal sequence must be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2), or

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.10.8.1, SR 3.10.8.2, and SR 3.10.8.3

2.a, 2.d, and 2.e,

LCO 3.3.1.1, Functions ~~2.a and 2.d,~~ made applicable in this Special Operations LCO, are required to have applicable Surveillances met to establish that this Special Operations LCO is being met (SR 3.10.8.1). However, the control rod withdrawal sequences during the SDM tests may be enforced by the RWM (LCO 3.3.2.1, Function 2, MODE 2 requirements) or by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. As noted, either the applicable SRs for the RWM (LCO 3.3.2.1) must be satisfied according to the applicable Frequencies (SR 3.10.8.2), or the proper movement of control rods must be verified (SR 3.10.8.3). This latter verification (i.e., SR 3.10.8.3) must be performed during control rod movement to prevent deviations from the specified sequence. These Surveillances provide adequate assurance that the specified test sequence is being followed.

SR 3.10.8.4

Periodic verification of the administrative controls established by this LCO will ensure that the reactor is operated within the bounds of the safety analysis. The 12 hour Frequency is intended to provide appropriate assurance that each operating shift is aware of and verifies compliance with these Special Operations LCO requirements.

SR 3.10.8.5

Coupling verification is performed to ensure the control rod is connected to the control rod drive mechanism and will perform its intended function when necessary. The verification is required to be performed any time a control rod is withdrawn to the "full-out" notch position or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved as well as operating experience related to uncoupling events.

(continued)

Retyped Technical Specifications pages

TABLE OF CONTENTS

1.0	USE AND APPLICATION	
1.1	Definitions	1.1-1
1.2	Logical Connectors	1.2-1
1.3	Completion Times	1.3-1
1.4	Frequency	1.4-1
2.0	SAFETY LIMITS (SLs)	
2.1	SLs	2.0-1
2.2	SL Violations	2.0-1
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY	3.0-1
3.0	SURVEILLANCE REQUIREMENT (SR) APPLICABILITY	3.0-4
3.1	REACTIVITY CONTROL SYSTEMS	
3.1.1	SHUTDOWN MARGIN (SDM)	3.1.1-1
3.1.2	Reactivity Anomalies	3.1.2-1
3.1.3	Control Rod OPERABILITY	3.1.3-1
3.1.4	Control Rod Scram Times	3.1.4-1
3.1.5	Control Rod Scram Accumulators	3.1.5-1
3.1.6	Rod Pattern Control	3.1.6-1
3.1.7	Standby Liquid Control (SLC) System	3.1.7-1
3.1.8	Scram Discharge Volume (SDV) Vent and Drain Valves	3.1.8-1
3.2	POWER DISTRIBUTION LIMITS	
3.2.1	AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)	3.2.1-1
3.2.2	MINIMUM CRITICAL POWER RATIO (MCPR)	3.2.2-1
3.2.3	LINEAR HEAT GENERATION RATE (LHGR)	3.2.3-1
3.3	INSTRUMENTATION	
3.3.1.1	Reactor Protection System (RPS) Instrumentation	3.3.1.1-1
3.3.1.2	Source Range Monitor (SRM) Instrumentation	3.3.1.2-1
3.3.2.1	Control Rod Block Instrumentation	3.3.2.1-1
3.3.2.2	Feedwater and Main Turbine High Water Level Trip Instrumentation	3.3.2.2-1
3.3.3.1	Post Accident Monitoring (PAM) Instrumentation	3.3.3.1-1
3.3.3.2	Remote Shutdown System	3.3.3.2-1
3.3.4.1	End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation	3.3.4.1-1
3.3.4.2	Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation	3.3.4.2-1
3.3.5.1	Emergency Core Cooling System (ECCS) Instrumentation	3.3.5.1-1

(continued)

3.3	INSTRUMENTATION (continued)	
3.3.5.2	Reactor Core Isolation Cooling (RCIC) System Instrumentation	3.3.5.2-1
3.3.6.1	Primary Containment Isolation Instrumentation	3.3.6.1-1
3.3.6.2	Secondary Containment Isolation Instrumentation	3.3.6.2-1
3.3.7.1	Control Room Emergency Filtration (CREF) System Instrumentation	3.3.7.1-1
3.3.8.1	Loss of Power (LOP) Instrumentation	3.3.8.1-1
3.3.8.2	Reactor Protection System (RPS) Electric Power Monitoring	3.3.8.2-1
3.4	REACTOR COOLANT SYSTEM (RCS)	
3.4.1	Recirculation Loops Operating	3.4.1-1
3.4.2	Jet Pumps	3.4.2-1
3.4.3	Safety/Relief Valves (SRVs) — $\geq$ 25% RTP	3.4.3-1
3.4.4	Safety/Relief Valves (SRVs) — < 25% RTP	3.4.4-1
3.4.5	RCS Operational LEAKAGE	3.4.5-1
3.4.6	RCS Pressure Isolation Valve (PIV) Leakage	3.4.6-1
3.4.7	RCS Leakage Detection Instrumentation	3.4.7-1
3.4.8	RCS Specific Activity	3.4.8-1
3.4.9	Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown	3.4.9-1
3.4.10	Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown	3.4.10-1
3.4.11	RCS Pressure and Temperature (P/T) Limits	3.4.11-1
3.4.12	Reactor Steam Dome Pressure	3.4.12-1
3.5	EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM	
3.5.1	ECCS — Operating	3.5.1-1
3.5.2	ECCS — Shutdown	3.5.2-1
3.5.3	RCIC System	3.5.3-1
3.6	CONTAINMENT SYSTEMS	
3.6.1.1	Primary Containment	3.6.1.1-1
3.6.1.2	Primary Containment Air Lock	3.6.1.2-1
3.6.1.3	Primary Containment Isolation Valves (PCIVs)	3.6.1.3-1
3.6.1.4	Drywell Air Temperature	3.6.1.4-1
3.6.1.5	Residual Heat Removal (RHR) Drywell Spray	3.6.1.5-1
3.6.1.6	Reactor Building-to-Suppression Chamber Vacuum Breakers	3.6.1.6-1
3.6.1.7	Suppression Chamber-to-Drywell Vacuum Breakers	3.6.1.7-1
3.6.2.1	Suppression Pool Average Temperature	3.6.2.1-1
3.6.2.2	Suppression Pool Water Level	3.6.2.2-1
3.6.2.3	Residual Heat Removal (RHR) Suppression Pool Cooling	3.6.2.3-1

(continued)

1.1 Definitions (continued)

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

(continued)

1.1 Definitions (continued)

RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3486 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM)	SDM shall be the amount of reactivity by which the reactor is subcritical or would be subcritical assuming that: a. The reactor is xenon free; b. The moderator temperature is 68°F; and c. All control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM.
STAGGERED TEST BASIS	A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function.
THERMAL POWER	THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify each SLC subsystem manual and power operated valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position or can be aligned to the correct position.	31 days
SR 3.1.7.6	Verify each pump develops a flow rate $\geq$ 41.2 gpm at a discharge pressure $\geq$ 1220 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.7	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.8	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months <u>AND</u> Once within 24 hours after solution temperature is restored within the limits of Figure 3.1.7-1
SR 3.1.7.9	Verify sodium pentaborate enrichment is $\geq$ 44.0 atom percent B-10.	Prior to addition to SLC Tank

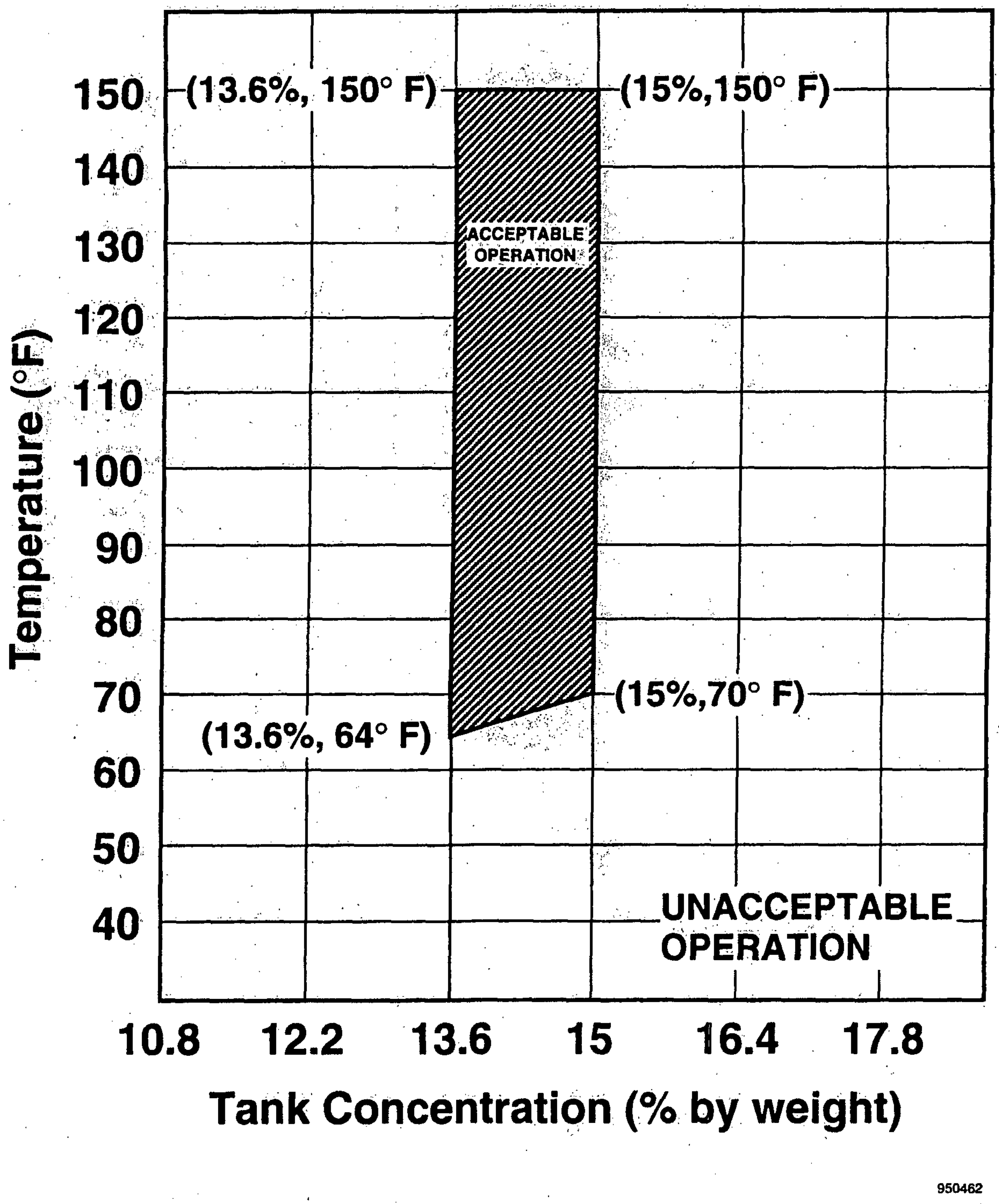


Figure 3.1.7-1 (Page 1 of 1)
Sodium Pentaborate Solution Temperature/Concentration Requirements

3.3 INSTRUMENTATION

3.3.1.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1.1 The RPS instrumentation for each Function in Table 3.3.1.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours
	<u>OR</u>	
	A.2 --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -------------------- Place associated trip system in trip.	12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. --------NOTE-------- Not applicable for Functions 2.a, 2.b, 2.c, 2.d, or 2.f. -------------------- One or more Functions with one or more required channels inoperable in both trip systems.	B.1	Place channel in one trip system in trip.	6 hours
	<u>OR</u>		
	B.2	Place one trip system in trip.	6 hours
C. One or more Functions with RPS trip capability not maintained.	C.1	Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1	Reduce THERMAL POWER to < 30% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1	Be in MODE 2.	6 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1	Be in MODE 3.	12 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1	Initiate alternate method to detect and suppress thermal hydraulic instability oscillations.	12 hours
	<u>AND</u>		
	I.2	---------NOTE-------- LCO 3.0.4 is not applicable --------------------- Restore required channels to OPERABLE.	 120 days
J. Required Action and associated Completion Time of Condition I not met.	J.1	Reduce THERMAL POWER to less than the value specified in the COLR.	4 hours

SURVEILLANCE REQUIREMENTS

-----NOTES-----

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

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	-----NOTE----- Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. ---------- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 25% RTP.	7 days
SR 3.3.1.1.3	-----NOTE----- Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	-------------------NOTE------------------- Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1130 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Deleted	

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 1, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Functions 2.b and 2.f, the recirculation flow transmitters that feed the APRMs are included. --- Perform CHANNEL CALIBRATION.	18 months for Functions 1, 3, 4, 6, 7, and 9 through 11 <u>AND</u> 24 months for Functions 2, 5, and 8
SR 3.3.1.1.11	Deleted.	
SR 3.3.1.1.12	Verify Turbine Throttle Valve — Closure, and Turbine Governor Valve Fast Closure Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is $\geq$ 30% RTP.	18 months
SR 3.3.1.1.13	Perform CHANNEL FUNCTIONAL TEST.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. Channel sensors for Functions 3 and 4 are excluded. 3. For Function 5, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. 4. For Function 2.e, "n" equals 8 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. Testing of APRM and OPRM outputs shall alternate. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS
SR 3.3.1.1.16	------------------NOTE------------------ 1. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 2. For Functions 2.b and 2.f, the CHANNEL FUNCTIONAL TEST includes the recirculation flow input processing, excluding the flow transmitters. -- Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.17	Verify the Oscillation Power Range Monitor (OPRM) is not bypassed when APRM Simulated Thermal Power is greater than or equal to the value specified in the COLR <u>AND</u> recirculation drive flow is less than the value specified in the COLR.	24 months

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.10 SR 3.3.1.1.14	$\leq$ 122/125 divisions of full scale
	5(a)	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.10 SR 3.3.1.1.14	$\leq$ 122/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5(a)	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – High (Setdown)	2	3(b)	G	SR 3.3.1.1.1 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.16	$\leq$ 20% RTP
b. Simulated Thermal Power – High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.16	$\leq$ 0.63W + 64.0% RTP and $\leq$ 114.9% RTP (c)

(continued)

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

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(c) $\leq$ 0.63W + 60.8% RTP and $\leq$ 114.9% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Neutron Flux – High	1	3(b)	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10(d)(e) SR 3.3.1.1.16	≤ 120% RTP
d. Inop	1,2	3(b)	G	SR 3.3.1.1.16	NA
e. 2-Out-of-4 Voter	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.16	NA
f. OPRM Upscale	1(f)	3(b)	I	SR 3.3.1.1.1 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.16 SR 3.3.1.1.17	NA (g)
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 1079 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 9.5 inches
5. Main Steam Isolation Valve – Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 12.5% closed

(continued)

(b) Each APRM/OPRM channel provides inputs to both trip systems.

(d) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, and the methodology used to determine this value, is specified in the Licensee Controlled Specifications.

(f) When greater than the RTP value specified in the COLR.

(g) The OPRM Upscale does not have an Allowable Value. The Period Based Detection Algorithm (PBDA) trip setpoints are specified in the COLR.

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Primary Containment Pressure − High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level − High					
a. Transmitter/Trip Unit	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 529 ft 9 inches elevation
8. Turbine Throttle Valve − Closure	≥ 30% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≤ 7% closed
9. Turbine Governor Valve Fast Closure, Trip Oil Pressure − Low	≥ 30% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.12 SR 3.3.1.1.14 SR 3.3.1.1.15	≥ 1000 psig
10. Reactor Mode Switch − Shutdown Position	1,2	2	G	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.13 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	2	G	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	2	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. One or more Reactor Mode Switch – Shutdown Position channels inoperable.	E.1	Suspend control rod withdrawal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

NOTES

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

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

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	184 days

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.2	-------------------NOTE------------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ 10% RTP in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	-------------------NOTE------------------- Not required to be performed until 1 hour after THERMAL POWER is ≤ 10% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.4	-------------------NOTE------------------- Neutron detectors are excluded. -- Verify the RBM is not bypassed: a. Low Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 28% and < 63% RTP and a peripheral control rod is not selected. b. Intermediate Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 63% and < 83% RTP and a peripheral control rod is not selected. c. High Power Range - Upscale Function is not bypassed when APRM Simulated Thermal Power is ≥ 83% RTP and a peripheral control rod is not selected.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.5	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.1.6	Verify the RWM is not bypassed when THERMAL POWER is ≤ 10% RTP.	24 months
SR 3.3.2.1.7	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	24 months
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with BPWS.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range - Upscale	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5(g)(h)	(f)
b. Intermediate Power Range - Upscale	(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5(g)(h)	(f)
c. High Power Range - Upscale	(c)	2	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.5(g)(h)	(f)
d. Inop	(a),(b),(c)	2	SR 3.3.2.1.1	NA
2. Rod Worth Minimizer	1(d),2(d)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.6 SR 3.3.2.1.8	NA

(continued)

(a) APRM Simulated Thermal Power is $\geq$ 28% and < 63% RTP and MCPR is less than the limit specified in the COLR and no peripheral control rod selected.

(b) APRM Simulated Thermal Power is $\geq$ 63% and < 83% RTP and MCPR is less than the limit specified in the COLR and no peripheral control rod selected.

(c) APRM Simulated Thermal Power is $\geq$ 83% and MCPR is less than the limit specified in COLR and no peripheral control rod selected.

(d) With THERMAL POWER $\leq$ 10% RTP.

(f) Allowable Value specified in the COLR.

(g) If the as-found channel setpoint is not the Nominal Trip Setpoint but is conservative to the Allowable Value, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(h) The instrument channel setpoint shall be reset to the Nominal Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoint, and the methodology used to determine this value, is specified in the Licensee Controlled Specifications.

Table 3.3.2.1-1 (page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Mode Switch — Shutdown Position	(e)	2	SR 3.3.2.1.7	NA

(e) Reactor mode switch in the shutdown position.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation provided that the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; and

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors, Simulated Thermal Power - High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop flow mismatch not within limits.	A.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours
B. Requirements of the LCO not met for reasons other than Condition A.	B.1 Satisfy the requirements of the LCO.	4 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> No recirculation loops in operation.	C.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop drive flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated recirculation loop drive flow when operating at < 70% of rated core flow; and b. $\leq$ 5% of rated recirculation loop drive flow when operating at $\geq$ 70% of rated core flow.	24 hours

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test — Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System Instrumentation," MODE 2 requirements for Function 2.a, 2.d, and 2.e of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 2 of Table 3.3.2.1-1, with banked position withdrawal sequence requirements of SR 3.3.2.1.8 changed to require the control rod sequence to conform to the SDM test sequence,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated control rod drive (CRD);

d. All control rod withdrawals during out of sequence control rod moves shall be made in notch out mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 940 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
-----NOTE----- Separate Condition entry is allowed for each control rod. ----------	-----NOTE----- Rod worth minimizer may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow insertion of inoperable control rod and continued operation. ----------	
A. One or more control rods not coupled to its associated CRD.	A.1 Fully insert inoperable control rod.	3 hours
	<u>AND</u>	
	A.2 Disarm the associated CRD.	4 hours
B. One or more of the above requirements not met for reasons other than Condition A.	B.1 Place the reactor mode switch in the shutdown or refuel position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.1	Perform the MODE 2 applicable SRs for LCO 3.3.1.1, Functions 2.a, 2.d, and 2.e of Table 3.3.1.1-1.	According to the applicable SRs
SR 3.10.8.2	-------------------NOTE------------------- Not required to be met if SR 3.10.8.3 satisfied. -- Perform the MODE 2 applicable SRs for LCO 3.3.2.1, Function 2 of Table 3.3.2.1-1.	According to the applicable SRs
SR 3.10.8.3	-------------------NOTE------------------- Not required to be met if SR 3.10.8.2 satisfied. -- Verify movement of control rods is in compliance with the approved control rod sequence for the SDM test by a second licensed operator or other qualified member of the technical staff.	During control rod movement
SR 3.10.8.4	Verify no other CORE ALTERATIONS are in progress.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.10.8.5	Verify each withdrawn control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling
SR 3.10.8.6	Verify CRD charging water header pressure $\geq$ 940 psig.	7 days

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 <u>Annual Radiological Environmental Operating Report</u>

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.2 <u>Radioactive Effluent Release Report</u>

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and the Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

(continued)

5.6 Reporting Requirements (continued)

5.6.3 CORE OPERATING LIMITS REPORT (COLR)

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

1. The APLHGR for Specification 3.2.1;

2. The MCPR for Specification 3.2.2;

3. The LHGR for Specification 3.2.3;

4. The Oscillation Power Range Monitor (OPRM) Instrumentation for Specification 3.3.1.1; and

5. The Rod Block Monitor Instrumentation for Specification 3.3.2.1.

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

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Exxon Nuclear Company

3. EMF-85-74(P) Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Power Corporation

4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation

5. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis," Exxon Nuclear Company

(continued)

LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM / ARTS / MELLLA IMPLEMENTATION
Attachment 5

Sample pages of proposed COLR changes (for information only)

5.0 **Oscillation Power Range Monitor (OPRM) Instrumentation Limits for Use in LCO 3.3.1.1**

5.1 Reactor Protection System (RPS) Instrumentation Setpoints for the OPRM Period Based Detection Algorithm (PBDA) support OPERABILITY for LCO 3.3.1.1. See Technical Specification 3.3.1.1 and the applicable Bases for further application details.

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	TRIP SETPOINT
2 Average Power Range Monitors		
f. OPRM Upscale	(d)	
Amplitude Trip (Sp)		1.15 Peak/Average
Confirmation Count (N2)		16

(d) THERMAL POWER ≥ 20% RTP

5.2 THERMAL POWER for Use in Technical Specification 3.3.1.1, Required Action J.1:

THERMAL POWER < 20% RTP

5.3 OPRM Trip Enable Values for Use in SR 3.3.1.1.17

APRM Simulated Thermal Power	≥ 25% RTP
Recirculation Drive Flow	< 60% rated recirculation drive flow

6.0 Control Rod Block Instrumentation Limits for Use in LCO 3.3.2.1

6.1 Rod Block Monitor instrument setpoints support OPERABILITY for LCO 3.3.2.1. See Technical Specification 3.3.2.1 and the applicable Bases for further application details.

FUNCTION		TRIP SETPOINT	ALLOWABLE VALUE
1 Rod Block Monitor			
a. Low Power Range—Upscale			
	Unfiltered	124.0	124.6
	Filtered	122.8	123.4
b. Intermediate Power Range—Upscale			
	Unfiltered	119.0	119.6
	Filtered	118.0	118.6
c. High Power Range—Upscale			
	Unfiltered	114.0	114.6
	Filtered	113.0	113.6

6.2 Rod Block Monitor (RBM) MCPR limits for use in Technical Specification Table 3.3.2.1-1, Footnotes (a), (b) and (c). See Technical Specification 3.3.2.1 and the applicable Bases for further application details.

THERMAL POWER (% RTP)	RBM MCPR Limit
≥ 28 and < 90	1.73
≥ 90	1.43

Affidavit to withhold proprietary information

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Edward D Schrull, state as follows:

(1) I am the Vice President of Regulatory Affairs, Services Licensing, GE-Hitachi Nuclear Energy Americas LLC (GEH), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in GEH proprietary report, NEDC-33507P, "Energy Northwest Columbia Generating Station APRM / RBM / Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)," Revision 0, dated April 2010. The proprietary information is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] Figures and large equation objects containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH and/or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GEH has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor (BWR). The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 1st day of April 2010.

Edward D. Schrull
Vice President, Regulatory Affairs
Services Licensing
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd.
Wilmington, NC 28401
edward.schrull@ge.com

Safety Analysis in support of ARTS/MELLLA
(non-proprietary version)

HITACHI

GE Hitachi Nuclear Energy

NEDO-33507
Revision 0
Class I
DRF 0000-0100-5300
April 2010

Non-Proprietary Information

ENERGY NORTHWEST

Columbia Generating Station

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

NON-PROPRIETARY INFORMATION NOTICE

This is a non-proprietary version of the document NEDC-33507P, Revision 0, from which the proprietary information has been removed. Portions of the document that have been removed are identified by white space within double square brackets, as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document are furnished for the purposes of supporting ARTS/MELLLA Safety Analysis Report for Energy Northwest Columbia Generating Station. The only undertakings of GEH with respect to information in this document are contained in the contracts between GEH and its customers or participating utilities, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GEH makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

TABLE OF CONTENTS

1.0 INTRODUCTION 1-1
1.1 Background 1-2
1.2 ARTS/MELLLA Bases 1-2
1.3 Average Power Range Monitor Improvements 1-6
1.4 Rod Block Monitor Improvements 1-8
2.0 OVERALL ANALYSIS APPROACH 2-1
3.0 FUEL THERMAL LIMITS 3-1
3.1 Limiting Core-Wide Anticipated Operational Occurrence Analyses 3-1
3.2 Input Assumptions 3-4
3.3 Analyses Results 3-4
3.4 Conclusion 3-7
4.0 ROD BLOCK MONITOR SYSTEM IMPROVEMENTS 4-1
4.1 Current Rod Block Monitor System Description 4-1
4.2 ARTS-Based Rod Block Monitor System Description 4-4
4.3 Rod Withdrawal Error Analysis 4-5
4.4 Filter And Time Delay Settings 4-9
4.5 Rod Block Monitor Operability Requirement 4-11
4.6 Rod Block Monitor Modification Compliance to NRC Regulations and Licensing Topical Reports 4-11
4.7 Conclusion 4-12
5.0 VESSEL OVERPRESSURE PROTECTION 5-1
6.0 THERMAL-HYDRAULIC STABILITY 6-1
6.1 Introduction 6-1
6.2 Stability Option III 6-1
6.3 Backup Stability Protection 6-2
7.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS 7-1
7.1 Conclusions 7-3
8.0 CONTAINMENT RESPONSE 8-1
8.1 Approach/Methodology 8-1
8.2 Assumptions and Initial Conditions 8-3
8.3 Analyses Results 8-4
8.4 Conclusions 8-6
8.5 Reactor Asymmetric Loads 8-7
9.0 REACTOR INTERNALS INTEGRITY 9-1
9.1 Reactor Internal Pressure Differences 9-1
9.2 Acoustic and Flow-Induced Loads 9-1
9.3 RPV Internals Structural Integrity Evaluation 9-2
9.4 Reactor Internals Vibration 9-3
9.5 Conclusion 9-5
10.0 ANTICIPATED TRANSIENT WITHOUT SCRAM 10-1
10.1 Approach/Methodology 10-1
10.2 Input Assumptions 10-2
10.3 Analyses Results 10-3

10.4 Conclusions 10-4
11.0 STEAM DRYER AND SEPARATOR PERFORMANCE 11-1
12.0 HIGH ENERGY LINE BREAK 12-1
13.0 TESTING 13-1
14.0 REFERENCES 14-1
ATTACHMENT A A-1

LIST OF TABLES

Table 1-1 Computer Codes Used for ARTS/MELLLA Analyses 1-9
Table 2-1 Analyses Presented In This Report 2-2
Table 2-2 Applicability of Analyses 2-2
Table 3-1 Base Conditions for ARTS/MELLLA Rated Transient Analyses 3-8
Table 3-2 Base Conditions for ARTS/MELLLA Off-rated Transient Analyses – Normal Feedwater Temperature and Reduced Feedwater Temperature 3-8
Table 3-3 MELLLA Transient Analysis Results at RTP Conditions 3-9
Table 3-4 MELLLA Transient Analysis Results at RTP Conditions 3-9
Table 3-5 Power Dependent Analysis Summary - Above P_{Bypass} at EOC 3-9
Table 3-6 Power Dependent Analysis Summary - Below P_{Bypass} at EOC with EOC-RPTOOS 3-10
Table 3-7 Power Dependent Analysis Summary - Below P_{Bypass} at EOC with TBVOOS and EOC-RPTOOS 3-10
Table 3-8 Power Dependent MCPR Limits for GE14 and Atrium-10 3-11
Table 3-9 Power Dependent LHGR Limits for GE14 and Atrium-10 3-11
Table 3-10 Flow Dependent Analysis Summary - All Application Groups 3-12
Table 3-11 Flow Dependent MCPR(F) Limits for GE14 and Atrium-10 3-12
Table 3-12 Flow Dependent LHGRFAC(F) Limits for GE14 and Atrium-10 3-12
Table 4-1 Rod Block Monitor System Improvements 4-13
Table 4-2 Rod Withdrawal Error Analysis Results 4-14
Table 4-3 RWE Analysis Results For Peripheral Rod Groups (108% Setpoint) 4-15
Table 4-4 RBM Signal Filter Setpoint Adjustment 4-16
Table 4-5 RBM System Setup 4-17
Table 4-6 ARTS RBM System Setpoints 4-18
Table 4-7 RBM Setup Setpoint Definitions 4-19
Table 5-1 CGS Cycle 20 Sensitivity of Overpressure Analysis Results to Initial Flow 5-1
Table 6-1 Option III Setpoint Demonstration with HCOM CF of 1.0 Hz 6-4
Table 6-2 BSP Region Endpoints for NFWT 6-5
Table 6-3 BSP Region Endpoints for RFWT 6-5
Table 7-1 ECCS-LOCA Analysis Bases for CGS ARTS/MELLLA 7-3
Table 7-2 ECCS-LOCA Peak Cladding Temperature for CGS ARTS/MELLLA 7-4
Table 8-1 Cases Analyzed For Short-Term Containment Response 8-11
Table 8-2 Conditions Reviewed for MELLLA 8-12
Table 8-3 Summary of Sensitivity Study Results for Peak Drywell Pressure and Temperature and Initial Drywell Pressurization Rate 8-13
Table 9-1 Flow-Induced Loads on Shroud and Jet Pumps for CGS 9-6
Table 9-2 Maximum Acoustic Loads on Shroud and Jet Pumps 9-6

Table 9-3 Maximum Acoustic Loads on Shroud Support (MELLLA) 9-6
Table 9-4 Governing Stress Results for RPV Internals 9-7
Table 10-1 Operating Conditions and Equipment Performance Characteristics for ATWS Analyses 10-5
Table 10-2 Summary of ATWS Calculation Results 10-6

LIST OF FIGURES

Figure 1-1 MELLLA Operating Range Power/Flow Map 1-11
Figure 4-1 Conceptual Illustration of Current Flow-Dependent RBM System with AC/BD LPRM Assignment 4-20
Figure 4-2 RBM Current AC/BD LPRM Assignment 4-21
Figure 4-3 Typical RBM System Configuration Limits (Typical for 106% Setpoint) 4-22
Figure 4-4 New Power-Dependent RBM System with $BCCD_1/BCCD_2$ LPRM Assignment 4-23
Figure 4-5 New RBM $BCCD_1/BCCD_2$ LPRM Assignment 4-24
Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs) 4-25
Figure 4-7 New RBM System Core Power Limit (Typical) 4-26
Figure 4-8 Design Basis RWE MCPR Requirement Versus RBM Setpoint 4-27
Figure 4-9 Design Basis MCPR Requirement for RWE (ARTS) 4-28
Figure 4-10 RBM Setpoint Versus Power (without Filter) 4-29
Figure 4-11 CGS Neutron Monitoring System 4-30
Figure 4-12 Rod Block Monitor Rod Group Geometries 4-31
Figure 4-13 Results of LPRM Failure Rate Sensitivity Studies 4-32
Figure 4-14 Power-Dependent RBM Trip Nomenclature 4-33
Figure 6-1 MELLLA OPRM Trip Enabled Region 6-6
Figure 6-2 Demonstration of Proposed BSP Regions for NFWT 6-7
Figure 6-3 Demonstration of Proposed BSP Regions for RFWT 6-8

ACRONYMS

Term	Definition
ΔCPR	Delta Critical Power Ratio
ΔW	Difference in % flow between two loop and single loop recirculation drive flow at the same core flow
ABA	Amplitude Based Algorithm
AC/BD	A Channel, C Channel/B Channel, D Channel
ADS	Automatic Depressurization System
AL	Analytical Limit
AOO	Anticipated Operational Occurrence
AP	Annulus Pressurization
APLHGR	Average Planar Linear Heat Generation Rate
APRM	Average Power Range Monitor
APEA	Time Independent part of Primary Element Accuracy
ARI	Alternate Rod Insertion
ARS	Amplified Response Spectra
ARTS	Average Power Range Monitor /Rod Block Monitor and Technical Specifications Improvement Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
BOC	Beginning-of-Cycle
BSP	Backup Stability Protection
BT	Boiling Transition
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CGS	Columbia Generating Station
CF	Corner Frequency
CH	Chugging
CLTP	Current Licensed Thermal Power
CO	Condensation Oscillation
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRGT	Control Rod Guide Tube
DAR	Design Assessment Report
DBA	Design Basis Accident
DIVOM	Delta CPR Over Initial MCPR Versus Oscillation Magnitude
DPEA	Time Dependent part of Primary Element Accuracy
DSRV	Dual Safety Relief Valve

Term	Definition
DTPF	Design Total Peaking Factor
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
ENW	Energy Northwest
FFWTR	Final Feedwater Temperature Reduction
FIV	Flow-Induced Vibration
FWLB	Feedwater Line Break
FRTP	Fraction of Rated Thermal Power
FSAR	Final Safety Analysis Report
FW	Feedwater
FWCF	Feedwater Controller Failure
FWHOOS	Feedwater Heater Out-of-Service
FWLB	Feedwater Line Break
FWTR	Feedwater Temperature Reduction
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas, LLC
GESTR	GE Stress and Thermal Analysis of Fuel Rods
GEXL	GE Critical Boiling Length
GNF	Global Nuclear Fuel
GRA	Growth Rate Algorithm
HCOM	Hot Channel Oscillation Magnitude
HELB	High Energy Line Break
HFCL	High Flow Control Line
HPCS	High Pressure Core Spray
HPCSDG	High Pressure Core Spray Diesel Generator
HPSP	High Power Setpoint
IBA	Intermediate Break Accident
ICF	Increased Core Flow
ICGT	Incore Guide Tube
ICPR	Initial Critical Power Ratio
IORV	Inadvertent Opening of a Relief Valve
IPSP	Intermediate Power Setpoint
IRLS	Idle Recirculation Loop Start-up
ISA	Instrumentation, Systems, and Automation Society
JPSL	Jet Pump Sensing Line
JR	Jet Reaction
LFWH	Loss of Feedwater Heating

Term	Definition
LHGR	Linear Heat Generation Rate
LHGRFAC	LHGR Multiplier
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPCI	Low Pressure Coolant Injection
LPRM	Local Power Range Monitor
LPSP	Low Power Setpoint
LRNBP	Generator Load Rejection with No Bypass
LSSS	Limiting Safety System Settings
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MFLPD	Maximum Fraction of Limiting Power Density
MOP	Mechanical Over-Power
MPS	Minimum Pump Speed
MSIV	Main Steam Isolation Valve
MSIVC	Main Steam Isolation Valve Closure
MSIVF	Main Steam Isolation Valve Closure with a Flux Scram
MSLB	Main Steam Line Break
NCL	Natural Circulation Line
NFWT	Normal FW Temperature
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
NUMAC™	Nuclear Measurement Analysis and Control
OFS	Orifice Fuel Support
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature
P/F	Power/Flow as in Power/Flow Map
PMA	Process Measurement Accuracy
PRNM	Power Range Neutron Monitor
PRNMS	Power Range Neutron Monitoring System
PRFO	Pressure Regulator Failure Open

Term	Definition
PS	Pool Swell
RBM	Rod Block Monitor
RCF	Rated Core Flow
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary Piping
RFI	Recirculation Flow Increase
RFWT	Reduced FW Temperature
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RHR	Residual Heat Removal (System)
RIPD	Reactor Internal Pressure Difference
RPT	Recirculation Pump Trip
RPTOOS	Recirculation Pump Trip Out-of-Service
RPV	Reactor Pressure Vessel
RRS	Reactor Recirculation System
RSLB	Recirculation Suction Line Break
RTP	Rated Thermal Power
RWCU	Reactor Water Cleanup
RWE	Rod Withdrawal Error
SER	Safety Evaluation Report
SLCS	Standby Liquid Control System
SLMCPR	Safety Limit Minimum Critical Power Ratio
SLO	Single Loop Operation
SRLR	Supplemental Reload Licensing Report
SRSS	Square Root of the Sum of the Squares
SRV	Safety-Relief Valve
SSE	Safe Shutdown Earthquake
STP	Simulated Thermal Power
TLO	Two Loop Operation
TOP	Thermal Over-Power
TS	Technical Specification
TTNBP	Turbine Trip with No Bypass
VPF	Vane Passing Frequency
V&V	Verification and Validation
W_T	% Of Rated Core Flow

1.0 INTRODUCTION

Many factors restrict the flexibility of a Boiling Water Reactor (BWR) during power ascension from the low-power/low-core flow condition to the high-power/high-core flow condition. Once rated power is achieved, periodic adjustments must be made to compensate for reactivity changes due to xenon effects and fuel burnup. Some of the factors at the Energy Northwest Columbia Generating Station (CGS) that restrict plant flexibility are:

1. The current operating power/flow (P/F) map,
2. The Average Power Range Monitor (APRM) flow-biased flux scram and flow-biased rod block setdown requirements, and
3. The Rod Block Monitor (RBM) flow-referenced rod block trip.

The current Extended Load Line Limit Analysis (ELLLA) P/F upper boundary is being modified to include the operating region bounded by the rod line which passes through the 100% of current licensed thermal power (CLTP) / 80.7% of rated core flow (RCF) point, the rated thermal power (RTP) line, and the rated load line, as shown in Figure 1-1. The P/F region shown in Figure 1-1 above the current ELLLA boundary is referred to as the Maximum Extended Load Line Limit Analysis (MELLLA) region. The MELLLA expansion of the power-flow map provides improved operational flexibility by allowing operation at RTP with less than RCF.

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

1. A power-dependent Minimum Critical Power Ratio (MCPR) thermal limit, similar to that used by BWR6 plants, is implemented as an update to reactor thermal limits administration.
2. The APRM trip setdown and Design Total Peaking Factor (DTPF) are replaced by more direct power-dependent and flow-dependent thermal limits to reduce the need for manual setpoint adjustments and to provide more direct thermal limits administration. This improves human/machine interface, improves thermal limits administration, increases reliability, and provides more direct protection of plant limits.
3. The flow-biased RBM trips are replaced by power-dependent trips. The RBM inputs are reassigned to: improve the response characteristics of the system, improve the response predictability, and reduce the frequency of nonessential alarms.
4. The Rod Withdrawal Error (RWE) analysis is performed in a manner that more accurately reflects actual plant operating conditions, and is consistent with the system changes.
5. Operability requirements are redefined to be consistent with the modified configuration and supporting analyses.

This report presents the results of the safety analyses and system response evaluations performed for operation of CGS in the region above the rated rod line.

1.1 Background

CGS has performed a Stretch Power Uprate, which increased the CLTP to 3486 MWt or 104.9% of the Original Licensed Thermal Power (OLTP), 3323 MWt (Reference 1). In this report, the terms CLTP and RTP are analogous, i.e. both refer to CGS operation at 3486 MWt.

CGS originally included minimum critical heat flux ratio (MCHFR) as the thermal margin criterion. This MCHFR basis included operating, overpower, and safety limit values that along with a design power peaking factor, translate to the rated power load line, and 108% load line respectively (thus, the APRM flow-biased rod block and scram protection functions). Therefore, these APRM flow-biased setpoint values originated with a deterministic overpower analysis. Later, with the change to the MCPR thermal margin basis under which CGS was originally licensed, studies concluded that the Safety Limit MCPR (SLMCPR) would be met for the design basis transients with the peaking restrictions being conservative for off-rated transients. The CGS Final Safety Analysis Report (FSAR) includes the results of rated power transients, which establish the Operating Limit MCPR (OLMCPR).

The ARTS changes replace the power peaking factor restrictions with power and flow dependent limits. However, the flow-biased APRM rod block and scram remain as defense in depth design features. A reduction in APRM flow-biased function slope from 0.66 to 0.58 has been implemented, to improve the ability to reach the rated load line at lower flow, the addition of setpoint uncertainties to the nominal values, and the restoring of margin to the operating load line for ELLLA. The original 0.66 flow-biased slope reflected the general relationship between power and flow of a 2 to 3 ratio, but using drive flow was deemed too conservative for low flows, thus the 0.58 slope was justified for ELLLA (Reference 1).

Plants with full ARTS/MELLLA including Increased Core Flow (ICF) implementation are: Nine Mile Point Unit 2, Hatch Units 1 and 2, Duane Arnold (no ICF), Cooper, Pilgrim, Fermi, Monticello, Brunswick Units 1 and 2, Peach Bottom Units 2 and 3, Limerick Units 1 and 2, and Browns Ferry Units 1, 2 and 3. Plants with partial ARTS/MELLLA including ICF implementation are: Fitzpatrick, Hope Creek, LaSalle Units 1 and 2, Dresden Units 2 and 3, Quad Cities Units 1 and 2, Susquehanna, and Vermont Yankee.

1.2 ARTS/MELLLA Bases

1.2.1 Analytical Bases

The P/F operating map (Figure 1-1) includes operating domain changes for ARTS/MELLLA consistent with approved operating domain improvements for other BWRs. The CGS MELLLA operating domain is defined by the following upper boundary:

- The MELLLA boundary line, extended up to the existing maximum CLTP of 3486 MWt. The MELLLA boundary is defined as the line that passes through the 100% of CLTP / 80.7% of RCF state point.
- The CLTP of 3486 MWt.
- The currently analyzed ICF condition of 106.0% of RCF.
- The MELLLA boundary is defined by the following equation in terms of current licensed core power, P (% of rated), versus core flow, W_T (% of RCF), as follows:

$$\underline{P=(A+B\cdot W_T+C\cdot W_T^2)\cdot K}$$

where: A = 22.191

B = 0.89714

C = -0.0011905

K = 1.152 for the MELLLA upper boundary.

The MELLLA boundary line defines an increase in the extent of the current operating domain above the current boundary. The current boundary is the ELLLA, corresponding to the 108% APRM Rod Block setpoint, and allows operation to approximately the 108% of CLTP rod line.

The currently analyzed P/F point for Single Loop Operation (SLO) operation remains unchanged from its current value of 2615 MWt (75% of RTP) for MELLLA. For CGS, SLO is not extended into the MELLLA region.

When compared to the current P/F operating domain, the MELLLA region allows a higher core power at a given core flow. This increases the fluid subcooling in the reactor vessel downcomer and changes the power distribution in the core, which can potentially affect the steady-state operating thermal limit and transient/accident analyses results. The effect of the MELLLA operating domain has been evaluated to support compliance with the Technical Specification (TS) fuel thermal margins during plant operation. This report presents the results of the safety analyses and system response evaluations performed for operation of CGS in the region above the ELLLA and up to the MELLLA boundary line. The scope of the analyses performed covers the initial application for CGS operation with ARTS/MELLLA. Upon ARTS/MELLLA approval, reload cycles will include the ARTS/MELLLA operating condition in the reload-licensing basis in accordance with Reference 2.

The safety analyses and system evaluations performed to justify operation in the MELLLA region consist of a non-fuel dependent portion and a fuel dependent portion that is fuel cycle dependent. In general, the limiting anticipated operational occurrences (AOOs) MCPR calculation and the reactor vessel overpressure protection analysis are fuel dependent. These analyses, discussed in this report, are based on the current Cycle 20 core design using GE14 and ATRIUM-10 fuel (Reference 3). Subsequent cycle-specific analyses will be performed in conjunction with the reload licensing activities. The non-fuel dependent evaluations such as containment response are based on the current plant design and configuration. The limiting AOOs identified in Reference 4 were reviewed for the MELLLA region based on existing thermal analysis limits at plants similar to CGS and use of generic power-dependent and generic flow-dependent MCPR and Linear Heat Generation Rate (LHGR). For the fuel-dependent evaluations of reactor pressurization events, these reviews indicate that there is a small difference in the OLMCPR for operation in the MELLLA region and the ICF condition (100% of RTP / 106% of RCF). The operating limit is calculated on a cycle specific basis in accordance with Reference 2 to bound the entire operating domain. The analysis results indicate that performance in the MELLLA region is within allowable design limits for overpressure protection, loss-of-coolant accident (LOCA), containment dynamic loads, flow-induced vibration, and reactor internals structural integrity. The response to the Anticipated Transient Without Scram (ATWS) demonstrates that CGS meets the licensing criteria in the MELLLA operating domain.

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

1.2.2 APRM High Flux (Flow-Bias) Scram and Rod Block Design Bases

The APRM Flow-Biased Simulated Thermal Power (STP) scram line is conservatively not credited in any CGS safety analyses. In addition, the APRM Flow-Biased STP rod block line is conservatively not credited in any CGS safety analyses, although it is part of the CGS design configuration.

This section discusses the setpoint changes for these systems for operational flexibility purposes and provides the inputs to the CGS TS changes.

For the current, ELLLA operating domain, P/F map, the APRM Flow-Biased STP scram line allowable value (AV) for two loop operation (TLO) is defined as: 0.58 Wd + 62%, and for SLO, 0.58 Wd +62%, of RTP. The APRM Flow-Biased STP Scram clamp AV is at 114.9% of RTP. Wd is defined as the recirculation drive flow for TLO in percent of rated, where 100% drive flow is that required to achieve 100% core power and flow. The APRM Flow-Biased STP rod block AV is currently set at: for TLO, 0.58 Wd + 53%, and for SLO, 0.58 + 53% of RTP. CGS does not have an APRM Flow-Biased STP Rod Block clamp. A Rod Block clamp AV of 111% will be implemented for ARTS/MELLLA.

To accommodate this expanded operating domain and to restore the original margin between the MELLLA boundary line and the APRM Flow-Biased STP rod block line, the following AVs are redefined:

Analytical Value		TLO	SLO
APRM Flow-biased STP High Scram	Flow-Biased Equation *	0.63(Wd -ΔW) + 64.0% = 0.63 Wd + 64.0%	0.63(Wd - ΔW) + 64.0% = 0.63 Wd + 60.8%
	Flow-Biased Clamp	No change	No change
APRM Flow-biased STP Rod Block	Flow-Biased Equation *	0.63(Wd - ΔW) + 60.1% = 0.63 Wd + 60.1%	0.63(Wd - ΔW) + 60.1% = 0.63 Wd + 56.9%
	Flow-Biased Clamp	111%	111%

* ΔW is the difference in percent flow between the TLO and SLO Recirculation drive flow at the same core flow. The TLO ΔW is 0% and the SLO ΔW is 5%.

The RBM Upscale Flow-Biased rod block line limits are currently set at:

- TS AVs

  TLO: 0.58 Wd + 51% of RTP

  SLO: 0.58 Wd + 51%, of RTP

- AL values

  TLO: 0.58 Wd + 54% of RTP

  SLO: 0.58 Wd + 54%, of RTP

ARTS changes the form of the RBM from a flow-biased to a power-biased function. In Section 4.3, the evaluation of the RWE event was performed taking credit for the mitigating effect of the power-dependent RBM. The power-dependent RBM ALs and AVs are presented in Table 4-5.

The AV revisions were performed using the General Electric (GE) instrument setpoint methodology (Reference 5). Attachment A provides the GE setpoint calculation for the power-based RBM setpoint function.

The RBM trip setpoints are determined by use of Nuclear Regulatory Commission (NRC) approved setpoint methodology. Using the GE setpoint methodology based on Instrumentation, Systems, and Automation Society (ISA) setpoint calculation method 2, the RBM AVs are determined from the AL, corrected for RBM input signal calibration error, process measurement error, primary element accuracy and instrument accuracy under trip conditions. The error due to the neutron flux measurement is accounted for in the non-linearity error from the Local Power Range Monitor (LPRM) detectors and is referred to in the setpoint calculation as the APRM Primary Element Accuracy. There is both a bias and random component to this APRM Primary Element Accuracy error. There is also an error due to tracking and neutron flux noise, and that is labeled as Process Measurement Accuracy (PMA). The RBM trip setpoint has no drift characteristic with no as-left or as-found tolerances because it only performs digital calculations on digitized input signals. The Nominal Trip Setpoint (NTSP) includes a drift allowance over the interval from rod selection to rod movement, which is not the surveillance interval. Drift of RBM channel components between surveillance intervals does not apply to the normalized RBM reading.

Surveillance procedures are used to establish operability of the RBM. The surveillance procedures include appropriate steps to ensure the RBM is functioning properly and that the proper setpoint values are established in the hardware. Other self-test functions are performed automatically and routinely in the RBM hardware modules (Central Processing Unit, Power Supplies, etc.) The periodic RBM calibration in the Technical Specifications requires a verification of only the trip setting. The trip setpoints are stored in computer memory as fixed numerical values and thus cannot drift due to the nature of the RBM instrument (digital hardware). The calibration method in the Technical Specification surveillance procedures ensures that the trip setting is proper. Because the trip setpoint is a numerical value stored in the digital hardware and not subject to drift, the as-found and as-left tolerance values for the setpoint are the same as the setpoint (i.e., there is no tolerance band). The surveillance procedures also perform a channel functional test, which assures the RBM is functioning properly.

The suggested notes in Regulatory Issue Summary (RIS) 2006-17 (Reference 6), which are intended to assure that the trip setpoints are verified to be within predefined limits, so appropriate actions can be taken if found to be outside the predetermined limits, cannot be applied to the RBM. The suggested notes cannot be applied to the RBM because the trip setpoints are keyed into the RBM module via a keyboard and are displayed and stored as digital values in computer memory. As such, the RBM trip setpoint is therefore not subject to drift, is not calibrated in the traditional sense, and does not have as-found and as-left tolerance bands. The calibration of the RBM verifies the trip setpoint is as it was set. No range of values is acceptable, only the exact keyed in values, thus assuring that the CGS RBM trip will be within safety limits and that the limiting condition for operation will be met.

The RBM trip setpoints are Limiting Safety System Setting (LSSS) because they are used in the RWE analysis for MCPR protection. However, the setpoints are exempt from the requirements of RIS 2006-17 based on this discussion (Reference 6).

The APRM Flow-Biased setpoints addressed in this report are not LSSSs because they are not used in any safety analyses and are therefore not affected by RIS 2006-17 (Reference 6).

1.3 Average Power Range Monitor Improvements

The functions of the APRM are integrated within the Nuclear Measurement Analysis and Control (NUMAC^{TM}) Power Range Neutron Monitoring System (PRNMS). The safety related functions of the APRM are to:

1. Generate trip signals to automatically scram the reactor during core-wide neutron flux transients before the neutron flux level exceeds the safety analysis design bases. This

prevents exceeding design bases and licensing criteria from single operator errors or equipment malfunctions.

2. Block control rod withdrawal before core power approaches the scram level when operation occurs in excess of set limits in the P/F map.
3. Provide an indication of the core average power level of the reactor in the power range.

The NUMAC^{TM} PRNMS APRM calculates an average LPRM chamber signal such that the APRM signal is proportional to the core average neutron flux and can be calibrated as a means of measuring core thermal power. The APRM signals are used to calculate the STP that closely approximates reactor thermal power during a transient. The STP signals are compared to a recirculation drive flow-referenced scram and a recirculation drive flow-referenced control rod withdrawal block.

CGS currently operates such that the Maximum Fraction of Limiting Power Density (MFLPD) is less than or equal to the Fraction of Rated Thermal Power (FRTP), which limits the local power peaking at lower core power and flows. If the ratio of the MFLPD to the FRTP is greater than 1, the flow-referenced APRM trips must be lowered (setdown) or the APRM gain must be increased (CGS current Technical Specification 3.2.4) to limit the maximum power that the plant can achieve. The basis for this "APRM trip setdown" requirement originated under the original BWR design Hench-Levy MCHFR thermal limit criterion and provides conservative restrictions with respect to current fuel thermal limits. The original MCHFR basis is described in Reference 7.

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

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

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

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

The safety analyses used to evaluate the OLMCPR are documented in Section 3.0 of this report. These analyses ensure that the SLMCPR and the fuel thermal-mechanical design bases are satisfied. These analyses also establish the power-dependent and flow-dependent MCPR and

LHGR curves for CGS. The effect on the LOCA response due to the ARTS program implementation is discussed in Section 7.0 of this report.

1.4 Rod Block Monitor Improvements

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

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

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

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

Task	Computer Code	Version or Revision	NRC Approved	Comments
Reactor Heat Balance	ISCOR	09	Y (1)	NEDE-24011-P Rev 0 SER
Reactor Core and Fuel Performance	TGBLA PANAC ISCOR	06 11 09	Y Y Y(1)	NEDE-30130-P-A (2) NEDE-30130-P-A (2) NEDE-24011-P Rev 0 SER
Thermal-Hydraulic Stability	ISCOR PANAC ODYSY OPRM TRACG	09 11 05 01 04	Y(1) Y Y Y(3) N(12)	NEDE-24011P Rev. 0 SER NEDE-30130-P-A (2) NEDC-33213P-A NEDO-32465-A NEDO-32465-A
Reactor Internal Pressure Differences	TRACG ISCOR	02 09	(5) Y (1)	NEDE-32176P, Rev 2, Dec 1999 NEDC-32177P, Rev 2, Jan 2000 NRC TAC No M90270, Sep 1994 NEDE-24011-P Rev 0 SER
Transient Analysis	PANAC ODYN ISCOR TASC	11 09 (11) 09 03	Y Y Y (1) Y	NEDE-30130-P-A (6) NEDE-24154P-A NEDC-24154P-A, Volume 4, NEDE-24011-P Rev 0 SER NEDC-32084P-A, Rev 2
Containment System Response	M3CPT LAMB	05(13) 08(13)	Y (4)	NEDO-10320, April 1971 (NUREG-0661) NEDE-20566P-A, September 1986
Annulus Pressurization Loads	ISCOR LAMB	09 08	Y (1) (4)	NEDE-24011-P Rev. 0 SER NEDE-20566P-A
Annulus Pressurization Loads-Reactor Pressure Vessel (RPV) and Internal Structural Analysis	GEAPL SAP4G SPECA	01 07V 03V	N(14) N(14) N(14)	NEDE-25199, October 1979 NEDO-10909, Revision 7, Dec. 1979 NEDE-25181, Addendum 1, Aug. 1996
ECCS-LOCA	LAMB GESTR SAFER ISCOR TASC	08 08 04 09 03	Y Y Y Y (1) Y	NEDE-20566P-A NEDE-23785-1P-A, Rev 1 (7) (8) (9) NEDE-24011-P Rev 0 SER NEDC-32084P-A Rev 2
Anticipated Transient Without Scram	PANAC ODYN STEMP	11 09 (11) 04	Y Y (10)	NEDE-30130-P-A (6) NEDC-24154P-A, Volume 4, Sup 1

Notes For Table 1-1:

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

(2) The use of TGBLA Version 06 and PANACEA Version 11 was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE) Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods," (TAC NO. MA6481), November 10, 1999.

(3) The methodology as implemented in the OPRM code (provided in NEDO-32465-A) has been approved by the NRC.

(4) The LAMB code is approved for use in ECCS-LOCA applications (NEDE-20566P-A), but no approving SER exists for the use of LAMB for the evaluation of reactor internal pressure differences or containment system response. The use of LAMB for these applications is consistent with the model description of NEDE-20566P-A.

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

(6) The physics code PANACEA provides inputs to the transient code ODYN. The use of PANAC Version 11 in this application was initiated following approval of Amendment 26 of GESTAR II by letter from S.A. Richards (NRC) to G.A. Watford (GE), Subject: "Amendment 26 to GE Licensing Topical Report NEDE-24011-P-A, GESTAR II Implementing Improved GE Steady-State Methods", (TAC NO. MA6481), November 10, 1999.

(7) "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-Jet Pump Plants," NEDE-30996P-A, General Electric Company, October 1987.

(8) "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," NEDC-32950P, January 2000.

(9) Letter, S.A. Richards (NRC) to J.F. Klapproth (GE), "General Electric Nuclear Energy Topical Reports NEDC-32950P and NEDC-32084P Acceptability Review," May 24, 2000.

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

(11) Version 9 of ODYN is applicable to plants that use recirculation valve for recirculation flow control.

(12) TRACG02 has been approved in NEDO-32465-A by the US NRC for the stability DIVOM analysis. The CLTP stability analysis is based on TRACG04, which has been shown to provide essentially the same or more conservative results in DIVOM applications as the previous version, TRACG02.

(13) The evaluation performed for ARTS/MELLLA did not explicitly include the use of these codes in analyses. However, the evaluation uses the results of previous analyses performed for CGS in support of Power Uprate/ELLLA (Reference 1), which applied these codes.

(14) The code application is reviewed and approved by GEH for "Level-2" application and is part of GEH's standard design process.

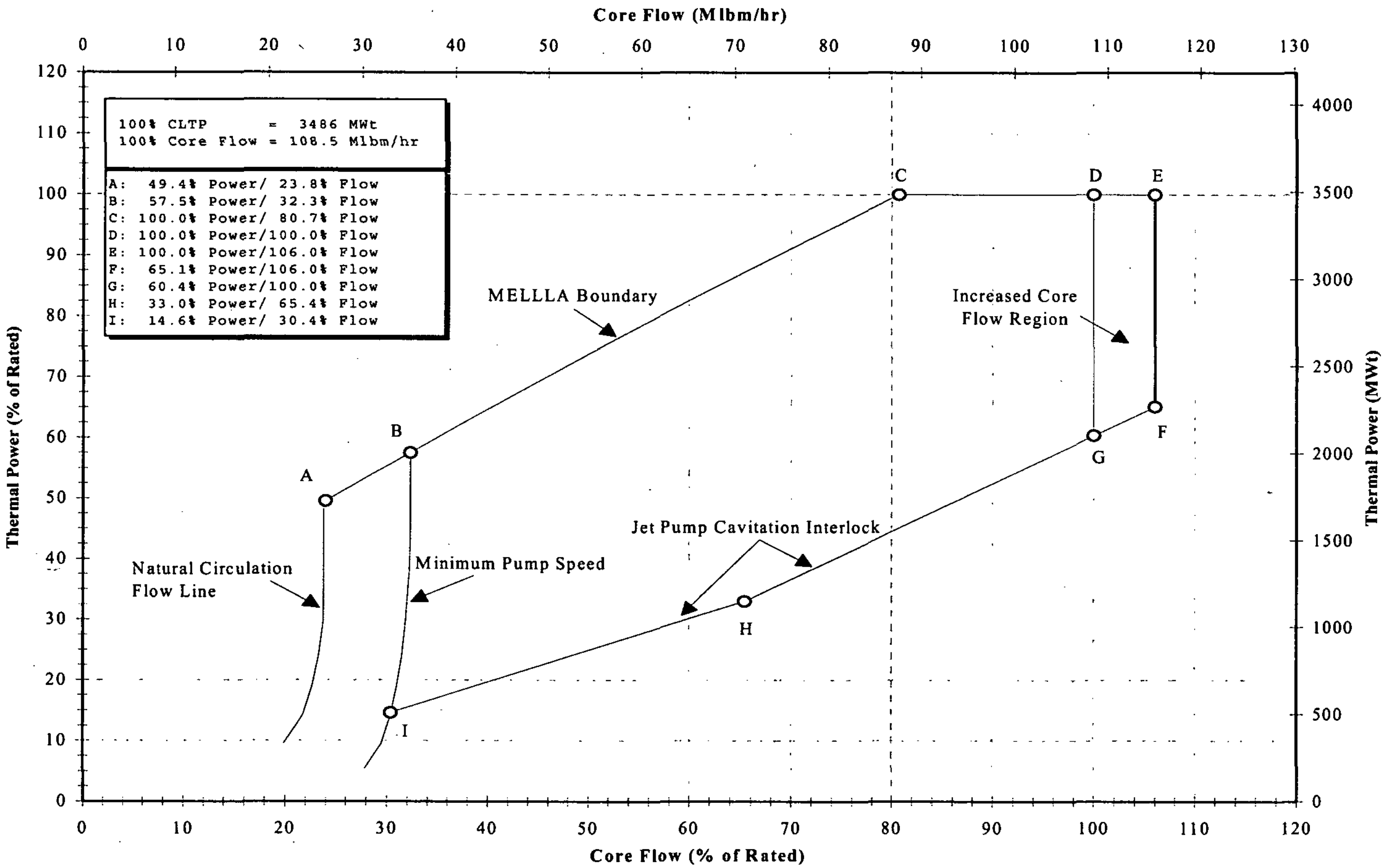


Figure 1-1 MELLLA Operating Range Power/Flow Map

2.0 OVERALL ANALYSIS APPROACH

This section identifies the analyses that may be affected by the proposed MELLLA region. The analyses performed in the following sections are based on the current plant operating parameters. For the transient and stability tasks, the CGS Cycle 20 core design was utilized. These tasks will be revalidated as part of the subsequent cycle-specific reload licensing analyses in accordance with Reference 2. The remainder of the ARTS/MELLLA scope of work is applicable to CGS, unless there is a plant configuration change that affects the analysis.

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

Table 2-1 Analyses Presented In This Report

Section	Item	Result
3.0	Fuel Thermal Limits	Acceptable - Bounded by Limits Presented in Section 3.0
4.0	Rod Block Monitor System Improvement	Acceptable for Cycle 20 Core
5.0	Vessel Overpressure Protection	Acceptable - Below ASME Limit
6.0	Thermal-Hydraulic Stability	Acceptable for Cycle 20 Core
7.0	LOCA Analysis	Acceptable for Cycle 20 Core
8.0	Containment Response	Acceptable – Bounded by Current Results
9.0	Reactor Internals Integrity	Acceptable – Bounded by Design Criteria
10.0	ATWS	Acceptable – Bounded by Design Criteria
11.0	Steam Dryer and Separator Performance	Acceptable – Bounded by Design Criteria
12.0	High Energy Line Break (HELB)	Acceptable – Bounded by Design Criteria
13.0	Testing	Acceptable with the performance of the identified tests

Table 2-2 Applicability of Analyses

Task Description	Generic or Plant-Specific	Cycle-Independent or Cycle-Dependent
Power-dependent MCPR and LHGR limits (between rated power and 30% of RTP)	Generic, with plant-specific confirmation for initial application	Cycle-independent unless change in plant configuration from licensing analysis basis
Power-dependent MCPR and LHGR limits (between 30% and 25% of RTP)	Plant-specific	Cycle-dependent review
Flow-dependent MCPR and LHGR limits	Generic	Cycle-independent unless change in plant configuration from licensing analysis basis.
RBM power-dependent setpoints	Generic, with plant-specific confirmation for initial application	Cycle-independent unless change in plant configuration from licensing analysis basis. Cycle-dependent RWE analysis performed with the applicable setpoints.

3.0 FUEL THERMAL LIMITS

The potentially limiting AOOs and accident analyses were evaluated to support CGS operation in the MELLLA region with ARTS off-rated limits. The P/F state points chosen for the review of AOOs are presented in Table 3-1 and Table 3-2. These state points include the MELLLA region and the current licensed operating domain for CGS. The AOO evaluations are discussed in Sections 3.1 through 3.3. Section 3.4 discusses the governing MCPR and LHGR limits. Section 4.0 includes consideration of the RWE analyses and the LOCA analyses are presented in Section 7.0.

3.1 Limiting Core-Wide Anticipated Operational Occurrence Analyses

The core-wide AOOs included in the current Cycle 20 reload licensing analyses (Reference 3) and the CGS FSAR were examined for operation in the ARTS/MELLLA region (including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MELLLA region and were reviewed as part of the ARTS program development:

- Generator Load Rejection with No Bypass (LRNBP) event;
- Turbine Trip with No Bypass (TTNBP) event;
- Feedwater Controller Failure (FWCF) maximum demand event;
- Loss of Feedwater Heating (LFWH) event;
- Inadvertent High Pressure Core Spray (HPCS) Startup event;
- Idle Recirculation Loop Start-up (IRLS) event; and
- Recirculation Flow Increase (RFI) event.

The LRNBP, TTNBP, FWCF, LFWH, and HPCS events were generally the source of the power-dependent thermal limits, while the IRLS and RFI events were generally the source of the flow-dependent thermal limits.

The initial ARTS/MELLLA assessment of these events for all BWR type plants concluded that for plant-specific applications, only the TTNBP, LRNBP, and FWCF events need to be evaluated at both rated and off-rated power and flow conditions.

The generic assessments were performed to determine the most limiting transients and characteristics for the BWR fleet. This was done by using the plant characteristics from the fleet of BWR/3 through BWR/5 plants that resulted in the most limiting transients. The plants were chosen to cover a wide range of conditions and characteristics including steam line volume, plants with and without the recirculation pump trip (RPT) feature, high and low feedwater runout capacity, and low bypass capacity. None of the BWR/5 plants had plant characteristics that were limiting for the fleet.

The key plant characteristics considered for off-rated limits calculations include:

- Steam Line Characteristics
- Feedwater (FW) Runout Capacity
- High Pressure Core Spray (HPCS) Flow Capacity
- Recirculation Pump Trip
- Steam Bypass Capacity
- Relief Capacity
- Design Conditions (Power Density, FW temperature, etc.)

To confirm the applicability of the generic assessment to CGS, plant-specific power dependent calculations were performed which included all of the key plant characteristics described above that applied to CGS. These analyses were performed with approved methods (see Table 1-1) and the most recent core designs. These analyses confirmed the applicability of the generic assessments for the limiting AOOs to CGS. The LFWH, HPCS, IRLS, and RFI events were not specifically evaluated for the following reasons.

- The LFWH event is not limiting for CGS and the effect of MELLLA on the LFWH severity is sufficiently small that the LFWH remains non-limiting for MELLLA. The required MCPR for Cycle 20 (Reference 3) LFWH transient is 1.23 based on an 87% initial core flow compared to an End-of-Cycle (EOC) Option B OLMCPR of 1.39 from the LRNBP event. At 80.7% initial core flow, the required MCPR for the LFWH event is also 1.23 thus maintaining a large margin to the rapid pressurization, LRNBP, TTNBP, and FWCF AOO events. Consequently, the LFWH does not factor into the determination of the off-rated limits. However, it should be noted that the LFWH event is analyzed on a cycle-specific basis.
- The inadvertent HPCS Startup results in the injection of cold water in the upper plenum area above the core. This results in a small depressurization and core power decrease as some of the steam generated by the core is quenched. The pressure regulator responds to maintain the pressure at the pressure setpoint, and the feedwater control system responds to the increased inventory provided by the HPCS system. The system would settle to a new steady state without a scram in this scenario with increased margins to thermal limits compared to the initial conditions due to the decreased power. Consequently, the inadvertent HPCS Startup event was not considered in the determination of the off-rated limits.
- [[

  ]] The SLO state is not expanded to the MELLLA domain for CGS.

  [[

  ]] As

previously stated, these events were considered generically in the development of the ARTS flow-dependent limits, which are generated based on a conservative two pump flow run-up analysis described in Sections 3.3.3 and 3.3.4.

3.1.1 Elimination of APRM Trip Setdown and DTPF Requirement

Extensive transient analyses at a variety of power and flow conditions were performed during the original development of the ARTS improvement program. These evaluations are applicable for operation in the MELLLA region. The analyses were utilized to study the trend of transient severity without the APRM trip setdown. A database was established by analyzing limiting transients over a range of power and flow conditions. The database includes evaluations representative of a variety of plant configurations and parameters such that the conclusions are applicable to all BWRs. The database was utilized to develop a method of specifying plant operating limits (MCPR and LHGR) such that margins to fuel safety limits are equal to or larger than those applied currently.

The generic evaluations determined that the power-dependent severity trends must be examined in two power ranges. The first power range is between rated power and the power level (P_{Bypass}) where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. The analytical value of P_{Bypass} for CGS is 30% of RTP. The second power range is between P_{Bypass} and 25% of RTP. No thermal monitoring is required below 25% of RTP, per CGS Technical Specification 3.2.

The power-dependent MCPR multiplier, K(P), was originally developed for application to all plants in the high power range (between rated power and P_{Bypass}). The values for K(P) increased at lower power levels based on the FWCF transient severity trends. As power is reduced from the rated condition in this power range, the LRNBP and TTNBP become less severe because the reduced steam flow rate at lower power results in milder reactor pressurization. However, for the FWCF, the power decrease results in greater mismatch between runout and initial feedwater flow, resulting in an increase in reactor subcooling and more severe changes in thermal limits during the event.

Between P_{Bypass} and 25% power, CGS specific evaluations were performed to establish the plant-unique MCPR and LHGR limits in the low power range (below P_{Bypass}). These plant-specific limits include sufficient conservatism to remain valid for future CGS core configurations containing ATRIUM-10 and/or Global Nuclear Fuel (GNF) fuel, except that the power-dependent MCPR limits below P_{Bypass} and flow dependent MCPR limits must be adjusted in accordance with Section 3.3.5 if the SLMCPR exceeds 1.09.

Generic flow-dependent MCPR and LHGR limits are applied to CGS. These generic limits include sufficient conservatism to remain valid for future CGS reloads of GNF and/or ATRIUM-10 fuel, utilizing the GEXL-PLUS correlation and the GEMINI analysis methods as defined in Reference 2, provided the core flow corresponding to the maximum two recirculation pump runout is ≤ 108.5% of RCF. The flow-dependent MCPR limits must be adjusted in accordance with Section 3.3.5 if the SLMCPR exceeds 1.09.

3.2 Input Assumptions

The maximum P/F state condition for the operating region analysis is the rated power and maximum flow point (100%P / 106%F). Figure 1-1 shows the P/F map used in the AOO analyses. Plant heat balance, core coolant hydraulics, and nuclear dynamic parameters corresponding to the rated and off-rated conditions were used for the analysis and reflect the CGS Cycle 20 core configuration (Reference 3). The initial conditions for the AOO analyses at rated and off-rated conditions are presented in Tables 3-1 and 3-2.

Because of the fuel cycle-independent nature of the ARTS thermal limits (for both above and below P_{Bypass} power ranges), the ARTS transient analyses are based on the CLTP of 3486 MWt. AOO analyses were performed with the approved reload licensing methodology (Reference 2).

The following assumptions and initial conditions were used in the AOO analyses:

Analytical Assumptions	Bases/Justifications
Initial core flow range of 80.7% to 106% flow for thermal limits transients at 100% of RTP	Bounding P/F state points for MELLLA
Conservative End-of-Cycle 20 nuclear dynamic parameters	Consistent with CGS current licensing bases
The lowest six opening setpoint safety-relief valves (SRVs) declared Out-of-Service (OOS)	Consistent with CGS current licensing bases
SLMCPR = 1.09	Consistent with CGS current licensing bases
[[ ]]	[[ ]]

3.3 Analyses Results

The limits associated with operation in the MELLLA region are presented in Tables 3-3 and 3-4. The MELLLA region will be incorporated into subsequent cycle specific reload licensing analyses in accordance with Reference 2. The analyses presented in Tables 3-3 and 3-4 are based on End-of-Cycle exposures. [[

]]

3.3.1 Power-Dependent MCPR Limit

As stated previously, the generic evaluations indicate that the power-dependent severity trends are to be examined in two power ranges, above and below P_{Bypass}. Above P_{Bypass}, bounding power-dependent trend functions have been developed. These trend functions, K(P), are used as multipliers to the rated MCPR operating limits to obtain the power-dependent MCPR limits, MCPR(P), or OLMCPR(P) = K(P) x OLMCPR(P=100% CLTP)

In the high power range (between rated power and P_{Bypass}), the trend for the power-dependent MCPR responses for the FWCF has been shown to be more severe than all other fast pressurization transient severity trends. As power is reduced from the rated condition in this power range, the LRNBP and TTNBP become relatively less severe because the reduced steam flow rate at low power results in milder reactor pressurization. However, for the FWCF, the power decrease results in greater mismatch between runout and initial feedwater flow, resulting in an increase in reactor subcooling and more severe changes in thermal limits during the event.

The results used to verify the generic MCPR(P) limits analyses are summarized in Tables 3-3 and 3-4. As previously stated, the MCPR(P) is derived from the generic K(P) multiplied by the rated power OLMCPR. A comparison of the plant-specific calculated values with the generic power-dependent MCPR limits verifies the applicability of the generic limits to CGS above $P_{Bypass.}$

Below P_{Bypass}, the transient characteristics change due to the bypass of the direct scram on the closure of the turbine stop valve and turbine control valve. Consequently, the scram signal is delayed until the vessel pressure reaches the high-pressure scram setpoint. The extensive transient analysis database shows significant sensitivity to the initial core flow for transients initiated below P_{Bypass}.. Therefore, the power-dependent limits are determined for power levels above 25% and below P_{Bypass} based on a core flow of 50% and 60%. The 60% core flow bounds the core flow range below 30% power based on the CGS P/F map.

Below Pbypass, the MCPR(P) limits are absolute OLMCPR values, rather than multipliers on the rated power OLMCPR. These absolute MCPR limits were chosen with sufficient conservatism such that they remain applicable to future operating cycles provided the SLMCPR is less than or equal to 1.09 (Technical Specification 2.1). The CGS specific analyses results used to establish the MCPR(P) at power levels below P_{Bypass} are summarized in Tables 3-6 and 3-7. The CGS MCPR(P) limits are given in Table 3-8.

3.3.2 Power-Dependent Linear Heat Generation Rate Limits

In the absence of the APRM trip setdown requirement, power-dependent LHGR limits, expressed in terms of a multiplier, LHGRFAC(P) are substituted to ensure adherence to the fuel thermal-mechanical design bases. The power-dependent LHGRFAC(P) multiplier was generated using the same database as used to determine the MCPR multiplier, K(P). These factors are also applied in a similar manner. Specifically,

$$\text{LHGR(P)} = \text{LHGRFAC(P)} \times \text{(rated LHGR limits)}.$$

The incipient centerline melting of the fuel (thermal over-power (TOP)) and plastic strain of the cladding (mechanical over-power (MOP)) are considered in determining the power-dependent LHGR limits.

Similar to the MCPR(P) limits, CGS-specific transient analyses were performed to demonstrate the applicability of the generic LHGR(P) limits. The transient and initial conditions selected are identical to that previously described for MCPR(P). The applicable results of these analyses for power levels above P_{Bypass} are shown in Table 3-5.

As previously discussed, significant sensitivity to initial core flow exists below P_{Bypass}. Therefore, below P_{Bypass} the power dependent LHGR multipliers are based on a core flow of 50% to 60% of rated. To prevent the situation where the limits are more restrictive after increasing power above P_{Bypass}, the extrapolation of the generic above P_{Bypass} limits are taken as the upper bound for the below P_{Bypass} limits. Appropriate LHGRFAC(P) multipliers are selected based on plant-specific transient analyses with suitable margin to ensure applicability to future CGS reloads. These limits are derived to ensure that the peak transient LHGR for any transient is not increased above the fuel design basis values. The CGS LHGRFAC(P) limits for application at power levels above and below P_{Bypass} are given in Table 3-9.

3.3.3 Flow-Dependent Minimum Critical Power Ratio Limit

Flow dependent MCPR limits, MCPR(F), are necessary to assure that the SLMCPR is not violated during recirculation flow increase events. The design basis flow increase event is a slow flow power increase event that is not terminated by scram, but which stabilizes at a new core power corresponding to the maximum possible core flow. [[

]]

The bounding generic flow dependent MCPR limits are shown in Table 3-11. To verify the applicability of the original ARTS generic flow dependent MCPR limits, RFI and IRLS events were re-performed generically for the GE14 fuel product line introduction. The Delta Critical Power Ratio (ΔCPR) results for the flow dependent limits are given in Table 3-10. For the application of ARTS, the IRLS basis is that there is an initial 50°F ΔT between the idle and operating loops. This is an appropriate assumption for thermal limits calculations and is consistent with Technical Specification requirements. The ARTS based MCPR(F) limit is specified as an absolute value and is generic and cycle independent provided the SLMCPR is less than or equal to 1.09.

3.3.4 Flow-Dependent Linear Heat Generation Rate Limits

Flow dependent LHGR limits were designed to assure adherence to all fuel thermal mechanical design bases. The same transient events used to support the MCPR(F) operating limits were analyzed, and the resulting overpowers were statistically evaluated as a function of the initial and maximum core flow. From the bounding overpowers, LHGRFAC(F) multipliers were derived such that the peak transient LHGR would not exceed fuel mechanical limits. The LHGR(F) limits are generic, cycle independent and are specified in terms of multipliers, LHGRFAC(F) , to be applied to the rated LHGR values. Specifically,

LHGR(F) = LHGRFAC(F) x (rated LHGR limits).

The LHGRFAC(F) multiplier formulas are shown in Table 3-12. The LHGRFAC(F) based on the CGS maximum runout flow of 108.5% RCF can be selected from a bounding curve or determined by interpolation.

3.3.5 Safety Limit Minimum Critical Power Ratio Adjustment Procedure

The MCPR limits, provided in Table 3-8 assume a dual-loop SLMCPR of 1.09. Only adjustment of the $P < P_{Bypass}$ portion of the MCPR(P) limits may be required because, at $P > P_{Bypass}$, the K(P) applies the rated power OLMCPR adjustment to the MCPR(P). The off-rated MCPR(F) is defined by Table 3-11. When necessary, adjustment to the entire MCPR(F) limit is required.

Should a future cycle SLMCPR exceed 1.09, the MCPR(F) and below P_{Bypass} MCPR(P) limits must be increased by the following factor:

$$\left(\frac{\text{Cycle specific SLMCPR}}{1.07}\right)$$

If a future cycle SLMCPR is less than 1.09, the MCPR(F) and below- P_{Bypass} MCPR(P) limits may optionally be reduced by the above factor.

3.3.6 Single Loop Operation Adjustment Procedure

When operating in SLO, an adjustment will be made to the rated power OLMCPR as well as the off-rated OLMCPR. The off-rated MCPR(F) is defined by Table 3-11. The off-rated MCPR(P) is defined by Table 3-8. Only adjustment of the $P < P_{Bypass}$ portion of the MCPR(P) curve is required because, at $P \geq P_{Bypass}$, the K(P) applies the rated power OLMCPR adjustment to the MCPR(P). The equation for the adjustment is as follows when operating in SLO:

$$\text{SLO OLMCPR} = \text{OLMCPR}_{\text{dual-loop}} + (\text{SLMCPR}_{\text{SLO}} - \text{SLMCPR}_{\text{dual-loop}})$$

3.4 Conclusion

The rated OLMCPRs and LHGRs are determined by the cycle-specific reload analyses in accordance with Reference 2. At any P/F state (P,F), all applicable off-rated limits are determined: MCPR(P), MCPR(F), LHGR(P), and LHGR(F). The most limiting MCPR (maximum of MCPR(P) and MCPR(F)) and the most limiting LHGR (minimum of LHGR(P) and LHGR(F)) will be the governing limits. The limits must be adjusted for SLMCPRs > 1.09 or SLO, as applicable.

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

	Rated	80.7%F MELLLA	106%F ICF
Power (MWt / % of RTP)	3486 / 100	3486 / 100	3486 / 100
Flow (Mlb/hr / % rated)	108.5 / 100	87.6 / 81	115 / 106
Steam Flow (Mlb/hr)	15.016	14.992	15.027
FW Temperature (°F)	421.2	421.2	421.2
Core Inlet Enthalpy (Btu/lb)	528.7	523.5	529.9
Dome Pressure (psia)	1035	1035	1035

Table 3-2 Base Conditions for ARTS/MELLLA Off-rated Transient Analyses – Normal Feedwater Temperature and Reduced Feedwater Temperature

(a) Normal Feedwater Temperature	85%P/106%F	85%P/63.4%F	60%P/100%F	45%P/85%F
Power (MWt / % of RTP)	2963 / 85	2963 / 85	2092 / 60	1569 / 45
Flow (Mlb/hr / % rated)	115.0 / 106	68.8 / 63.4	108.5 / 100	92.2 / 85
Steam Flow (Mlb/hr)	12.467	12.416	8.402	6.083
FW Temperature (°F)	403.8	403.4	368.9	342.3
Core Inlet Enthalpy (Btu/lb)	529.0	516.4	528.1	526.8
Dome Pressure (psia)	1019	1019	996	985
	30%P/60%F	25%P/60%F	30%P/50%F	25%P/50%F
Power (MWt / % of RTP)	1046 / 30	872 / 25	1046 / 30	872 / 25
Flow (Mlb/hr / % rated)	65.1 / 60	65.1 / 60	54.3 /50	54.3 / 50
Steam Flow (Mlb/hr)	3.874	3.171	3.869	3.167
FW Temperature (°F)	307.6	293.1	307.5	293.0
Core Inlet Enthalpy (Btu/lb)	524.5	526.2	521.2	523.3
Dome Pressure (psia)	975	973	975	973
(b) Reduced Feedwater Temperature	85%P/106%F	85%P/63.4%F	60%P/100%F	45%P/85%F
Power (MWt / % of RTP)	2963 / 85	2963 / 85	2092 / 60	1569 / 45
Flow (Mlb/hr / % rated)	115.0 / 106	68.8 / 63.4	108.5 / 100	92.2 / 85
Steam Flow (Mlb/hr)	11.535	11.489	7.882	5.759
FW Temperature (°F)	341.7	341.4	314.9	294.3
Core Inlet Enthalpy (Btu/lb)	522.9	506.9	524.4	524.1
Dome Pressure (psia)	1012	1012	992	982
	30%P/60%F	25%P/60%F	30%P/50%F	25%P/50%F
Power (MWt / % of RTP)	1046 / 30	872 / 25	1046 / 30	872 / 25
Flow (Mlb/hr / % rated)	65.1 / 60	65.1 / 60	54.3 /50	54.3 / 50
Steam Flow (Mlb/hr)	3.708	3.048	3.704	3.044
FW Temperature (°F)	267.2	255.8	267.2	255.8
Core Inlet Enthalpy (Btu/lb)	522.6	524.8	519.0	521.7
Dome Pressure (psia)	974	972	974	972

Table 3-3 MELLLA Transient Analysis Results at RTP Conditions

Initial Condition	Event	Peak Neutron Flux (% NBR)	Peak Heat Flux (% Initial)	GE14 ΔCPR		ATRIUM-10 ΔCPR	
				Option B	Option A	Option B	Option A
100% RTP 106% RCF	LRNBP	275.00	111.00	0.30	0.33	0.30	0.33
	TTNBP	277.89	110.77	0.30	0.33	0.30	0.33
	FWCF	210.47	113.71	0.27	0.30	0.27	0.30
100% RTP 80.7 % RCF	LRNBP	269.13	112.41	0.28	0.31	0.26	0.29
	TTNBP	275.43	112.31	0.28	0.31	0.26	0.29
	FWCF	260.32	117.74	0.25	0.28	0.23	0.26

Table 3-4 MELLLA Transient Analysis Results at RTP Conditions

Initial Condition	Event	Peak Vessel Pressure (psig)	GE14 TOP/MOP		ATRIUM-10 TOP/MOP	
			TOP	MOP	TOP	MOP
100% RTP 106% RCF	LRNBP	1260	22.1	22.1	22.2	22.2
	TTNBP	1260	21.7	21.7	21.8	21.8
	FWCF	1168	18.1	18.7	16.8	19.3
100% RTP 80.7 % RCF	LRNBP	1260	19.9	20.2	19.6	19.8
	TTNBP	1260	19.6	19.9	19.3	19.7
	FWCF	1175	20.7	21.1	19.9	20.9

Table 3-5 Power Dependent Analysis Summary - Above P_{Bypass} at EOC

Power (%)	Limiting Transient	Peak Neutron Flux (% NBR)	Peak Heat Flux (% Initial)	GE14 ΔCPR		Atrium-10 ΔCPR	
				Option B	Option A	Option B	Option A
85	FWCF	380.1	133.8	0.45	0.62	0.45	0.62
60	FWCF	129.6	116.0	0.49	0.52	0.46	0.49
45	FWCF	102.8	119.4	0.54	0.57	0.51	0.54
30	FWCF	59.9	119.5	0.51	0.54	0.48	0.51

Table 3-6 Power Dependent Analysis Summary - Below P_{Bypass} at EOC with EOC-RPTOOS

Power (%)	Flow (%)	Limiting Transient	Peak Neutron Flux (% NBR)	Peak Heat Flux (% Initial)	GE14 Option A ΔCPR	Atrium-10 Option A ΔCPR
30	>50%	LRNBP	71.6	154.1	0.92	0.83
30	≤50%	LRNBP	74.5	147.9	0.94	0.81
25	>50%	FWCF	48.9	180.3	0.98	0.96
25	≤50%	LRNBP	53.8	152.4	1.02	0.83

Table 3-7 Power Dependent Analysis Summary - Below P_{Bypass} at EOC with TBVOOS and EOC-RPTOOS

Power (%)	Flow (%)	Limiting Transient	Peak Neutron Flux (% NBR)	Peak Heat Flux (% Initial)	GE14 Option A ΔCPR	Atrium-10 Option A ΔCPR
30	>50%	FWCF	99.8	211.9	1.61	1.45
30	≤50%	FWCF	95.8	201.2	1.43	1.30
25	>50%	FWCF	79.3	238.8	1.95	1.80
25	≤50%	FWCF	75.0	221.3	1.81	1.63

Table 3-8 Power Dependent MCPR Limits for GE14 and Atrium-10

Application Group	MCPRp Below P_{Bypass} (1)				Kp Multiplier Above P_{Bypass} (2)				
	25% P ≤50% F	25% P >50% F	30% P ≤50% F	30% P >50% F	30% P	45% P	60% P	85% P	100% P
1	2.24	2.24	2.15	2.15	1.483	1.280	1.150	1.072	1.000
2	2.24	2.24	2.15	2.15	1.483	1.280	1.150	1.072	1.000
3	3.12	3.28	2.69	2.89	1.483	1.280	1.150	1.072	1.000
4	3.12	3.28	2.69	2.89	1.483	1.280	1.150	1.072	1.000
Generic	NA	NA	NA	NA	1.483	1.280	1.150	1.056	1.000

Notes:
Application Group 1: Equipment in Service
Application Group 2: EOC-RPT OOS
Application Group 3: TBV OOS
Application Group 4: EOC-RPT OOS + TBV OOS

(1) MCPR(P) below P_{Bypass} are CGS plant-specific OLMCPR values.

(2) K(85%) does not bound the ARTS generic value, therefore CGS specific value is reported. The more limiting generic values are reported for all other Kp multipliers above P_{Bypass}.

Table 3-9 Power Dependent LHGR Limits for GE14 and Atrium-10

Application Group	LHGRFACp Below P_{Bypass} (1)				LHGRFACp Multiplier Above P_{Bypass} (2)				
	25% P ≤50% F	25% P >50% F	30% P ≤50% F	30% P >50% F	30% P	45% P	60% P	85% P	100% P
1	0.608	0.608	0.634	0.634	0.634	0.713	0.791	0.922	1.000
2	0.608	0.608	0.634	0.634	0.634	0.713	0.791	0.922	1.000
3(1)	0.480	0.480	0.524	0.524	0.634	0.713	0.791	0.922	1.000
4(1)	0.480	0.480	0.524	0.524	0.634	0.713	0.791	0.922	1.000
Generic	NA	NA	NA	NA	0.634	0.713	0.791	0.922	1.000

Notes:
Application Group 1: Equipment in Service
Application Group 2: EOC-RPT OOS
Application Group 3: TBV OOS
Application Group 4: EOC-RPT OOS + TBV OOS

(1) LHGRFAC(P) below P_{Bypass} are calculated CGS plant-specific values.

(2) LHGRFAC(P) above P_{Bypass} the more limiting generic values are reported.

Table 3-10 Flow Dependent Analysis Summary - All Application Groups

Power (%)	Flow (%)	GE14 ΔCPR	ATRIUM-10 ΔCPR
105	100	0.03	0.02
98	90	0.06	0.05
90.7	80	0.09	0.08
83.1	70	0.13	0.12
75.3	60	0.17	0.15
67.3	50	0.20	0.19
59	40	0.24	0.22
50.4	30	0.27	0.26

Table 3-11 Flow Dependent MCPR(F) Limits for GE14 and Atrium-10

30% F	90% F	108.5% F
1.65	1.25	1.25

Table 3-12 Flow Dependent LHGRFAC(F) Limits for GE14 and Atrium-10

Maximum Flow Limit (% RCF)	LHGRFAC(F) Limit Formula
102.5	MIN{ 1.0, [0.4860 + 0.6784 x (W_c / 100)]}
107.0	MIN{ 1.0, [0.4574 + 0.6758 x (W_c / 100)]}
112.0	MIN{ 1.0, [0.4214 + 0.6807 x (W_c / 100)]}
117.0	MIN{ 1.0, [0.3828 + 0.6886 x (W_c / 100)]}

Note: W_c = % Rated Core Flow

4.0 ROD BLOCK MONITOR SYSTEM IMPROVEMENTS

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

(a) Initiating a rod block to prevent violation of the fuel integrity safety criteria during withdrawal of a single control rod, and

(b) Providing a signal to permit operator evaluation of the change in local relative power during control rod movement.

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

4.1 Current Rod Block Monitor System Description

The generic RBM system descriptions in Sections 4.1, 4.2 and 4.4 are obtained from Reference 8.

4.1.1 Current System Description

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

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

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

The RBM has three drive flow-biased trip levels (rod withdrawal permissive removed). The trip levels may be adjusted and are nominally 8% of reactor power apart. Current CGS settings are 106%, 98%, and 90% CLTP at 100% flow. Each trip level is automatically varied with recirculation system flow to protect against fuel overpower at lower flows. The operator may encounter any number (up to three) of the trip points, depending on the starting power of a given control rod withdrawal. The lower two points may be successively bypassed (acknowledged) by manual operation of a pushbutton. The reset permissive is actuated (and indicated by a light) when the RBM indicates a power within the reset band of the trip point. The operator then assesses the local power and either acknowledges or selects a new rod. The highest trip point cannot be bypassed.

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

One RBM channel may be manually bypassed by operator action. Automatic bypass occurs if the APRM level is below a prescribed value or reactor core outer boundary control rods are selected.

An illustration of the current CGS RBM system is presented in Figure 4-1.

4.1.2 Limitations of Current Rod Block Monitor System

Since the 1960s, there have been significant technological advances in the field of two-phase heat transfer. The GE Critical Boiling Length (GEXL) Critical Power Ratio has replaced the Hench-Levy Critical Heat Flux Ratio as the approved means of determining departure from nucleate boiling. This means that optimum evaluation of fuel thermal margins is not as effective when performed solely on a local basis, compared against information about the entire fuel bundle. For the RBM to fulfill its intended function more effectively, changes in the RBM signal(s) must correlate closely with the thermal margin changes during control rod withdrawal. The current RBM signals do not always correlate well with thermal margin changes during control rod withdrawal, and the system performs its function at the expense of significant operational penalties due to the conservatism required by the current limitations.

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

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

Figure 4-3 illustrates an ideal startup path in which rated power is attained without control rod movement after recirculation flow has been increased. Figure 4-3 also shows the relationship between the RBM trip settings and the ideal startup path relative to the highest RBM trip setting. Because these two lines cross at low flow, the RBM prevents withdrawal of control rods necessary to attain the ideal startup path, thus control rods must be withdrawn at higher core power where fuel thermal margins may be smaller and more difficult to achieve.

Table 4-1 summarizes the limitations of the current CGS RBM system, the effects of these limitations, and the proposed improvements to the system.

4.2 ARTS-Based Rod Block Monitor System Description

The ARTS Based RBM system will:

(a) Eliminate the restrictions imposed on gross core overpower by the current flow-referenced RBM trips (this function is fulfilled by the APRM flow-biased rod block), and

(b) Enhance operator confidence in the system by reducing the frequency of nonessential rod blocks and by making the occurrence of rod blocks more predictable and therefore avoidable.

The following is a description of the functional changes to the RBM for ARTS.

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

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

(a) Similarity of channel responses,

(b) High response to rod motion (allows higher setpoints, which reduces the effect of random signal noise, calibration inaccuracies, and instrument drift),

(c) Less restrictive MCPR limits with high setpoints,

(d) High availability (tolerance of LPRM failures), and

(e) Ease of implementation.

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

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

To the maximum extent possible, while achieving the above objectives, the new RBM system design meets the same separation and isolation requirements as the previous RBM system. The only exceptions are the sharing of LPRM signals from the "C" level detectors by both RBM channels and the calibration of the RBM signals to isolated, fixed reference signals instead of isolated APRM reference signals. As for the current system, the new RBM system is fail safe for failed LPRM input signals. As for the current system, a count of active LPRMs is made automatically and the RBM channel declared inoperative if too few detectors are available.

The impact on the availability of the new RBM system due to the sharing of the "C" level detectors has been shown to be small and the benefits of the improved signal response outweigh any perceived loss in signal redundancy.

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

The RBM has no safe shutdown function, and cannot prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.67.

The RBM is a system that mitigates the consequences of an RWE by automatically initiating a rod block to ensure that the MCPR safety limit is not exceeded. The RWE is not an accident. It is an AOO, which, as defined in 10 CFR 50, Appendix A is part of normal operations. An RWE does not challenge the integrity of the reactor coolant pressure boundary, and thus, the RBM is not used to maintain the integrity of the reactor coolant pressure boundary.

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

4.3 Rod Withdrawal Error Analysis

4.3.1 Analysis

The improved RBM system for CGS with power-dependent setpoints requires that new RWE analyses be performed to determine the MCPR requirements and corresponding setpoints. A generic statistical analysis for application to all BWRs including CGS has been performed and is summarized in Table 4-2. The application of these results is validated for GNF and/or ATRIUM-10 fuel and core design for each reload analysis in accordance with the Reference 2 CPR correlation.

The generic ARTS RWE database in Table 4-2 was drawn from actual plant operating states and covers the spectrum of plant designs and power densities (BWR/2, 3, 4, and 5) and BP/P8x8R fuel designs. Cases were selected with low MCPRs and high LHGRs in bundles near deep control rods to yield meaningful results. Three operating state case groups were examined in the generic studies. All State A cases were selected near rated power and rated flow. The actual rod patterns were modified to reduce the MCPR(s) of bundle(s) near the deep rods to approximately 1.20. To cover the P/F map, two other P/F points were included in the database. State B was obtained from the State A case utilizing the same rod pattern and a core flow of 40% of rated.

This represents an equilibrium xenon power level of about 60% of rated. State C represents a modification of the State B case rod pattern to a 40% power condition (with 40% of rated core flow) with no xenon. The total database consisted of 91 cases (39 State A, 26 each State B and C).

The RWE analyses were performed utilizing the approved models described in Reference 2. The outputs (MCPR and LPRM readings, and gross core power as a function of error rod position) were inputs to the statistical analysis. From each case studied, 100 simulated RWEs were generated by randomly varying the initial position of the error rod and the location and number of failed LPRMs. Only initial error rod positions that were either fully inserted or that required a rod block to limit MCPR were considered, and a random failure probability of 15% was assigned to each LPRM. The 15% failure ratio is atypically high based on evaluations of actual operating experience. A sensitivity study was also performed on LPRM failures (Subsection 4.3.2.2) that show that the new system is fairly insensitive to LPRM failure rates.

The RBM responses were generated for both channels for each RWE analyzed. From these responses, the error rod position at the rod block trip level was generated as a function of RBM setpoint. The results were tabulated as a function of RBM setpoint. The parameter of interest is the normalized MCPR change, i.e., Delta Critical Power Ratio over Initial Critical Power Ratio (ΔCPR/ICPR). From the 100 RWEs analyzed for each rod pattern, the mean and standard deviation and components of the standard deviation were calculated for each RBM setpoint, which were then used to determine the mean and standard deviation of the entire database at each State A, B, and C.

The overall results were determined for each P/F point for each RBM channel. The limiting parameter is the MCPR, and a value of $(\Delta CPR/ICPR)_{95/95}$ for each channel for each setpoint was determined which is expected to bound 95% of the RWE consequences with 95% confidence. The initial MCPR necessary to provide 95% confidence that the SLMCPR will not be violated in 95% of the RWEs initiated from that value is:

$$MCPR_{95/95} = \frac{SLMCPR}{1 - (\Delta CPR/ICPR)_{95/95}}$$

The results for both RBM channels for each P/F state for a range of RBM setpoints are summarized in Table 4-2, which also shows the bounding MCPR requirement for each setpoint. This bounding MCPR requirement was used to generate the design basis MCPR requirement as a function of the RBM setpoint (Figure 4-8).

The results in Table 4-2 show that, for setpoints of interest, the MCPR limits do not vary significantly over the P/F map. The primary parameters affecting an RWE are initial rod pattern and void fraction. Because these parameters are essentially fixed along a given flow control line, [[

]]

The generic ARTS results presented thus far were performed utilizing an SLMCPR of 1.07. In order to accommodate any potential future change in the SLMCPR, the RBM setpoints are selected based on the limiting rated ΔCPR. The limiting rated ΔCPR is that value, which when added to the plant SLMCPR, establishes the rated plant OLMCPR. Power-dependent RBM setpoints shown in Table 4-5 were determined based on the power-dependent MCPR requirements (Table 3-8). Table 4-5 is provided so that appropriate setpoints can be selected such that the RWE will not significantly limit plant operation. These RBM setpoints are analytical values and verified to be applicable to CGS. Figure 4-8 and Figure 4-9 were used to determine the RBM analytical setpoints such that the RWE required MCPR is less than or equal to the core-wide transient power-dependent MCPR requirement. The resultant power dependent RBM setpoint requirements for a 1.20 rated MCPR limit are shown in Figure 4-10.

The generic RWE analyses also verified the conformance to the fuel thermal-mechanical limit (i.e., 1% plastic strain) for GNF fuel designs. Plant-specific RWE evaluations have been performed for CGS using the reference core loading for Cycle 20, which included GE14 and ATRIUM-10 fuel. The results show that the SLMCPR and 1% cladding plastic strain fuel safety limit criteria are met. Specifically, the RWE MCPR requirement for the CGS ARTS/MELLLA evaluation is 1.27, compared to an EOC Option B OLMCPR of 1.39 from the LRNBP. In addition, calculations will be performed as part of the reload analyses in accordance with Reference 2 to confirm the applicability of the ARTS based statistical RWE result for subsequent fuel cycles at CGS. If the confirmatory RWE calculation is more limiting than the generic 95/95 requirement, then the cycle-specific RWE MCPR requirement will be based on the RWE calculation.

4.3.2 Sensitivity Analyses

4.3.2.1 Peripheral Rod Groups

The RBM setpoints discussed above were based on analysis of RWEs occurring in four-rod cells surrounded by four LPRM strings. The RBM cells near the core periphery may possess fewer than four control rods and have one, two, or three LPRM strings.

A study was performed to verify that the results obtained in the previous sections are valid for peripheral cells with less than four LPRM strings. The locations of LPRM strings and control rods in the CGS core are shown in Figure 4-11. The rod group geometries and error rods studied are shown in Figure 4-12. A single case was selected from the database used to establish the RBM setpoints. This case was re-analyzed with the various geometries of Figure 4-12 substituted for the standard four-string geometry. For this study, the RBM setpoint was fixed at 108%. Results of the study (Table 4-3) show no significant differences between the base (four string) case and the limiting peripheral geometries. [[

]]

[[

]]

4.3.2.2 Local Power Range Monitor Failures

[[ ]] A study was performed to determine the sensitivity of the MCPR requirement to the failure probability. Failure probabilities of 0%, 15%, and 30% were evaluated for a 10-case subset of the 39 full-power cases. [[ ]] A low sensitivity to LPRM failure probability is demonstrated in this figure. It is concluded that the RBM setpoints are adequate for any realistically expected incidence of LPRM failures.

4.3.2.3 Effect of Filter on Rod Block Monitor Signal

[[

1

]]

4.3.3 Effect of Safety Limit and Critical Power Ratio Correlations on Rod Withdrawal Error Results

Generic ARTS results presented in the sections above were achieved utilizing the original GEXL correlation and a SLMCPR of 1.07. A sensitivity study has been performed to assess the effect of the GEXL-PLUS correlation (applicable to ATRIUM-10 and GE14 fuel in CGS core) on these generic RWE results because of the greater sensitivity of GEXL-PLUS to power distribution changes. Core designs were evaluated at rated conditions under equilibrium xenon conditions. RWE results were calculated using both the GEXL and GEXL-PLUS correlations initiated from identical core exposure distributions and control rod pattern conditions. Differences in the required ARTS RWE MCPR limits were less than 0.01 (with the GEXL-PLUS limits above the GEXL limits) for all proposed RBM setpoints. Transient analysis results in Section 3.0 show the OLMCPR associated with operation at ICF conditions as well as in the MELLLA region. A comparison of these ΔCPRs with limiting ARTS RWE ΔCPR of 0.18 (corresponding to the RBM setpoints in Table 4-5) indicates that a minimum margin of 0.12 exists before the RWE event would become limiting in terms of establishing OLMCPR. These margins are more than adequate to accommodate the calculated increase in RWE severity due to GEXL-PLUS correlations. Furthermore, these results are verified as part of the cycle-specific reload analyses in accordance with Reference 2.

4.4 Filter And Time Delay Settings

The ARTS based RBM system has the capability to include two adjustable time delays and two adjustable signal filters.. The first filter on the RBM signal (T_{c1}) smoothes the averaged LPRM signal to reduce trips due to signal noise. A second filter (T_{c2}) on the APRM signal input to the power-dependent trip selection logic was provided on pre-NUMAC™ ARTS implementations to improve the accuracy of the trip selection logic by reducing noise and oscillation between setpoints. For the CGS NUMAC™ RBM implementation, this filter (T_{c2}) is eliminated because the incoming APRM signal is the STP signal, which already has a 6 second filter on it.

---

[1] The setpoints here are "Analytical Limits;" other adjustments are recommended for inaccuracy, calibration, and drift effects to obtain the "Nominal Trip Setpoint." Some adjustment ranges have been fixed by design such that surveillance can be performed by simply establishing that the adjustments are in the limiting position.

The first delay, T_{d1}, delays gain adjustment and signal normalization for a preset time interval following rod selection and is necessary to allow the filtered RBM signal to approach its asymptotic value. (No rod withdrawal is possible during this period.) For optimum performance based on experience from plants operating with the ARTS based RBM, this time delay (T_{d1}) has been hard coded into the NUMAC™ at 10 times T_{c1}, and is not user adjustable. The second delay, T_{d2}, is between the time the signal is nulled to the reference and the time the signal is passed on to the trip logic (withdrawal is not restricted during this interval).

The adjustable trip time delay (T_{d2}) is designed to allow for both a noise reduction feature and for a system bypass function when sufficient fuel margins are available. The following discussion focuses on the justification for the adjustable trip time delay (T_{d2}) as a means of bypassing the RBM system when permitted.

For applications when extreme signal noise characteristics exist, the signal noise may be too severe for a filtering system to handle adequately (i.e., the required filter time lag setpoint penalty would result in setpoints too low to be operationally acceptable). The ARTS based RBM includes an adjustable trip time delay (T_{d2}) that interrupts the transmission of the RBM signal for a specified time period beginning with the rod withdrawal permissive following successful nulling of the signal to the reference value. The purpose of this delay is to allow a plant that is within thermal limits to withdraw a control rod at least a single notch despite extremely noisy signals that would normally block rod withdrawal. Therefore, specifications of standard RBM setpoints coupled with this time delay would assure that at least one 6-inch notch control rod withdrawal could be made on each rod selection.

The time delay option (T_{d2}) will not be used at CGS because additional supporting analyses for T_{d2} are required but have not been included as part of this evaluation. When and if T_{d2} is utilized, analyses will be performed under the CGS design process based on unrestricted continuous rod withdrawal during the T_{d2} period. Preliminary evaluations include the feasibility of a value of T_{d2} of approximately 10 seconds. The inclusion of this feature is considered totally consistent with the ARTS objective of eliminating unnecessary RBM rod block alarm on normal rod maneuvers in order to improve the human factors of the RBM system.

The ARTS RBM licensing bases support any combination of the adjustable RBM filter time constant (T_{c1}) and the null sequence delay time (T_{d1}) with the applicable adjustment setpoints defined in Tables 4-5. However, time delay T_{d1} has been hard coded at 10 times T_{c1}, and is not user adjustable. If RBM filtering is required, the nominal setting will be determined based on plant conditions. The maximum time constant setting of 0.55 seconds will result in a null sequence time delay of 5.5 seconds. The trip setpoints and power intervals are defined in Tables 4-5 and 4-6 and shown in Figure 4-14.

4.5 Rod Block Monitor Operability Requirement

The RBM system design objective is to block erroneous control rod withdrawal initiated by the operator before the SLMCPR is violated. If any control rod in the core threatens to violate this limit upon complete withdrawal, operability of the RBM system is required. The RBM system basis is limited to consideration of single control rod withdrawal errors and does not accommodate multiple errors. Therefore, in defining "limiting control rod patterns," only single control rod withdrawals are considered. The entire generic RWE analysis database was evaluated to determine the pre-RWE MCPR margin that would assure that the complete withdrawal of any single control rod from any initial position would not violate the safety limit.

The requirements were evaluated at the 95% probability and 95% confidence level as follows: First, the 95/95 maximum MCPR changes were determined for complete rod withdrawal:

$$\left(\frac{\Delta CPR}{ICPR}\right)_{95/95,\ \text{Full Withdrawal}}$$

Then, pre-RWE MCPR requirement was determined:

$$MCPR_{RBM}\ \text{Operation Required} = \frac{SLMCPR}{1 - \left(\frac{\Delta CPR}{ICPR}\right)_{95/95,\ \text{Full Withdrawal}}}$$

The following limiting MCPR values were determined to provide the required margin for full withdrawal of any control rod:

For Power < 90%: MCPR $\geq$ 1.70

For Power $\geq$ 90%: MCPR $\geq$ 1.40

Whenever operating MCPR is below the preceding values, the RBM system must be operable; whenever the operating MCPR is above these values, complete RBM bypass is supported. These MCPRs were developed utilizing a SLMCPR of 1.07, thus are conservative for lower values of SLMCPRs and must be adjusted for higher values of SLMCPRs.

For the higher CGS Cycle 20 safety limit of 1.09 these limits are 1.73 and 1.43 respectively.

4.6 Rod Block Monitor Modification Compliance to NRC Regulations and Licensing Topical Reports

Modifications to the RBM firmware will be performed, consistent with the quality requirements as addressed in Reference 9, Section 9, "Quality Assurance Programs." The RBM firmware was developed using the same Verification and Validation (V&V) program as previously reviewed by the NRC in NEDC-32410P-A (Reference 9). This program specifically addresses issues such as design control, change control, documentation, record keeping, independent verification, and software development specific requirements as delineated in NRC Regulatory Guide (RG) 1.152 (Reference 10). The basic approach of this V&V methodology is as follows: (1) the design process is divided into logical steps, starting at the top, with each step resulting in a documented output; (2) independent technical verification reviews are performed for each step of the design process, including verification of test methods and results; (3) the design steps are divided into logical groups, starting from the top, each of which comprise a baseline for the next step of design steps; (4) an independent process review is performed after each group of design steps to assure that the process, including technical verification reviews, is being followed and issues resolved; (5) a final comprehensive validation test is performed of the completed software in the target hardware; and (6) all steps of the process are documented. The existing qualification envelop for PRNM hardware is unchanged with the modification. Operator bench board changes have been reviewed and are adequate with the changes (see Section 2.3.3.6.2 of Reference 9).

4.7 Conclusion

The firmware change for the CGS NUMAC™ PRNM system and Technical Specification implementation of ARTS will:

- Eliminate the restrictions imposed on gross core power by the current flow-referenced RBM trips (this function will be fulfilled by the APRM flow-biased rod block).
- Enhance operator confidence in the system by reducing the frequency of nonessential rod blocks and by making the occurrence of rod blocks more predictable and avoidable.
- Upgrade the performance of the system such that the RWE will never be the limiting transient. The RWE transient MCPR is determined by the rod block setpoints. These setpoints will be selected based on the OLMCPR, as established by other AOOs.

Table 4-1 Rod Block Monitor System Improvements

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

Table 4-2 Rod Withdrawal Error Analysis Results

Setpoint	Channel	Approximate Power/Flow	(Δ/I) Mean	Δ /I Std Deviation	$MCPR_{95/95}$	Bounding $MCPR_{95/95}$
[[						
						]]

Table 4-3 RWE Analysis Results For Peripheral Rod Groups (108% Setpoint)

Number of Strings	Number of LPRM Inputs	BCCD$_1$ Channel $\frac{\Delta MCPR}{IMCPR}$		BCCD$_2$ Channel $\frac{\Delta MCPR}{IMCPR}$	
		Mean	Std. Dev.	Mean	Std. Dev.
[[					
					]]

Note: See Figure 4-5 for BCCD scheme of LPRM assignments

Table 4-4 RBM Signal Filter Setpoint Adjustment

Power/Flow (%/%)	RBM Channel	Number of Cases Evaluated	RBM Setpoint (%)	Mean Difference of Filtered and Unfiltered Signals Where Unfiltered Signals Equals Setpoint	Standard Deviation of Difference
[[					
					]]

Table 4-5 RBM System Setup

Function	Trip Level Setting (Note a)	
	Analytical Limit (AL) Unfiltered / Filtered	Allowable Value (AV) Unfiltered / Filtered
LPSP	30 / 30	28 / 28
IPSP	65 / 65	63/ 63
HPSP	85 / 85	83 / 83
LTSP	127.0 / 125.8	124.6 / 123.4
ITSP	122.0 / 121.0	119.6 / 118.6
HTSP	117.0 / 116.0	114.6 / 113.6
DTSP	N/L (Note b)	N/L (Note b)
T_{c1}	N/L (Note b)	N/L (Note b)
T_{c2}	N/A (Note c)	N/A (Note c)
T_{d1}	N/L (Note b)	N/L (Note b)
T_{d2}	N/L (Note b)	N/L (Note b)

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

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

Note (c): N/A – Not Applicable; this item is eliminated because filtering is provided by the STP APRM signal.

Table 4-6 ARTS RBM System Setpoints

ARTS Generic RWE MCPR Limit (SL=1.07/1.09)	Function	OLMCPR (SL = 1.09)	Trip Level Setting (%) (Without Filter)	Trip Level Setting (%) (With Filter)
1.20 /1.22	HTSP	1.27	108.0	107.4
	ITSP		112.0	111.2
	LTSP		118.0	117.0
1.25 / 1.27	HTSP	1.29	111.0	110.2
	ITSP		116.0	115.2
	LTSP		121.0	120.0
1.30 / 1.32	HTSP	1.32	114.0	113.2
	ITSP		119.0	118.0
	LTSP		124.0	123.0
1.35 / 1.37	HTSP	1.37	117.0	116.0
	ITSP		122.0	121.0
	LTSP		127.0	125.8

Table 4-7 RBM Setup Setpoint Definitions

AL	Analytical limit
AOO	Anticipated Operation Occurrence
AV	Allowable value
NTSP	Nominal trip setpoint
LPSP	Low power setpoint; RBM trips automatically bypassed below this level.
IPSP	Intermediate power setpoint
HPSP	High power setpoint
LTSP	Low trip setpoint
ITSP	Intermediate trip setpoint
HTSP	High trip setpoint
DTSP	Downscale trip setpoint to avoid an RBM trip if the readings occasionally decrease slightly as a rod is initially withdrawn.
T_{d1}	Delays the nulling sequence after rod selection so RBM filtered signal nears equilibrium before calibration. It adds an additional time delay from rod selection to allowable rod withdrawal start. The value is fixed at 10 times the T_{c1} input value.
T_{d2}	Adjustable Time delay 2 that delays passing RBM filter signal to RBM trip logic after signal has been nulled successfully to reference signal.
T_{c1}	Adjustable RBM signal filter time constant. Adjustment within the hardware capability must be consistent with the basis of the setpoints.
T_{c2}	Variable APRM signal filter constant. This filter is eliminated.
Reference Level	The level the RBM is automatically calibrated to upon control rod selection.

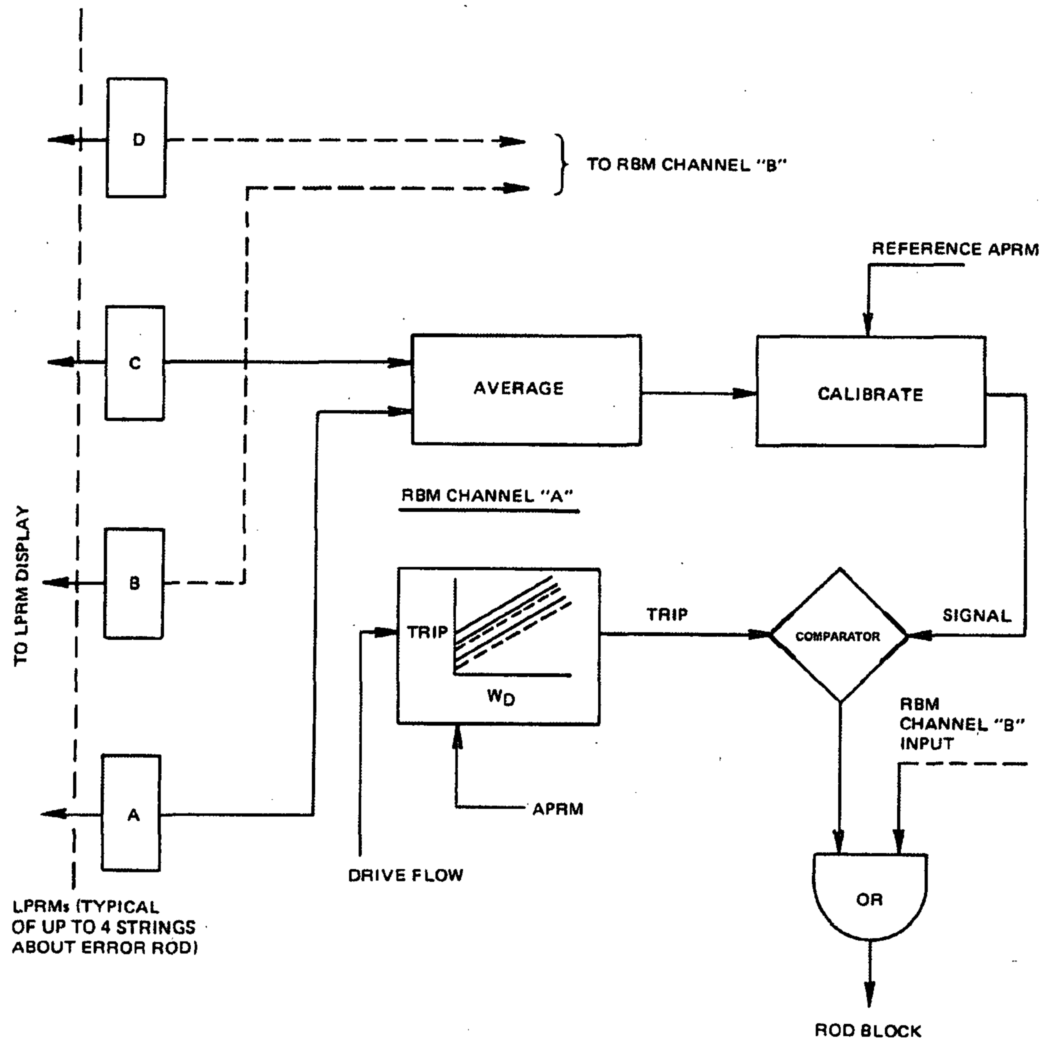


Figure 4-1 Conceptual Illustration of Current Flow-Dependent RBM System with AC/BD LPRM Assignment

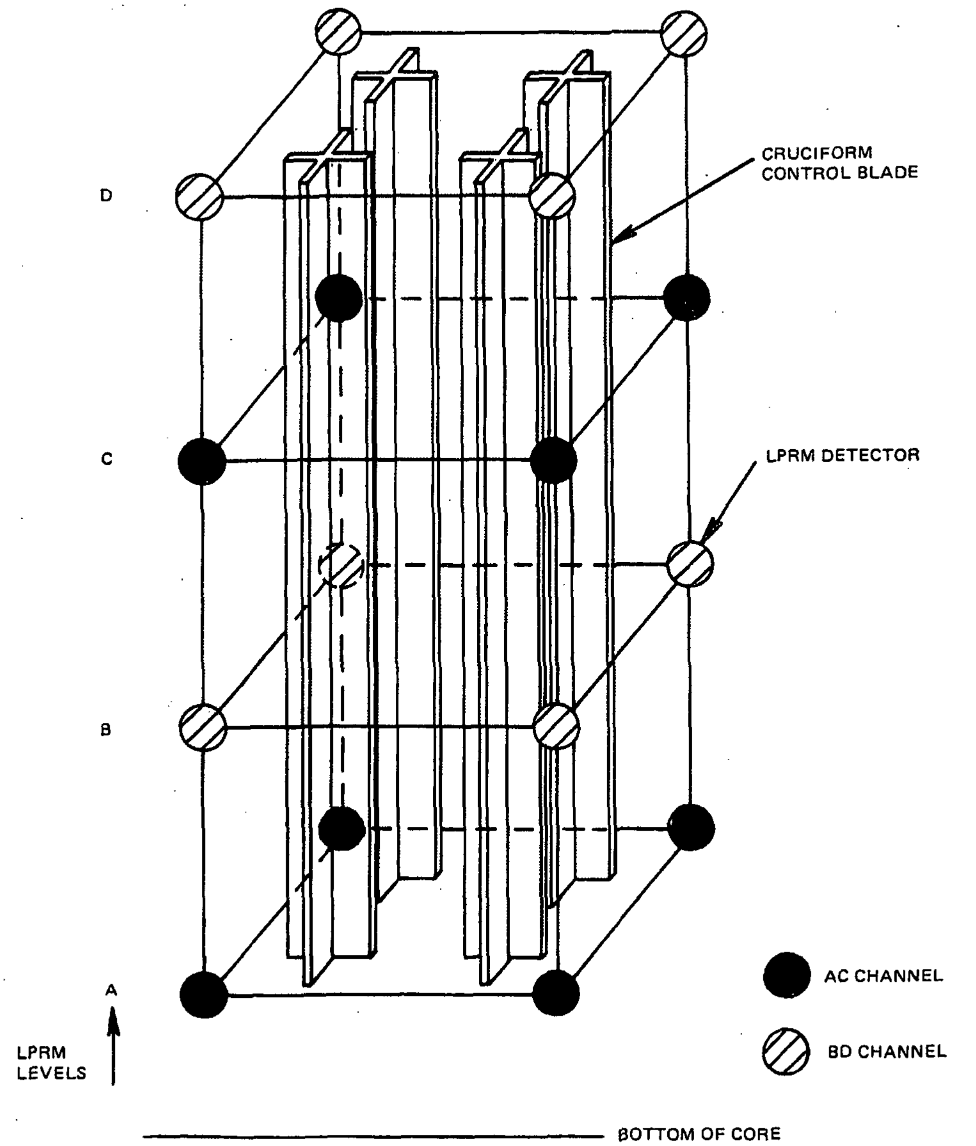


Figure 4-2 RBM Current AC/BD LPRM Assignment

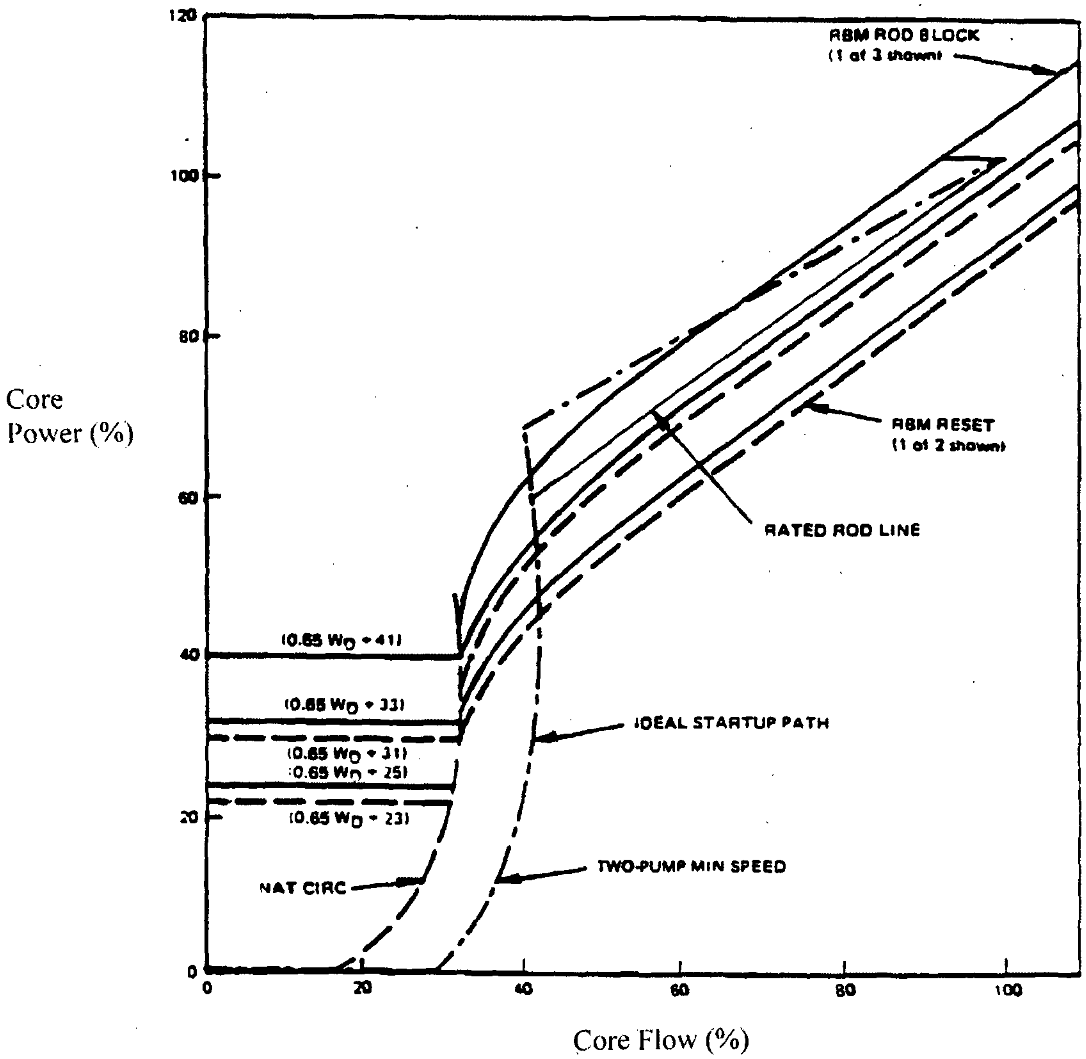


Figure 4-3 Typical RBM System Configuration Limits (Typical for 106% Setpoint)

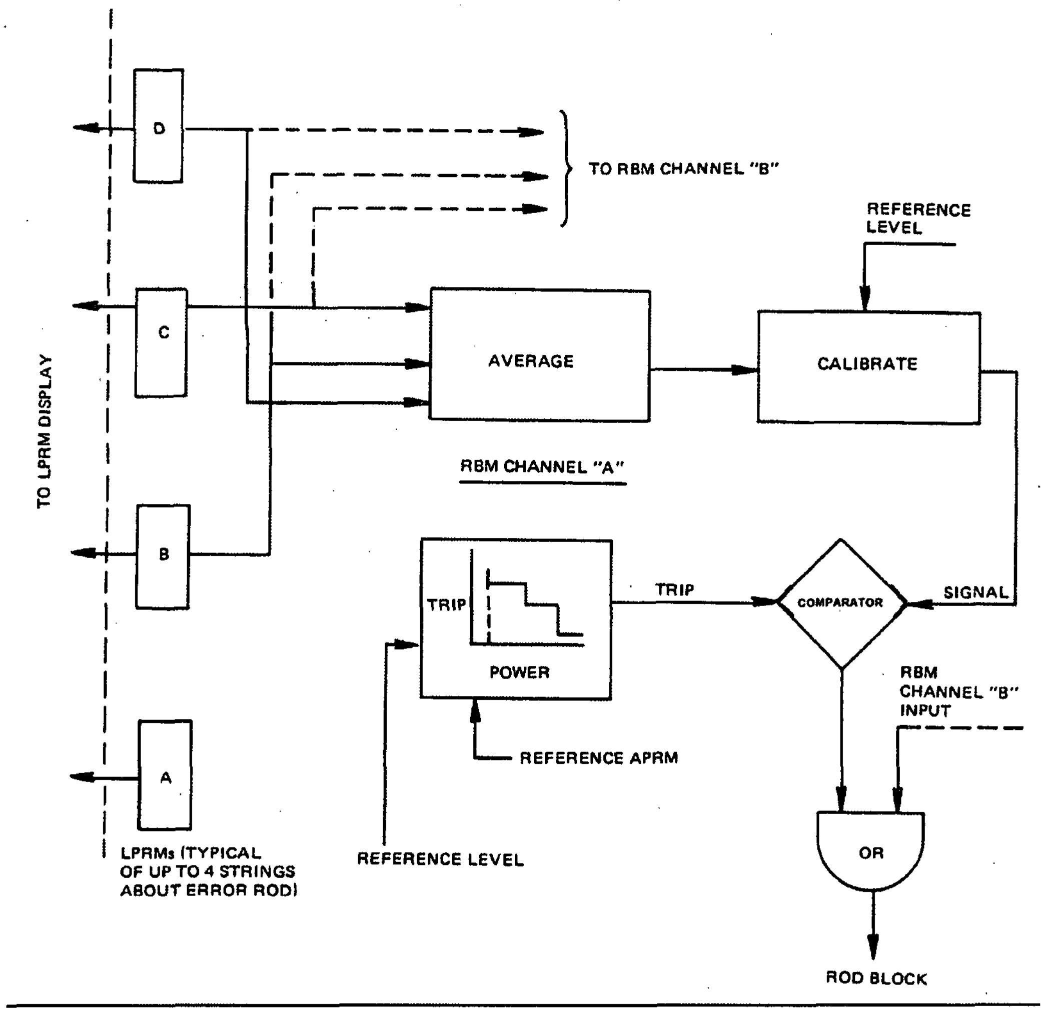


Figure 4-4 New Power-Dependent RBM System with BCCD$_1$/BCCD$_2$ LPRM Assignment

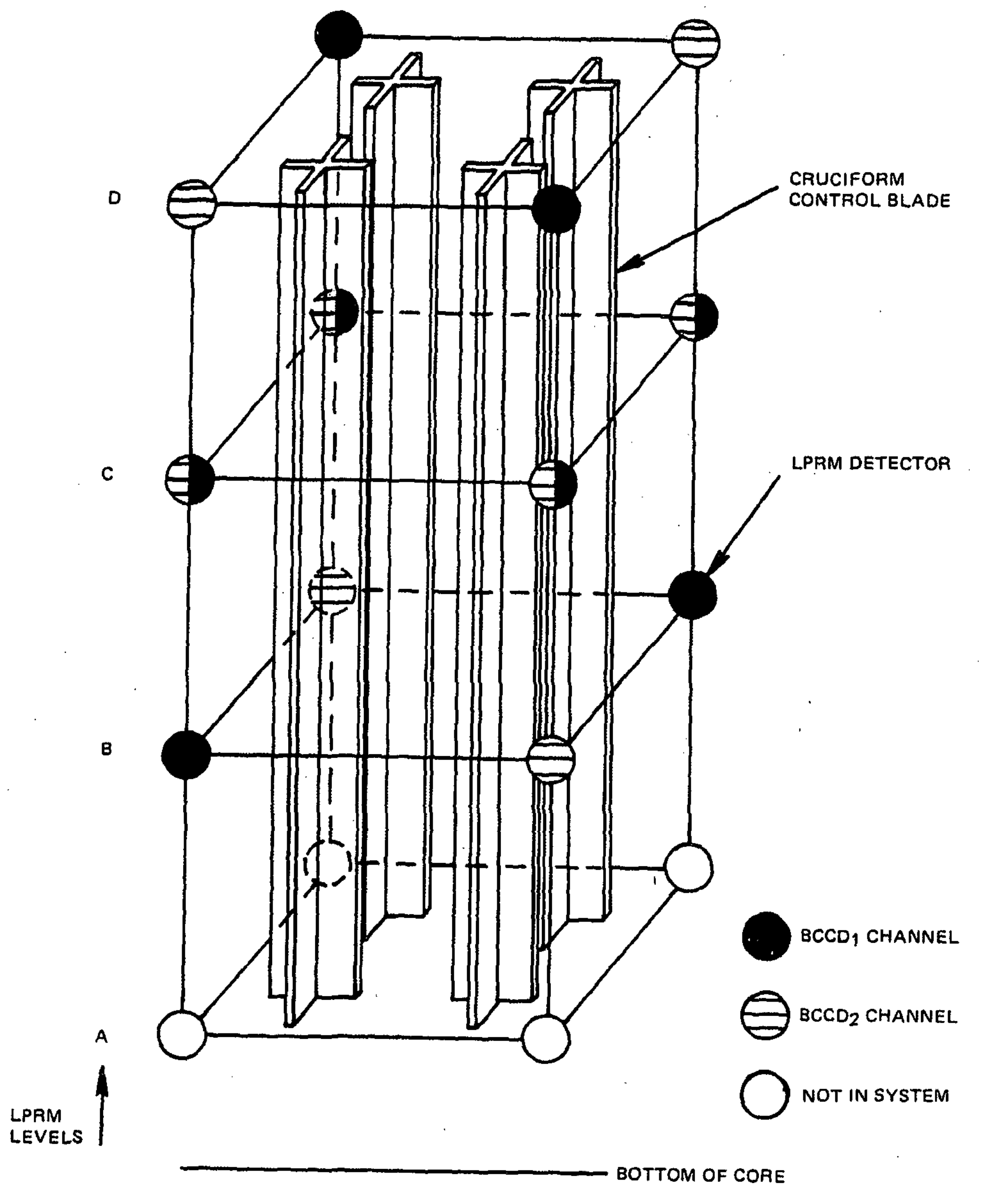


Figure 4-5 New RBM $BCCD_1/BCCD_2$ LPRM Assignment

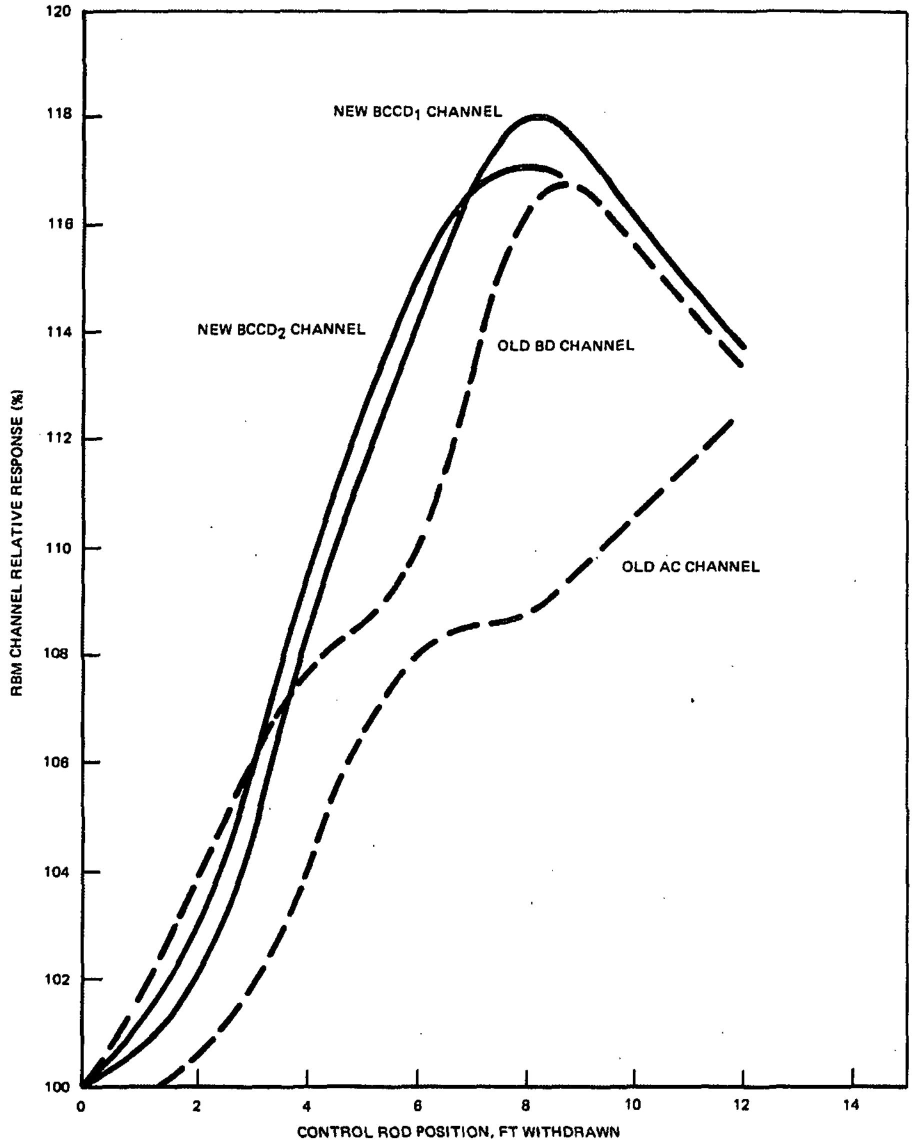


Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs)

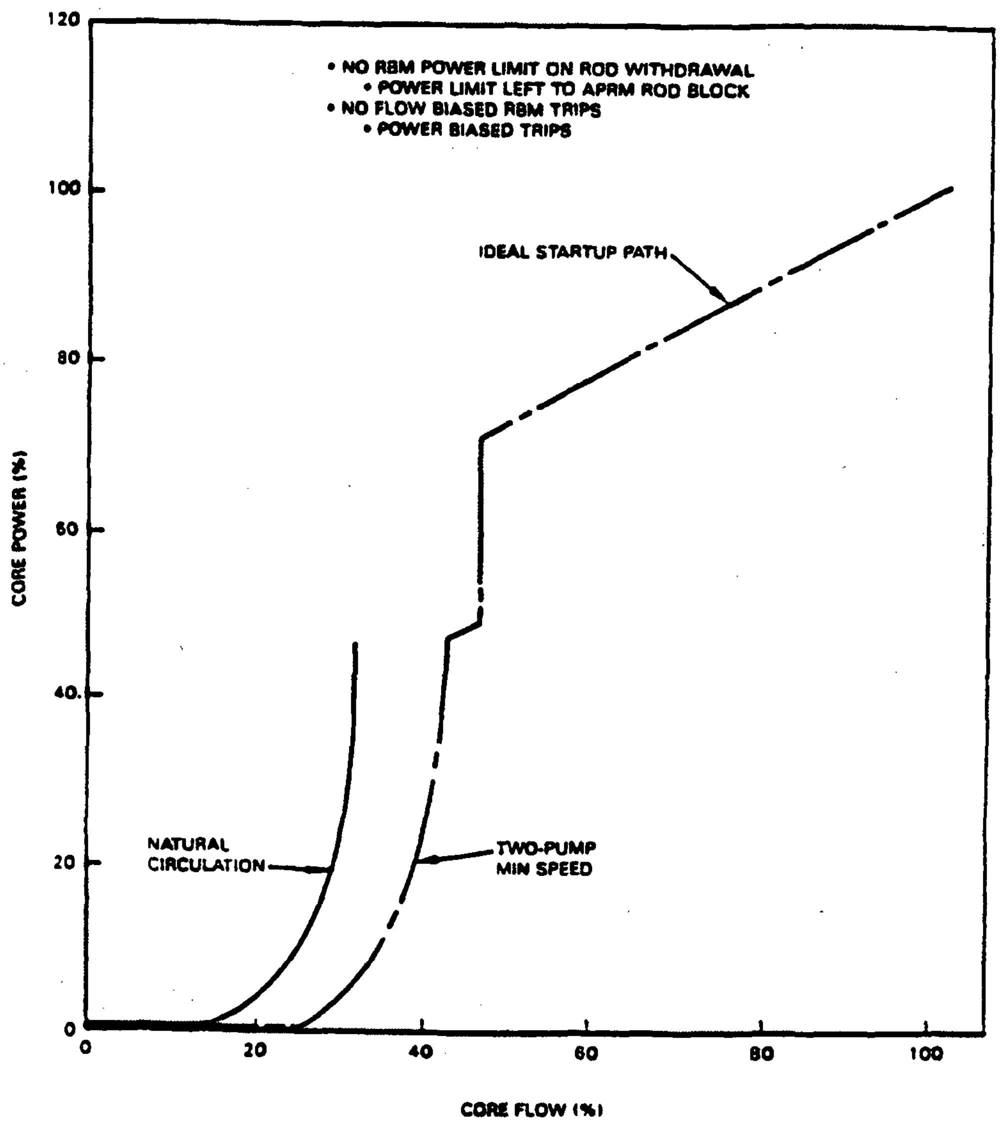


Figure 4-7 New RBM System Core Power Limit (Typical)

[[

]]

Figure 4-8 Design Basis RWE MCPR Requirement Versus RBM Setpoint

[[

]]

Figure 4-9 Design Basis MCPR Requirement for RWE (ARTS)

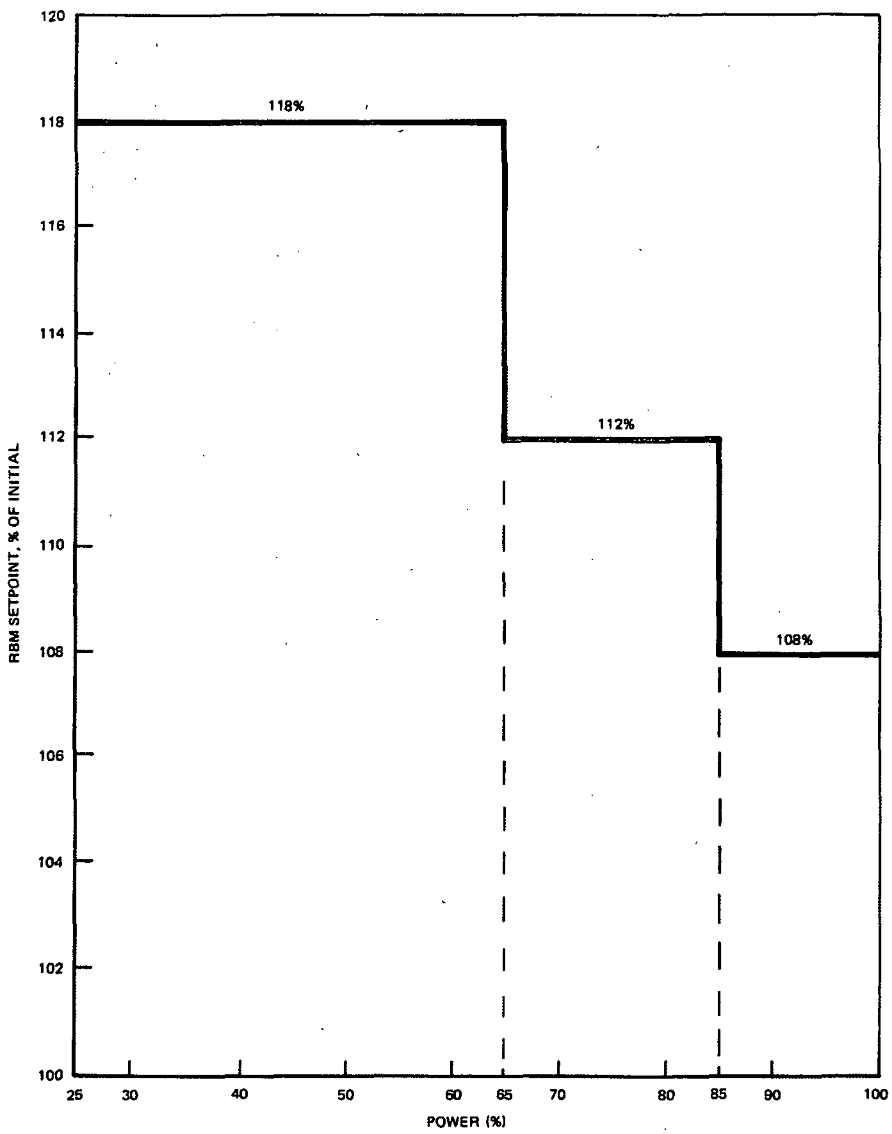


Figure 4-10 RBM Setpoint Versus Power (without Filter)

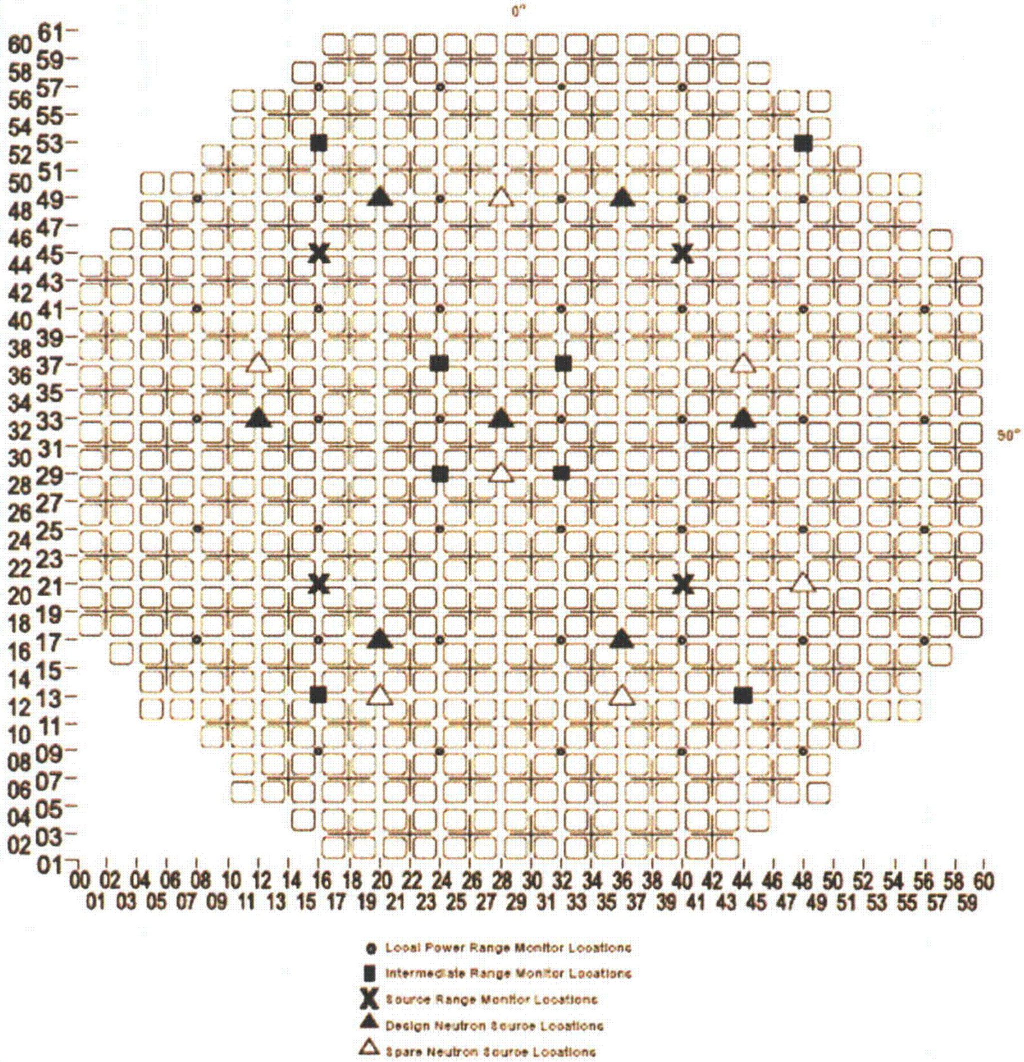


Figure 4-11 CGS Neutron Monitoring System

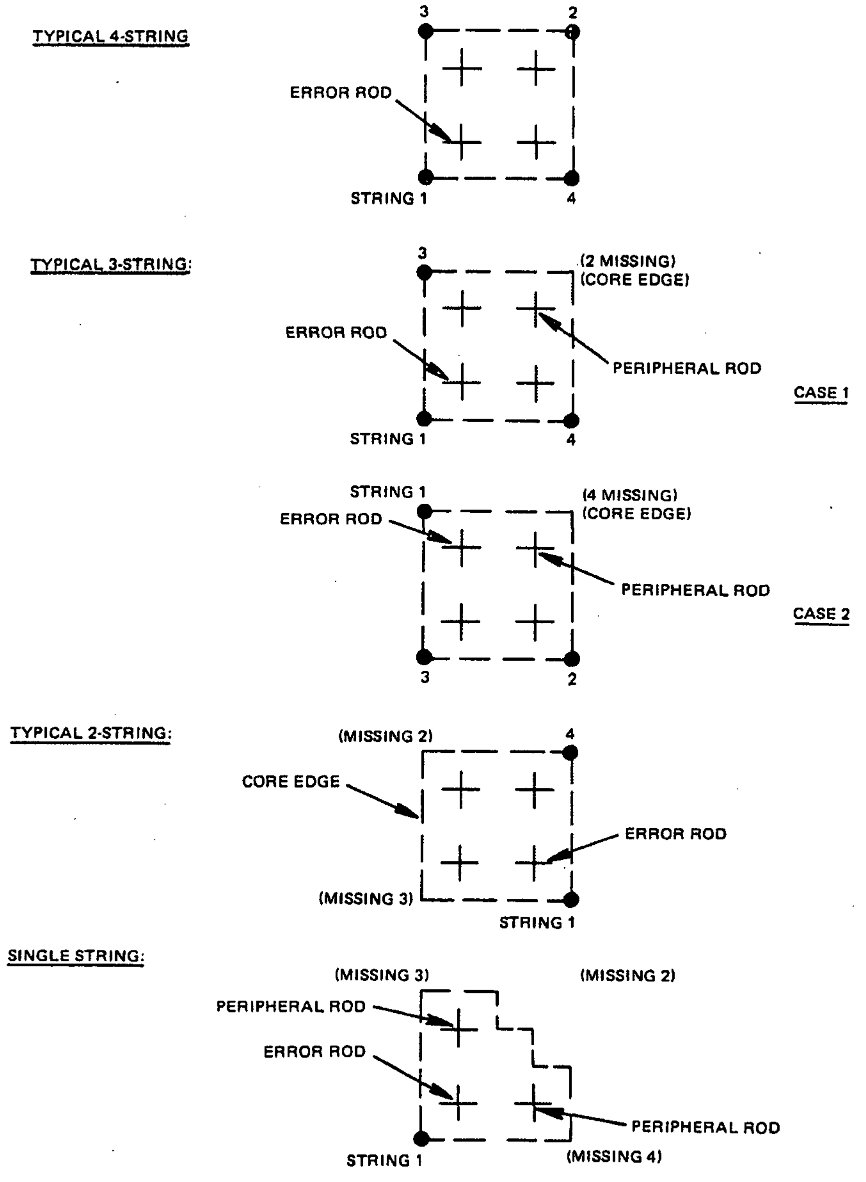


Figure 4-12 Rod Block Monitor Rod Group Geometries

[[

]]

Figure 4-13 Results of LPRM Failure Rate Sensitivity Studies

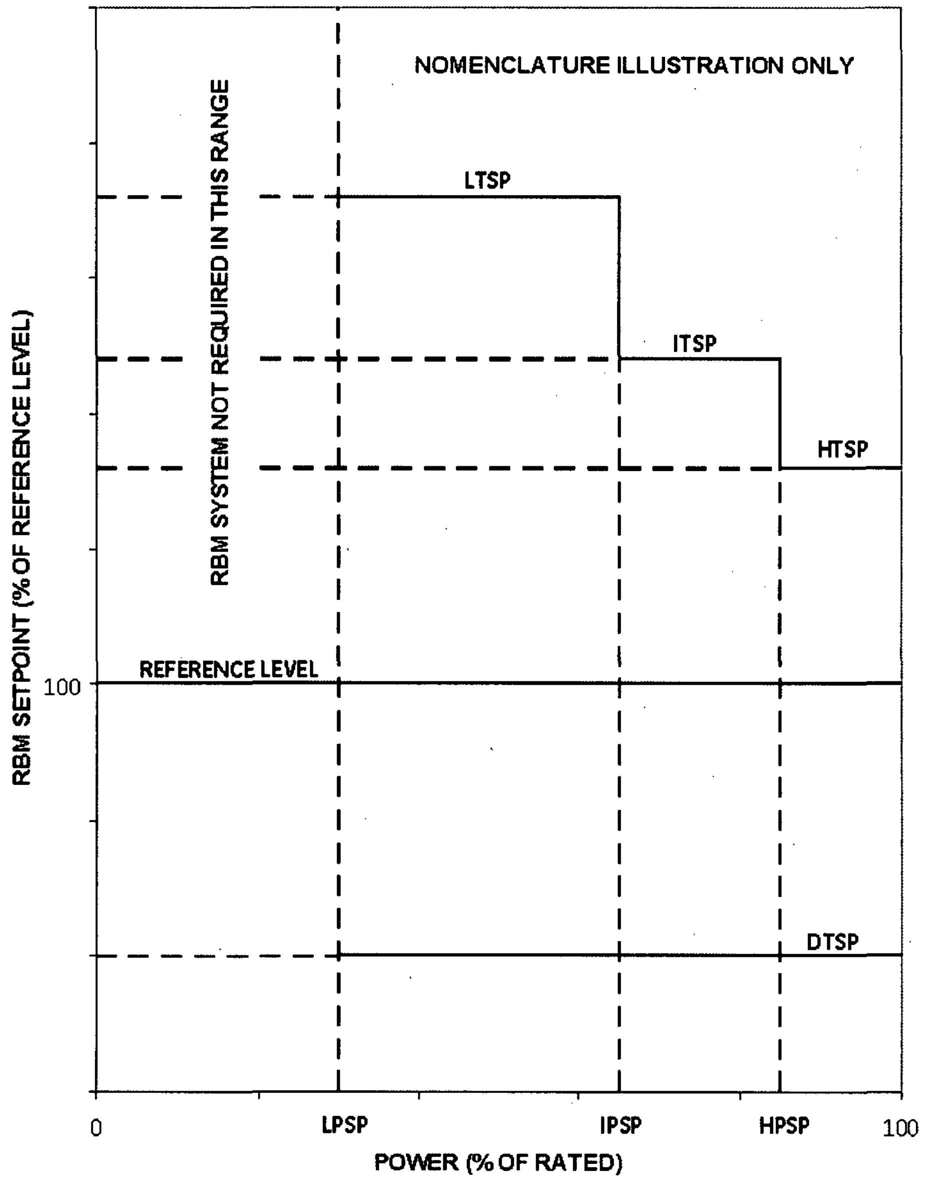


Figure 4-14 Power-Dependent RBM Trip Nomenclature

5.0 VESSEL OVERPRESSURE PROTECTION

The Main Steam Isolation Valve Closure with a Flux Scram (MSIVF) event is used to determine compliance to the American Society of Mechanical Engineers (ASME) Pressure Vessel Code. This event was previously analyzed at the 102%P / 106%F state point for the CGS Cycle 20 reload licensing transient analysis. This is a cycle-specific calculation performed in accordance with Reference 2 at 102% of RTP and the maximum licensed core flow (maximum flow is limiting for this transient for CGS). Because the implementation of ARTS/MELLLA does not change the maximum core flow, ARTS/MELLLA does not affect the vessel overpressure protection analysis. However, the sensitivity of operation at the MELLLA condition (102%P / 80.7%F for this analysis) for CGS Cycle 20 is provided in Table 5-1.

The MSIVF is the limiting event for the ASME overpressure analysis. Note that for the ASME overpressure analysis, the MSIVF includes an additional failure in the RPS system and is therefore not an AOO where MCPR is calculated.

The MSIVF results are primarily [[
]] associated with the cycle specific core design. A demonstration was provided in Table 5-1 that shows that the increased core flow condition (106% core flow) produces the more limiting peak vessel pressure for CGS. The higher initial core flow has a higher core pressure drop and a higher initial pressure in the lower plenum and results in higher peak vessel pressures. Therefore, MELLLA initial condition does not adversely affect the peak vessel pressure.

Table 5-1 CGS Cycle 20 Sensitivity of Overpressure Analysis Results to Initial Flow

Initial Power / Flow (%Rated)	Peak Steam Dome Pressure (psig)	Peak Vessel Pressure (psig)
102 / 106	1305	1341
102 / 80.7	1296	1321

6.0 THERMAL-HYDRAULIC STABILITY

6.1 Introduction

The stability compliance of GNF fuel designs with NRC regulatory requirements is documented in Section 9 of Reference 2. NRC approval of the stability performance of GE fuel designs also includes operation in the MELLLA region of the P/F map.

The above NRC acceptance of thermal-hydraulic stability includes the condition that the plant has systems and procedures in place, supported by Technical Specifications, as appropriate, which provide adequate instability protection. CGS has licensed Option III (Reference 11) as the stability long-term solution and has an approved Technical Specification for the Option III hardware. The Option III hardware has been installed and connected to the Reactor Protection System (RPS). In the event that the Oscillation Power Range Monitor (OPRM) system is declared inoperable, CGS will operate under alternate methods.

The Option III detect and suppress stability solution has been implemented at CGS. The demonstration calculations that are included in Sections 6.2 and 6.3 are based on the current Cycle 20 core design at the increased MELLLA P/F map upper boundary. When the MELLLA upper boundary domain is implemented, cycle specific setpoints will be determined in accordance with Reference 2 and documented in the Supplemental Reload Licensing Report (SRLR).

6.2 Stability Option III

The Option III solution combines closely spaced LPRM detectors into "cells" to effectively detect either core-wide or regional (local) modes of reactor instability. These cells are termed OPRM cells and are configured to provide local area coverage with multiple channels. Plants implementing Option III have hardware to combine the LPRM signals and to evaluate the cell signals with instability detection algorithms. The Period Based Detection Algorithm (PBDA) is the only algorithm credited in the Option III licensing basis. Two defense-in-depth algorithms, referred to as the Amplitude Based Algorithm (ABA) and the Growth Rate Algorithm (GRA), offer a high degree of assurance that fuel failure will not occur as a consequence of stability related oscillations.

The Option III Trip Enabled Region has been generically defined as the region (less than or equal to 60% rated core flow and greater than or equal to 30% rated power) where the OPRM system is fully armed. The Backup Stability Protection (BSP) evaluation described in Section 6.3 shows that the generic Option III Trip Enabled Region should be expanded for operation in the MELLLA region. The BSP analysis recommends extending the power boundary of the generic Option III OPRM Trip-Enabled Region to greater than or equal to 25% rated CLTP and keeping the flow boundary at less than or equal to 60% rated core flow. The OPRM Trip-Enabled Region is shown in Figure 6-1.

The minimum power at which the OPRM should be confirmed operable is 20% rated CLTP. A 5% absolute power separation between the OPRM Trip-Enabled Region power boundary and the power at which the OPRM system should be confirmed operable, is deemed adequate for the Option III solution.

Stability Option III provides SLMCPR protection by generating a reactor scram if a reactor instability, which exceeds the specified trip setpoint, is detected. The demonstration setpoint for the Cycle 20 core design at the increased MELLLA P/F map upper boundary is determined per the NRC approved methodology (Reference 12). The Option III stability reload licensing basis calculates the limiting OLMCPR required to protect the SLMCPR for both steady-state and transient stability events as specified in the Option III methodology. These OLMCPRs are calculated for a range of OPRM setpoints for MELLLA operation. Selection of an appropriate instrument setpoint is then based upon the OLMCPR required to provide adequate SLMCPR protection. This determination relies on the DIVOM curve (Delta CPR Over Initial MCPR Versus Oscillation Magnitude) to determine an OPRM setpoint that protects the SLMCPR during an anticipated instability event. The DIVOM slope was developed based on a TRACG evaluation in accordance with the BWR Owner's Group (BWROG) Regional DIVOM Guideline (Reference 13). The analysis is performed with the Cycle 20 nominal core simulator wrap-ups at limiting conditions.

Hot Channel Oscillation Magnitude (HCOM) analyses was performed in Reference 14, with a corner frequency (CF) of 1.0Hz. The analysis with a HCOM CF of 1.0Hz is shown in Table 6-1. Assuming an estimated OLMCPR of 1.33 and an estimated SLMCPR of 1.09, an OPRM Amplitude Setpoint of 1.11 is the highest setpoint that may be used without stability setting the OLMCPR, according to the results in Table 6-1. The OPRM Amplitude Setpoint of 1.11 requires an OPRM Successive Confirmation Count Setpoint of 14 or less. The actual setpoint will be established on a cycle specific basis.

Therefore, ARTS/MELLLA operation is justified for plant operation with stability Option III.

6.3 Backup Stability Protection

CGS implements the associated BSP regions (Reference 15) as the stability-licensing basis if the Option III OPRM system is declared inoperable.

The BSP regions consist of two regions (I-Scram and II-Controlled Entry). The Base BSP Scram Region and Base BSP Controlled Entry Region are defined by state points on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL) in accordance with Reference 15. The bounding plant-specific BSP region state points must enclose the corresponding Base BSP region state points on the HFCL and on the NCL. If a calculated BSP region state point is located inside the corresponding Base BSP region state point, then it must be replaced by the corresponding Base BSP region state point. If a calculated BSP region state point is located outside the corresponding Base BSP region state point, this point must be used.

That is, the selected points will result in the largest, or most conservative, region sizes. The proposed BSP Scram and Controlled Entry region boundaries are constructed by connecting the corresponding bounding state points on the HFCL and the NCL using the Modified Shape Function (MSF) as defined in Reference 16.

The demonstration BSP regions for both Nominal Feedwater Temperature (NFWT) and Reduced Feedwater Temperature (RFWT) were expanded in the MELLLA region in accordance with the guidance in Reference 15. The demonstration of the proposed BSP regions, based on Cycle 20, is shown in Table 6-2 for NFWT and Table 6-3 for RFWT. Plots of the BSP regions on the P/F map are show in Figure 6-2 for NFWT and Figure 6-3 for RFWT. The BSP regions, as described in Reference 15, are confirmed or expanded on a cycle-specific basis.

Therefore, ARTS/MELLLA operation is justified for plant operation with stability BSPs.

Table 6-1 Option III Setpoint Demonstration with HCOM CF of 1.0 Hz

OPRM Amplitude Setpoint	Δ_i*	OLMCPR(SS) MELLLA	OLMCPR(2RPT) MELLLA
1.05	0.166	1.212	1.177
1.06	0.197	1.240	1.204
1.07	0.229	1.268	1.231
1.08	0.260	1.297	1.260
1.09	0.292	1.329	1.290
1.10	0.323	1.351	1.311
1.11	0.353	1.361	1.322
1.12	0.383	1.372	1.332
1.13	0.413	1.383	1.343
1.14	0.443	1.394	1.353
1.15	0.473	1.405	1.364
1.16	0.501	1.416	1.375
1.17	0.530	1.431	1.389
1.18	0.558	1.456	1.413
1.19	0.587	1.481	1.438
1.20	0.615	1.507	1.463
		Off-rated OLMCPR @45% core flow	**Rated Power OLMCPR**

*Δ_i represents the Hot Channel Oscillation Magnitudes (Reference 14).

Table 6-2 BSP Region Endpoints for NFWT

Case Name	Region Boundary	Power (% Rated)	Flow (% Rated Core Flow)
A1 – Base	HFCL, Scram Region	64.7	40.0
B1	NCL, Scram Region	33.8	23.8
A2 – Base	HFCL, Controlled Entry Region	73.8	50.0
B2	NCL, Controlled Entry Region	25.1	23.8

Table 6-3 BSP Region Endpoints for RFWT

Case Name	Region Boundary	Power (% Rated)	Flow (% Rated Core Flow)
A1	HFCL, Scram Region	67.9	43.5
B1	NCL, Scram Region	28.5	23.7
A2 – Base	HFCL, Controlled Entry Region	73.8	50.0
B2	NCL, Controlled Entry Region	24.5	23.4

Figure 6-1 MELLLA OPRM Trip Enabled Region

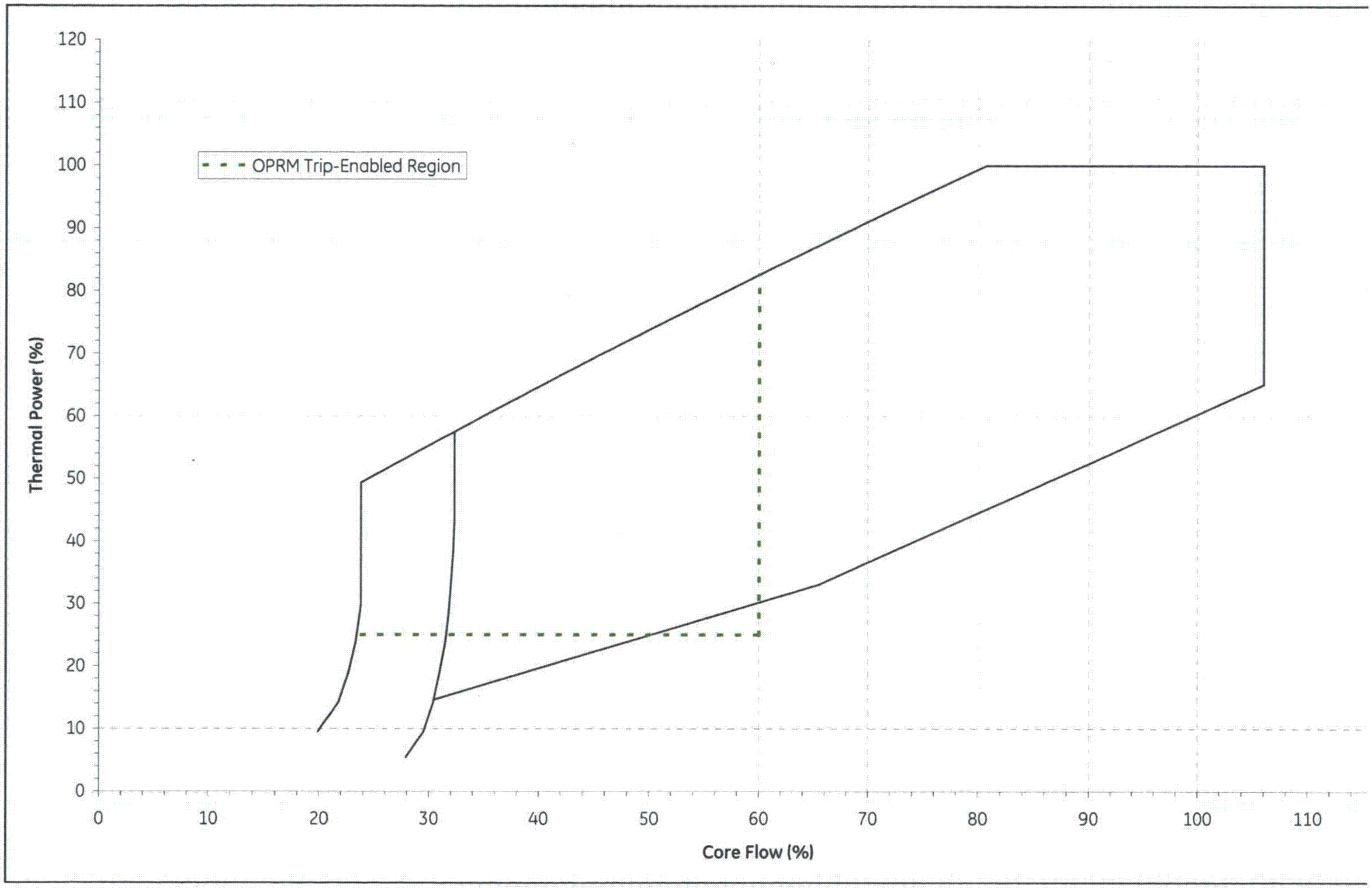


Figure 6-2 Demonstration of Proposed BSP Regions for NFWT

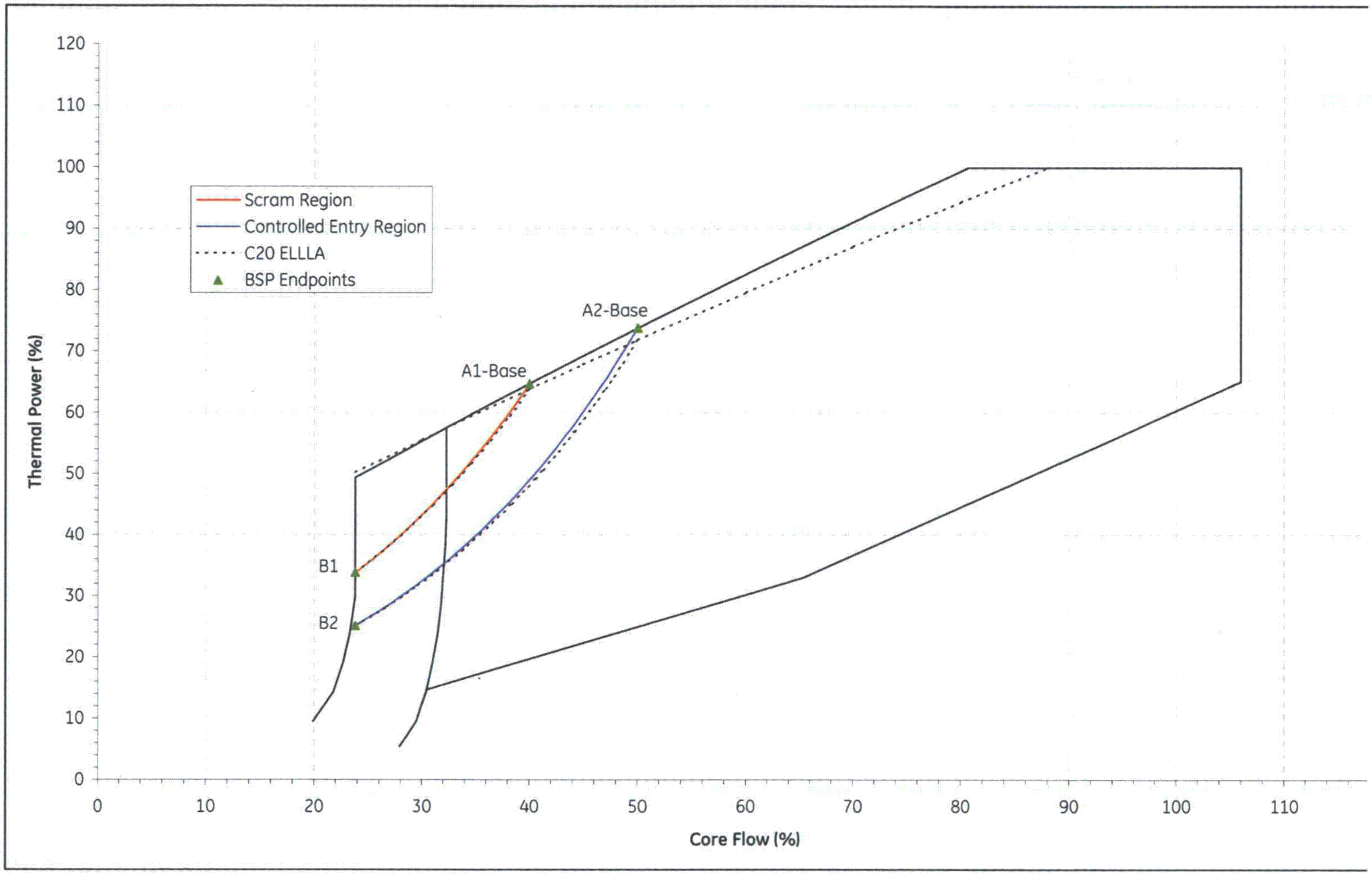
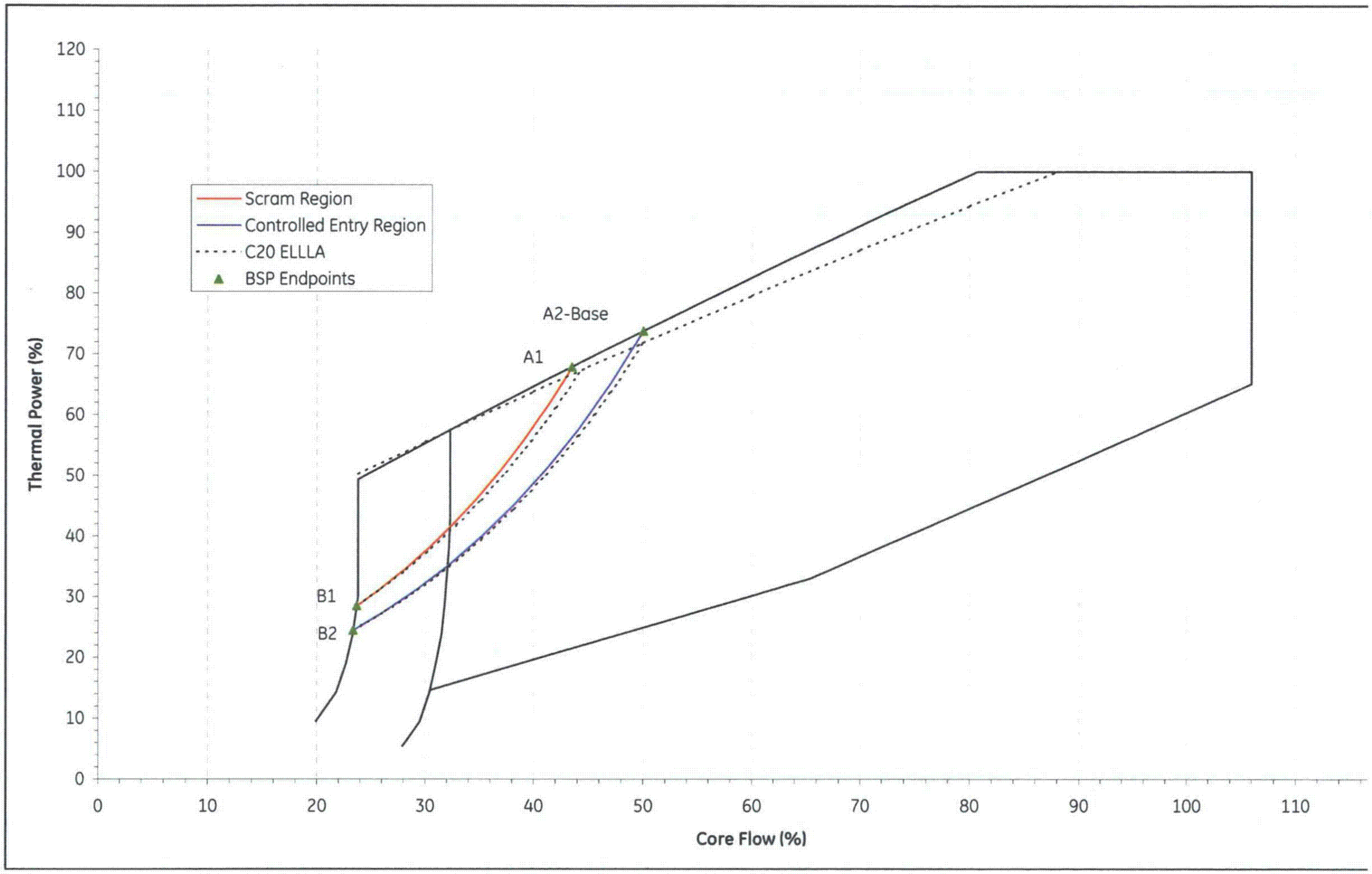


Figure 6-3 Demonstration of Proposed BSP Regions for RFWT



7.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

The current licensing basis SAFER/GESTR-LOCA analysis for CGS (Reference 17 for the base SAFER/GESTR analysis and Reference 18 for GE14 fuel) has been reviewed to determine the effect on the Emergency Core Cooling System (ECCS) performance resulting from CGS operation in the MELLLA domain. The Reference 18 analyses considered CGS operation in the ELLLA domain. The LOCA analysis for CGS operation in the MELLLA domain are in conformance with the error reporting requirements of 10 CFR 50.46 through notification number 2008-01. Therefore, all known ECCS-LOCA analysis errors in accordance with 10 CFR 50.46 have been accounted for in the analysis in support of the application of ARTS/MELLLA for CGS. The CGS current licensing basis PCT for GE14 fuel is shown in Table 7-2. This current licensing basis PCT is 1710°F and is set by the results of the LOCA analysis for the recirculation suction line break (RSLB) at 104.1% CLTP/RCF with a top-peaked axial power distribution (Reference 18).

The two major parameters that affect the fuel peak cladding temperature in the design basis LOCA calculation, which are sensitive to the higher load line in the operating P/F map, are the time of boiling transition (BT) at the high power node of the limiting fuel assembly and the core recovery time. Initiation of the postulated LOCA at lower core flow may result in earlier BT at the high power node, compared to the 100% of RCF results, resulting in a higher calculated PCT. Similarly, initiation of the postulated LOCA at lower core flow affects break flow rate and core reflooding time, compared to the 100% of RCF results, which can also result in a higher calculated PCT. The effect on the calculated PCT is acceptable as long as the results remain less than the Licensing Basis PCT limits.

The ARTS-related changes will not affect the LOCA analysis. The current CGS licensing basis specifies a requirement in maximum LHGR as a function of drive flow, known as the APRM set down requirement. With the implementation of ARTS, this lower LHGR requirement is being replaced with direct core power and flow fuel thermal limits by the ARTS improvement option. If the direct core power and flow fuel thermal limits were modeled in the LOCA analysis, a reduction of PCT would result, leaving the reported cases as limiting. Acknowledging this credit, these reduced thermal limits are not modeled in the LOCA analysis, and the LOCA analysis is not required for the implementation of ARTS.

The nominal and Appendix K PCT response following a large recirculation line break for most plants show that the PCT effect due to MELLLA is small. In some cases, there may be a significant PCT increase if early boiling transition penetrates down to the highest-powered axial node in the fuel bundle. This can happen at core flows in the MELLLA region. [[

]] For small breaks, the fuel remains in nucleate boiling until uncovery and MELLLA is expected to have no adverse effect on the small break LOCA response.

Calculations assuming the MELLLA extended operation domain were performed to quantify the effect on PCT to the allowed operation envelope. The MELLLA assumptions for the limiting large recirculation line break case resulted in an [[

]]

MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46. Because cladding oxidation is primarily determined by PCT, MELLLA can affect the amount of cladding oxidation in those cases where there is a significant PCT increase. Jet pump BWRs have significant margin to the local cladding oxidation and core-wide metal-water reaction acceptance criteria, even for PCTs at the 2200°F limit. The compliance with the 2200°F limit ensures compliance with the local cladding oxidation and core-wide metal-water reaction acceptance criteria for GE14 fuel. Compliance with the coolable geometry and long-term cooling acceptance criteria were demonstrated generically for GE BWRs (Reference 19). MELLLA does not affect the basis for these generic dispositions. Therefore, MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46.

The CGS MELLLA evaluation is based on plant-specific calculations with GE14 fuel using SAFER/GESTR methodology (References 19 through 24). Calculations were performed for rated flow and power conditions in the last ECCS-LOCA analysis using the SAFER/GESTR methodology (Reference 18). Bases from the reference analysis were retained. Specifically:

- Recirculation suction leg break location is the limiting break location, and remains the break location considered in the MELLLA analysis.
- The limiting single failure identified in the previous LOCA analysis (i.e., High Pressure Core Spray Diesel Generator (HPCSDG)) has not changed.
- [[

  ]]
- A full core of GE14 fuel is assumed to comprise the core.
- The Upper Bound PCT has been addressed in the current analysis (Reference 18). The Upper Bound PCT has been shown bounded by the Licensing Basis PCT. The Upper Bound PCT does not need to be recalculated for ARTS/MELLLA implementation.
- ECCS operation parameters are consistent with those used in the Reference 18 analysis.
- The bottom head drain line is included in analysis of the small break as the evaluation model is applied. The small break area includes the full guillotine bottom head break area plus additional recirculation suction line area to obtain the total break area represented. With this procedure, the consequences of the double-ended guillotine rupture of the bottom vessel head

drain line is always covered by the small break spectrum, including consideration of single failure and break location.

A summary of analysis inputs is presented in Table 7-1. Results from these calculations are presented in Table 7-2.

7.1 Conclusions

The calculations for CGS show that the MELLLA option will meet the PCT acceptance criteria for a representative core with GE14 fuel and has no effect on any other LOCA criteria. Therefore, no additional restrictions on fuel power to account for LOCA criteria compliance are required. Calculations at the 104.1% CLTP/MELLLA flow condition result in the highest PCT for the large break LOCA. Calculations at the 104.1% CLTP / rated flow condition result in the highest PCT for the small break LOCA and set the licensing basis PCT for CGS.

Table 7-1 ECCS-LOCA Analysis Bases for CGS ARTS/MELLLA

Parameter	Units	Value
Original Licensed Thermal Power	MWt	3323
Current Licensed Thermal Power	MWt	3486
ECCS-LOCA Rated Thermal Power	MWt	3629
Vessel Steam Dome Pressure	psia	1055
Rated Core Flow	Mlb/hr	108.5
MELLLA Core Flow (85.75% rated flow at 104.1% CLTP)	Mlb/hr	93.04

Table 7-2 ECCS-LOCA Peak Cladding Temperature for CGS ARTS/MELLLA

Case Description	PCT (°F) Current Analysis (Reference 18)	PCT (°F) (ARTS/MELLLA)
DBA Break:		
Appendix K Assumptions		
[[		]]
Nominal Assumptions		
[[		]]
Small Break:		
Appendix K Assumptions		
[[	]]Note 1	Not Analyzed
Nominal Assumptions		
[[	]]	Not Analyzed
Licensing Basis PCT:	1710	Not Analyzed Note 2

Appendix K – 10.CFR50.46 Appendix K assumptions

ADS – Automatic Depressurization System

Note 1 - Case Description (break size, axial power shape, limiting single failure) that sets the Licensing Basis PCT. The Licensing Basis PCT is based on the Upper Bound PCT for this case description.

Note 2 – Licensing basis PCT is set by the Rated Flow condition

8.0 CONTAINMENT RESPONSE

8.1 Approach/Methodology

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

For this evaluation, a qualitative evaluation is performed which uses the results of previous short-term DBA-LOCA analyses performed for the CGS Power Uprate/ELLLA (Reference 1) and also the results of similar analyses performed for other BWR plants with Mark II containments.

Previously, the effect of MELLLA operation on the Mark II containment response for the DBA-LOCA RSLB and on the associated Mark II containment DBA-LOCA hydrodynamic loads has been evaluated with plant-specific containment analyses with the M3CPT containment analysis code (References 25, 26) using mass and energy release rates obtained with the detailed LAMB blowdown model (Reference 19). The purpose of these analyses has been to quantify the effect of changes in break subcooling on the mass and energy release rates and consequently on the containment response. Similarly, the effect of other off-rated conditions such as ELLLA, ICF, SLO, or operation with RFWT have also been evaluated with plant-specific containment analyses. This process was applied for CGS to evaluate the different reactor conditions associated with the current license thermal power in support of the CGS Power Uprate/ELLLA (Reference 1).

A review of the results of the CGS plant-specific containment analyses, indicate that changes in reactor conditions associated with MELLLA operation have a small effect on the containment response. A review of similar analyses performed for other plants with Mark II containment have shown similar results.

A qualitative evaluation approach to the short-term DBA-LOCA containment evaluation was applied for the CGS MELLLA containment assessment. In this approach the results of the Reference 1 M3CPT/LAMB DBA-LOCA analyses are used to establish trends with respect to the effect of reactor conditions. With these trends determined, the effect of MELLLA operation can be assessed. The results obtained from analyses performed for other plants with Mark II containments are also reviewed in support of this evaluation. The results of this trend evaluation are used as a basis to assess the impact of ARTS/MELLLA operation on the CGS license basis containment analyses and on the CGS LOCA hydrodynamic load definition.

8.1.1 Short-Term Pressure/Temperature Response

The short-term containment response covers the blowdown period during which the maximum drywell pressure and temperature, wetwell pressure, and maximum drywell to wetwell differential pressure occur. Consequently, analyses were performed for various cases that cover the full extent of CGS operation including the ELLLA and ICF domain in support of the Power Uprate/ELLLA (Reference 1). The objective of performing these analyses was to demonstrate that the containment pressure and temperature design limits, as stated in the CGS FSAR, are not exceeded. The results of these analyses were also used to evaluate the various containment hydrodynamic loads.

For this qualitative evaluation, the results of DBA-LOCA short-term analyses performed in support of the CGS Power Uprate/ELLLA (Reference 1) are reviewed to establish trends. Additionally, the results of similar analyses for other BWR plants with Mark II containments are also reviewed. The purpose of this review was to establish a general trend of containment response characteristics with differences in reactor conditions which control the initial break flow subcooling and can affect the reactor blowdown response.

For this evaluation, the results of analyses performed, in support of Reference 1, at the following reactor conditions were considered, including cases with NFWT and cases with a 65°F RFWT.

1. 106.2% of RTP / 100.0% of core flow (NFWT & RFWT)
2. 106.2% of RTP / 106.0% of core flow (ICF, NFWT)
3. 106.2% of RTP / 94% of core flow (ELLLA, NFWT & RFWT)
4. 74.7% of RTP /56.4% of core flow (SLO, NFWT)
5. 59.2% of RTP / 36.4% of core flow (Minimum Recirculation Pump Speed, RFWT)
6. 64.1% of RTP/ 38.6% of core flow (Minimum Recirculation Pump Speed @ ELLLA, NFWT)

These cases were selected for the Power Uprate/ELLLA containment evaluation to conservatively cover the full extent of the current licensed P/F boundary including the ELLLA and ICF regions.

8.1.2 LOCA Containment Hydrodynamic Loads

The CGS LOCA containment hydrodynamic loads assessment includes PS, CO and CH loads. These loads are evaluated based on the short-term containment response analysis.

Plant operation in the ARTS/MELLLA region changes the mass flux and the subcooling of the break flow, which may affect the containment short-term LOCA response and subsequently the containment hydrodynamic loads. These loads were generically defined for Mark II plants during the Mark II Containment Program as described in Reference 27 and accepted by the NRC in References 28 and 29. The plant-specific dynamic loads are also defined in the CGS Design Assessment Report (DAR) (Reference 30). The current evaluation of these loads for CGS is described in the Safety Analysis Report for Power Uprate/ELLLA (Reference 1).

The containment hydrodynamic loads evaluation presented in this section also include considerations of the currently licensed 20°F Feedwater Heater Out-of-Service (FWHOOS) and future applications for 65°F Final Feedwater Temperature Reduction (FFWTR) and FWHOOS.

8.2 Assumptions and Initial Conditions

The CGS MELLLA containment evaluation relies on the results of the containment analyses performed for Reference 1; therefore, it is assumed that there are no significant differences in initial conditions or plant configuration parameters that potentially affect the containment response, relative to inputs, used for the Reference 1 analyses. This assumption was confirmed as part of the MELLLA containment evaluation.

The following initial containment conditions were used in the Reference 1 DBA-LOCA short-term containment pressure/temperature response analysis.

Parameter	Value
Drywell Pressure (psig)	0.7
Wetwell Pressure (psig)	0.7
Drywell Temperature (°F)	135
Suppression Pool Temperature (°F)	90
Drywell humidity (%)	50%
Wetwell humidity (%)	100%

The initial conditions shown above are common to all cases performed for Reference 1. An additional NFWT case with an initial drywell and wetwell pressure of 2.0 psig was performed at 106.2% of RTP / 100.0% of rated core flow.

The key assumptions used in the Reference 1 analyses of the short-term containment response for CGS operation in the Power Uprate/ELLLA domain are listed below.

1. Reactor power generation is assumed to cease concurrently with the time of the accident initiation. There is no delay period.

2. The break being analyzed is an instantaneous double-ended rupture of a recirculation suction line. This results in the maximum discharge rates to the drywell.
3. GE's LAMB computer code (Reference 19) is used to calculate the break flow rates and break enthalpies. These values are then used as inputs to the M3CPT computer code (References 25 and 26) to calculate the containment pressure and temperature response.
4. The vessel blowdown flow rates are based on the Moody Slip flow model. (Reference 31)
5. The Main Steam Isolation Valves (MSIVs) start closing at 0.50 seconds (the delay is associated with the maximum instrument signal response) after initiation of the accident. They are fully closed in the shortest possible time of 3.50 seconds after initiation of the accident.
6. No credit is taken for passive structural heat sinks in the containment. Steam condensation on structures and components in the containment is therefore conservatively neglected.
7. The wetwell airspace is in thermal equilibrium with the suppression pool at all times.
8. The flow of liquid, steam, and air in the vent system is assumed to be a homogenous mixture based on the instantaneous mass fractions in the drywell.
9. The feedwater flow is assumed to begin to coast down at 3.9 seconds and entirely stop at 43.5 seconds.

8.3 Analyses Results

8.3.1 Short-Term Pressure/Temperature Response

Table 8-1 provides a description of the reactor conditions associated with the cases performed in support of the Power Uprate/ELLLA (Reference 1). Table 8-2 provides the conditions associated with MELLLA used for this evaluation. A review of the reactor conditions shown in Tables 8-1 and 8-2 show that reactor conditions with MELLLA are effectively enveloped by the conditions already analyzed in support of Reference 1, based on a comparison of initial break subcooling. For this evaluation the subcooling is defined as the difference between the initial break enthalpy and the liquid enthalpy corresponding to the initial reactor dome pressure. The subcooling associated for the minimum pump speed condition with MELLLA in Table 8-2 is slightly higher than previously considered for the supporting analyses for Reference 1; however, as is identified in the following paragraphs, the effects of higher subcooled conditions are relatively small and typically produce a reduced containment response.

Table 8-3 summarizes key results from analyses performed in support of Power Uprate/ELLLA (Reference 1) and reviewed for the MELLLA evaluation.

The key parameter for the DBA-LOCA short-term pressure/temperature analysis is the peak drywell pressure, which is shown in Table 8-3. For the DBA-LOCA RSLB events, near saturation conditions exist in the drywell at the time of peak drywell pressure, so the peak drywell pressure establishes the peak drywell temperature, with higher peak drywell temperatures occurring with higher peak drywell pressures. The results presented in Table 8-3 indicate that with the exception of Case 5 in Table 8-3 (Minimum Pump Speed (MPS), with RFWT), higher values for peak drywell pressure occur for full power conditions, with core flow and feedwater temperature having a relatively small effect on peak drywell pressure. This trend is consistent with the trends observed in similar analyses performed for other Mark II plants. It was determined that the high drywell pressure obtained for Case 5 was caused by a more conservative application of the LAMB break flow enthalpy history for Case 5 than for the other cases. This produced an artificially high peak drywell pressure for Case 5 relative to the other cases. More recent calculations performed for other Mark II plants, use a newer, automated process, which uses all LAMB break flow data. The newer analyses show more definitive trends in the peak drywell pressures with maximum peak drywell pressures occurring with minimum initial subcooling, such as occurs with full reactor power, rated or ICF, and with NFWT.

Based on trends observed in the CGS Power Uprate/ELLLA analyses, and analyses performed for other plants with Mark II containments, it was concluded that operation with MELLLA will not adversely affect the DBA-LOCA short-term containment response, relative to the response previously evaluated for Reference 1, and will not result in the exceeding of containment pressure and temperature design limits.

The CLTP peak drywell pressure remains bounding for MELLLA.

8.3.2 LOCA Containment Hydrodynamic Loads

Three types of hydrodynamic loads are addressed for the DBA-LOCA: a) PS loads, b) CO loads, and c) CH loads. The effect of ARTS/MELLLA on these loads is evaluated based on a review of the containment responses obtained for the Reference 1 Power Uprate/ELLLA analyses and trends determined from this review.

8.3.2.1 Pool Swell

The PS loads include the vent clearing loads, the LOCA bubble wall pressure and submerged structure loads, wetwell airspace pressurization and the PS effect and drag loads. All of these loads are controlled by the initial drywell pressurization (first 2 seconds) following the initiation of the DBA-LOCA.

A measure of the initial drywell pressurization rate is provided by the peak drywell-to-wetwell pressure difference. This parameter occurs during the initial 2 seconds of the event, which is coincident with the pool swell period. A review of Table 8-3 shows that the maximum value for this parameter occurs for Case 3. Case 3 had the smallest associated reactor subcooling. This trend is similar to results seen in analyses performed for other Mark II plants.

As part of the Power Uprate/ELLLA pool swell loads evaluation for Reference 1, the drywell pressure history for this case was compared to the drywell pressure history used to define the PS load, and the comparison confirmed that the load definition drywell pressure history remains bounding. Additionally, confirmatory calculations were performed using the GEH PS model (Reference 32), which confirmed that the pool swell response, used to define the PS load, remains bounding.

Because the containment response conditions controlling the PS load were shown to be bounding with lower subcooling, the slight increase in reactor subcooling associated with MELLLA will not produce a more severe drywell pressurization than already assessed for Power Uprate/ELLLA, and that response will be bounded by the containment response used to define the CGS PS load in Reference 30.

8.3.2.2 Condensation Oscillation

CO loads result from oscillation of the steam-water interface that forms at the vent exit during the region of high vent steam mass flow rate. This occurs after PS and ends when the steam mass flux is reduced below a threshold value. CO loads increase with higher steam mass flux and higher suppression pool temperature. The generic Mark II CO definition is based on Mark II 4TCO tests (Reference 33). The 4TCO tests were designed to simulate LOCA containment thermal-hydraulic conditions (i.e., steam mass flux and pool temperature), which bound all Mark II plants including CGS.

According to the description given in Section 3.2.4.1.2 of the CGS DAR (Reference 30), the CO load for the CGS plant was eliminated based on a review of the JAERI multivent CO test results. Based on this test data when multiple vent effects were considered, the CO load is significantly reduced relative to the CO load from the single vent tests. Per Reference 30, with multiple vent effects considered, the CGS CH load definition provides a bounding load for both CO and CH. A review of the short-term DBA-LOCA responses for Cases 1 through 8 of Table 8-1, performed in support of the Power Uprate/ELLLA, determined that differences in the DBA-LOCA vent flow and suppression pool temperature response introduced by the reviewed differences in reactor conditions are small. Thus, it was concluded that MELLLA would also not adversely affect CO loads, and that there is no effect of MELLLA on the existing basis in Reference 30 for elimination of the CO load for CGS.

8.3.2.3 Chugging

The CH load definition for CGS is an alternative load to the Mark II generic CH load (Reference 27), but uses the same CH test data from the Mark II 4TCO tests (Reference 33). The 4TCO tests covered the full range of thermal-hydraulic conditions with CH expected for Mark II containment geometry. Because the thermal-hydraulic conditions for the Reference 33 tests (i.e., steam mass flux, air content and suppression pool temperature) were selected to produce the maximum CH amplitudes for a Mark II containment, any changes to the containment response due to MELLLA will not affect the CH load definition.

8.4 Conclusions

It is concluded that ARTS/MELLLA has no adverse effect on the current CGS definition of the dynamic loads of (1) PS, (2) CO and (3) CH and that the existing definitions of LOCA dynamic loads of PS, CO, and CH for CGS remain applicable for ARTS/MELLLA.

8.5 Reactor Asymmetric Loads

In support of MELLLA implementation, the effect of expanding the reactor operating domain from the current ELLLA P/F map boundary to the MELLLA P/F map boundary on HELB mass and energy releases to the annulus region between the RPV and the sacrificial shield wall (FSAR Figure 6.2.23) were evaluated. The change in mass and energy release from the break may affect the asymmetrical loads acting on the primary and containment SSCs important to safety (e.g., RPV, reactor internals, shield wall, and piping). The drywell head sub compartment pressurization was also evaluated for the effect of MELLLA on the differential pressure loading across the drywell head bulkhead plate. These evaluations were performed over the range of P/F conditions associated with the MELLLA boundary.

8.5.1 Annulus Pressurization Analysis

The reactor asymmetric loads during the DBA LOCA include the annulus pressurization (AP) loads, the jet reaction loads / jet impingement loads, and the pipe whip loads.

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

- Recirculation Suction Line Break
- Feedwater Line Break (FWLB)

The methodology for calculating the current RSLB blowdown mass and energy release profile for AP loads is the conservative methodology documented in NEDO-24548 (Reference 34). A more realistic blowdown mass and energy release profile was determined for the MELLLA AP loads analysis using the GEH code LAMB. The LAMB code has been used in the plant licensing application to calculate the blowdown mass flow rate and energy profile for AP loads in the event of a RSLB and has been accepted for LOCA evaluations in support of licensing applications for P/F map extensions such as MELLLA. The LAMB mass and energy release analysis considers the pipe break separation time history and ignores the fluid inertia effect.

The methodology used for calculating the current RSLB sub compartment pressurization transients for AP loads were the RELAP model combined with the GEH analytical method for determining mass and energy release referenced in Section 6.2.1.2 of the Columbia FSAR. For the MELLLA evaluation, the pressurization transients for the AP load analysis were determined using the GOTHIC code (Reference 35). The use of the GOTHIC code allowed for a much finer nodalization of the annulus region (approximately 400 nodes versus 30 nodes in the current RELAP analysis). The GOTHIC code also provides a more realistic treatment of the loss coefficients and momentum flux in the annulus region. The pressure multiplier factor of 1.4 specified in NUREG-0800 Section 6.2.1.2-5 that had been used in the construction permit was also eliminated by CGS.

The methodology used for calculating the current FWLB blowdown mass and energy was based on RELAP. The FWLB mass flow may increase slightly due to the increased vessel liquid subcooling associated with MELLLA. However, the effect on the critical break flow rate on the energy flux into the annulus is more than offset by the effects of reduced break flow enthalpy. Therefore, MELLLA operation is expected to have a negligible effect on the annulus pressurization loads and structural response for the FWLB.

8.5.2 Impact on Structural Response

Evaluations were performed to determine the effect of the AP load methodology change and MELLLA operation on the dynamic structural response of the RPV, reactor internals, piping and containment structures. These evaluations used the same mathematical lumped mass beam model as the original analyses of record.

Effect of Methodology Change

The results from the updated dynamic analyses using the more realistic LAMB/GOTHIC methodology were compared against those used as input to the component structural analyses of record based on the current NEDO/RELAP methodology. The change to the more realistic LAMB/GOTHIC methodology generally resulted in a reduction in the structural response. Most components saw a reduction in loads on the order of 6%-100%. However, significant increases in loads were observed for some components: fuel (101%), shroud and shroud support (38%), shroud head (48%), and steam separator (22%). A small increase (less than 4%) was also observed in the primary containment loads.

The amplified response spectra (ARS) envelopes were also compared to determine if the change in methodology resulted in any significant shifts in frequency content (up to the original design basis frequency of 60 Hz). The envelopes based on the more realistic LAMB/GOTHIC methodology are in general, bounded by the original design basis envelopes in frequency range from 10 Hz to 60 Hz. The envelope spectra show new peaks in the frequencies below 10 Hz at a few locations on the Shroud, Steam Separator, RPV, BSW, BOP, and Primary Containment. The effects of the increases in loads and changes in frequency content are dispositioned in Section 8.5.3.

Effect of ARTS/MELLLA

With the more realistic modeling, the evaluation results show that MELLLA operation has only a minor effect on the structural response due to a RSLB between the RPV and the sacrificial shield wall. The largest increase in structural response associated with MELLLA implementation was less than 3% compared to the current operating conditions, with the results for most components showing little or no change. MELLLA operation had no notable effect on the frequency content of the amplified response spectra envelopes. The increases in loads are dispositioned in Section 8.5.3.

8.5.3 Evaluation of Structural Response

The results of the structural response evaluation in Section 8.5.2 shows that MELLLA operation resulted in only a minor effect on the structural responses. The change to a more realistic AP Load methodology resulted in a reduction in the loads for most components, however some components saw a significant increase in the loads or additional frequency content in the ARS envelopes. The affected components and systems were evaluated to confirm that these SSCs could accommodate the change in the AP loads. The AP loads are combined with the safe shutdown earthquake (SSE) seismic loads in the faulted load combination using the square root of the sum of the squares (SRSS). The SSE loads in the load combination are not affected by MELLLA. Because the SSE loads tend to be the dominant term in the load combination, the SRSS process diminishes the AP loads contribution to the total component stresses.

8.5.3.1 RPV Integrity Components

Analyses are performed for the design, the normal and upset, and the emergency and faulted conditions. If there is an increase in annulus pressurization, jet reaction, pipe restraint or fuel lift loads, the changes are considered in an analysis of the components affected by the annulus pressurization associated loading increase.

Faulted Conditions

Only annulus pressurization related faulted loads for the RPV Shroud Support component increase for ARTS/MELLLA conditions relative to the existing design basis. All other RPV component faulted loads remain bounded by the existing design basis for ARTS/MELLLA conditions. The Shroud Support is evaluated for the increases in annulus pressurization associated loads, and the design basis, bounding stresses of this component are found to remain unaffected. Therefore, ASME Code, Section III, and Sub-section Nuclear Boiler (NB) requirements are met for all RPV components for annulus pressurization associated faulted conditions.

8.5.3.2 Reactor Internals

The Reactor Internals are qualified in Section 9.3 for all applicable MELLLA-based loads.

8.5.3.3 Reactor Coolant Pressure Boundary Piping Evaluation (Inside Containment)

As noted in Section 8.5.2, the ARS envelopes are in general, bounded by the original design basis envelopes in the frequency range from 10 Hz to 60 Hz. However, the envelope spectra showed additional frequency content below 10 Hz at locations that could affect the reactor coolant pressure boundary piping (RCPB). The RCPB piping, piping supports, and restraints were evaluated to confirm that these components could accommodate the change in the AP loads. The results of those evaluations showed that there was sufficient margin to accommodate the change in AP loads and that the stresses on the piping, supports, and restraints will continue to meet the applicable ASME Code allowables.

8.5.4 Drywell Head Region

The drywell head subcompartment pressurization was evaluated for the effect of MELLLA for the following breaks:

- RSLB
- RCIC Head Spray Line Break

A RSLB in the lower drywell region produce an upward loading on the bulkhead plate in the drywell head region. The pressure loads for this event are predominantly controlled by the break energy flux, which is not affected by extension of operating domain to MELLLA. The CLTP peak drywell pressure remains bounding for MELLLA (Section 8.3.1).

A break of RCIC head spray line (steam break) in the upper drywell head region causes the downward loads on the bulkhead plate. The break flow for steam line breaks is mainly controlled by RPV pressure at rated condition. MELLLA operation does not increase the RPV pressure. Therefore, there is no effect of MELLLA on the bulkhead plate loading this break.

Table 8-1 Cases Analyzed For Short-Term Containment Response

Case No.	Point[1]	Power (MWt)	Core Flow (Mlbm/hr)	Core Inlet Enthalpy (Btu/lbm)	Dome Pressure (psia)	Initial Break Subcooling (Btu/lbm)
1	106.2%P/100%F (Rated)	3702	108.5	530.8	1055.0	20.1
2	106.2%P/100%F (65°F FFWTR)	3702	108.5	522.8	1055.0	28.1
3	106.2%P/106%F (ICF)	3702	115.0	532.1	1055.0	18.8
4	74.7.%P/56.4%F (SLO)	2604	61.2	516.4	1032.0	31.0
5	59.2%P/36.4%F (Min Pump Speed -65°F FFWTR)	2064	39.5	494.8	1017.0	50.3
6	106.2%P/94%F (ELLLA)	3702	102.0	529.4	1055.0	21.5
7	106.2%P/94%F (ELLLA 65°F FFWTR)	3702	102.0	518.0	1048.0	31.8
8	64.1%P/38.6%F (Min Pump Speed)	2234	41.9	503.3	1023.6	42.9

1 P = 3486 MWt

Table 8-2 Conditions Reviewed for MELLLA

Case No.	Point[1]	Power (MWt)	Core Flow (Mlbm/hr)	Core Inlet Enthalpy (Btu/lbm)	Dome Pressure (psia)	Initial Break Subcooling (Btu/lbm)
1	102%P/80.7%F (MELLLA)	3555.7	87.56	525.4	1050	24.7
2	102%P/80.7%F (MELLLA, RFWT)	3555.7	87.56	514.2	1035	33.7
3	58.65%P/32.3%F (Min Pump Speed	2044.5	35.05	500.3	1035	47.6
4	58.65%P/32.3%F (Min Pump Speed, RFWT)	2044.5	35.05	491.2	1035	56.9

1 P = 3486 MWt

Table 8-3 Summary of Sensitivity Study Results for Peak Drywell Pressure and Temperature and Initial Drywell Pressurization Rate

Case No.	Point[1]	Drywell Pressure (psig)[2]	Drywell-to-Wetwell Differential Pressure (psid)[3]
	Design Limit	45.0	25.0
1	106.2%P/100%F (Rated)	34.8	24.49
2	106.2%P/100%F (65°F FFWTR)	34.6	24.17
3	106.2%P/106%F (ICF)	34.7	24.62
4	74.7%P/56.4%F (SLO)	33.4	23.49
5	59.2%P/36.4%F (Min Pump Speed - 65°F FFWTR)	35.1	21.90
6	106.2%P/94%F (ELLLA)	35.1	24.57
7	106.2%P/94%F (ELLLA 65°F FFWTR)	35.0	23.83
8	64.1%P/38.6%F (Min Pump Speed)	33.2	22.74

1 P = 3486 MWt

2 The values shown in this column are based on an initial wetwell and drywell pressure of 0.7 psig. Case 1 was also performed with an initial drywell and wetwell pressure of 2.0 psig. This case, which is reported in Table 6.2-5 of the CGS FSAR, produced a peak drywell pressure of 37.4 psig.

3 Drywell-to-wetwell differential pressures shown in this table are obtained from the M3CPT output directly, and do not account for wetwell airspace compression effects due to pool swell. The maximum predicted drywell-to-wetwell differential pressure, with the effect of pool swell considered, is 21.70 psid, as shown in Table 4-1 of Reference 1 and is well below the design value of 25 psid. The direct values from M3CPT were selected to quantify trends in the early drywell pressurization history when the peak drywell-to-wetwell pressure occurs.

9.0 REACTOR INTERNALS INTEGRITY

9.1 Reactor Internal Pressure Differences

The reactor internals pressure differences (RIPDs) across the reactor internal components and the fuel channels in the MELLLA condition are bounded by the ICF (106% of RCF) conditions due to the higher core flow condition. Thus, no new RIPDs, fuel bundle lift and Control Rod Guide Tube (CRGT) conditions are generated by the MELLLA operating domain. The current RIPD basis remains applicable to the MELLLA condition.

9.2 Acoustic and Flow-Induced Loads

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

9.2.1 Approach/Methodology

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

The flow-induced loads were calculated for an RSLB utilizing the specific CGS geometry and fluid conditions applied to a reference BWR calculation. The loads were calculated by applying scaling factors that account for plant-specific geometry differences (e.g., size of the shroud, reactor vessel, and recirculation line) and thermal-hydraulic condition differences (e.g., downcomer subcooling) from the reference plant. The reference calculation was based on the GE methods utilized to support NRC Generic Letter 94-03 (Reference 36) that was issued to address the shroud cracks detected at some BWRs.

The acoustic loads on the jet pumps and shroud applied for CGS represent CGS-specific plant geometry configuration and operating conditions. The bounding natural frequencies for the jet pumps and shroud along with the bounding subcooling are applied. For acoustic loads on the shroud support, generic bounding BWR loads based on the GEH approved methods were used. For CGS, the most limiting subcooling condition is at the intersection of the minimum pump speed and the MELLLA boundary line. The initial thermal hydraulic conditions including the subcooling at this point are applied to the reference BWR calculation, along with the CGS geometry, to determine the plant-specific flow-induced loads.

9.2.2 Input Assumptions

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

Initial Conditions	Bases/Justifications
102%P / 100%F	Consistent with the CGS current licensing basis.
102%P / 100% F	Consistent with the CGS current licensing basis with feedwater temperature reduction.
102%P / 80.7%F	MELLLA corner at rated power with feedwater temperature reduction.
58.7%P / 32.3%F	Minimum pump speed (MPS) point on the MELLLA boundary line, with feedwater temperature reduction.
58.7%P / 32.3%F	MPS point on the MELLLA boundary line, with normal feedwater temperature.
60.2%P / 34%F	MPS point on the ELLLA boundary line, with feedwater temperature reduction.
60.2%P / 34%F	MPS point on the ELLLA boundary line, with normal feedwater temperature.

9.2.3 Results

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

The flow-induced loads in the MELLLA condition (at the CLTP and 80.7% RCF) are slightly higher than the current uprated ELLLA condition (at the CLTP and 88% RCF) due to the increased subcooling in the downcomer associated with the MELLLA condition. From ELLLA to MELLLA, the downcomer subcooling increases thereby increasing the critical flow and the mass flux out of the break in a postulated RSLB. As a result, the flow-induced loads in MELLLA conditions increase slightly.

9.3 RPV Internals Structural Integrity Evaluation

The structural integrity of the RPV internals was qualitatively evaluated for the loads associated with MELLLA operation for CGS. The loads considered for MELLLA include Dead weights, Seismic Loads, RIPDs, Acoustic and Flow induced Loads due to RSLB LOCA, SRV, LOCA, AP loads, Jet Reaction (JR) loads, Thermal loads, Flow Loads and Fuel Lift loads. The limiting flow conditions and thermal conditions were considered. The RPV internals (excluding CRD Mechanism) are not certified to the ASME Code; however, the requirements of the ASME Code Section III are used as guidelines in their design basis analysis. The following RPV internal components were evaluated:

- Shroud
- Shroud support

- Core Plate
- Top Guide
- CRD Housing/CRD Mechanism
- Control Rod Guide Tube
- Orificed Fuel Support
- Fuel channel
- Shroud Head and Separator Assembly (Including Shroud Head Bolts)
- Jet Pump Assembly
- Access hole cover
- Core Spray Line and Sparger
- Feedwater Sparger
- Low Pressure Coolant Injection (LPCI) Coupling
- Steam Dryer
- In-core housing and Guide Tube
- Core Differential Pressure & Liquid Control Line

The above RPV internals are currently qualified for CLTP with FFWTR operation. All applicable loads except the AP/JR and RIPD loads are unaffected, remain bounded, or change insignificantly with respect to CLTP with FFWTR. The MELLLA-based AP/JR and RIPDs loads have increased for some RPV internals with respect to their current design basis loads. However, adequate stress margin exists to accommodate increases in the MELLLA-based AP/JR and RIPD loads. It was concluded based on the evaluation that the Normal, Upset, Emergency and Faulted condition stresses and fatigue usage factors remain within the design basis ASME Code Section III allowable stress limits for all RPV internals for ARTS/MELLLA. The results of the structural evaluation of the RPV internals components are shown in Table 9-4. All RPV internals remain structurally qualified for operation in the MELLLA condition.

9.4 Reactor Internals Vibration

9.4.1 Approach/ Methodology

To ensure that the flow-induced vibration (FIV) response of the reactor internals is acceptable, a single reactor for each product line and size undergoes an extensively instrumented vibration test during initial plant startup. After analyzing the results of such a test and assuring that all responses fall within acceptable limits of the established criteria, the tested reactor is classified as a valid prototype in accordance with Regulatory Guide 1.20 (Reference 37). All other reactors of the same product line and size are classified as non-prototype and undergo a less rigorous confirmatory test.

Tokai Unit 2 was designated as the prototype plant for BWR5, 251-inch diameter reactors in accordance with Regulatory Guide 1.20 (Reference 37). An FIV test was performed at Tokai 2 and data collected during plant start-up between October 1977 and July 1978. An FIV test also was performed at CGS and data collected during plant start-up between September 1984 and December 1984. The critical reactor internals were instrumented with vibration sensors and the reactor was tested up to 106% core flow at 100% rod line. These data were used in the current CGS ARTS/MELLLA evaluation. For the components that were not instrumented in above two plants, test data from other plants and test facilities are used.

CGS is currently licensed to operate at an ICF of up to 106% of RCF (108.5 Mlbs/hr) at 100% of CLTP. For ARTS/MELLLA operation, the rated power output remains the same, but core flow is reduced to 80.7% of RCF at 100% of CLTP.

9.4.2 Inputs/Assumptions

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

Parameter	Input
Plant data selected for flow induced vibration (FIV) evaluation	Tokai Unit 2 was designated as the prototype plant for BWR5, 251-inch diameter reactors in accordance with Regulatory Guide 1.20 (Reference 37). FIV test was performed at Tokai 2 and data collected during plant start-up between October 1977 and July 1978. FIV test also was performed at CGS and data collected during plant start-up between September 1984 and December 1984 (Reference 38). The critical reactor internals were instrumented with vibration sensors and the reactor was tested up to 106% core flow at 100% rod line. These data were used in the current CGS ARTS/MELLLA evaluation. For the components that were not instrumented in above two plants, test data from other plants and test facilities are used.
Target plant conditions in the MELLLA region selected for component evaluation	CLTP of 3486 MWt and 80.7% of RCF at 100% of CLTP (100% rod line).
GE stress acceptance criterion of 10,000 psi is used for all stainless steel components	Limit is lower than the more conservative value allowed by the current ASME Section III design codes for the same material (Reference 39), and is bounding for all stainless steel material. The ASME Section III value is 13,600 psi for service cycles equal to 10^{11}.

9.4.3 Analyses Results

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

The reactor internals vibration measurements report for plants Tokai 2, CGS and other plants if needed were reviewed to determine which components are likely to have significant vibration at the MELLLA conditions.

For the shroud/top guide, shroud head, separators, and the steam dryer, the vibrations are a function of the steam flow, which at MELLLA conditions is bounded by the steam flow at CLTP. For the Feedwater sparger, the vibrations are a function of the Feedwater flow, which at MELLLA conditions is bounded by the Feedwater flow at CLTP.

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

For Jet Pumps, the vibration depends on the core flow. There is no increase in the maximum flow during MELLLA compared to CLTP; therefore, vibrations due to flow are acceptable. In addition, CGS has proactively installed slip joint clamps at all 20 jet pumps to eliminate any potential slip joint leakage induced vibration.

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

For jet pump sensing lines (JPSLs), the VPF at MELLLA conditions was compared with the JPSL natural frequency and it was concluded that they were acceptable.

The FIV evaluation is conservative for the following reasons:

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

Therefore, the FIV will remain within acceptable limits.

9.5 Conclusion

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

Table 9-1 Flow-Induced Loads on Shroud and Jet Pumps for CGS

Component	Parameter	Loads [(1)]
Shroud	Baseline Force (kips)	95.498
	Baseline Moment at the Shroud Centerline (10^6 in-lbf)	8.390
Jet Pump	Baseline Force (kips)	6.229
	Baseline Moment at the Jet Pump Centerline (10^6 in-lbf)	0.370
Component	**Operating Condition**	**Load Multiplier**
Jet Pump Shroud	102%P / 100%F	1.0000
	102%P / 100%F FWTR	1.0484
	102 %P / 80.7%F (MELLLA) FWTR	1.1650
	58.7%P / 32.3%F NFWT (MELLLA) MPS	1.5558
	58.7%P / 32.3%F FWTR (MELLLA) MPS	1.8246
	60.2%P / 34%F NFWT (ELLLA) MPS	1.5052
	60.2%P / 34%F FWTR (ELLLA) MPS	1.7794

(1) Loads at rated conditions (102% power/100% core flow).

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

Component	Conditions	Force[(1)] (kips)	Effective[(1)] Force (kips)	Moment [(1)] (10^6 in-lbf)	Effective Moment[(1)] (10^6 in-lbf)
Shroud	All Conditions	2182.412	1079.391	291.708	121.563
Jet Pump	All Conditions	30.994	26.866	1.770	1.607

(1) The results are applicable for all rated and off-rated conditions

Table 9-3 Maximum Acoustic Loads on Shroud Support (MELLLA)

Component	Parameter	Unit	Loads [(1)]
Shroud Support	Total Vertical Force	kips	2202
	Moment at the Shroud Support Plate Outside Edge Nearest the Break	10^6 in-lbf	323.6
	Half Period	sec	0.037

(1) The results are applicable for all rated and off-rated conditions

Table 9-4 Governing Stress Results for RPV Internals

No	Component	CLTP			ARTS/MELLLA					
		Service Level	Unit	Value	Location	Service Level	Stress Category /Other	Unit	Value [1]	Allowable Value [2]
1	Shroud	B	psi	12,320	Top Guide Wedge	B	$P_m + P_b$	psi	12,730	21,450
2	Shroud Support	B	psi	25,540	Legs	B	$P_m + P_b$	psi	25,540	28,100
3	Core Plate	B	lbs./CRGT	986	Longest Beam	B	Buckling	lbs./ CRGT	1,016	1,179
4	Top Guide	B	psi	28,548	Longest Beam	B	$P_m + P_b$	psi	28,548	31,690
5.a	CRD Housing (Outside-RPV Portion)	B	psi	15,450	CRD Housing @ RPV Bottom Head	B	$P_m + P_b$	psi	15,450	24,900
5.b	CRD Housing (Inside-RPV Portion)	B	psi	11,925	CRD Housing @ RPV Stub Tube	B	$P_m + P_b$	psi	11,925	16,185
5.c	CRD Mechanism	B	psi	24,700	CRD Outer Tube	B	$P_m + P_b$	psi	24,700	26,100
6.a	Control Rod Guide Tube	B	psi	8,189	CRGT Flange (Base)	B	$P_m + P_b$	psi	8,189	24,000
6.b	Control Rod Guide Tube	B	psi	9,037	Mid-span	B	$P_m + P_b$	psi	9,100	16,000
6.c	Control Rod Guide Tube	B	N/A	0.39	Body	B	Buckling	N/A	0.40	0.45
7	Orificed Fuel Support (OFS)	B	lbs.	14,894 [3]	OFS Body	B	Load	lbs.	14,895[3]	35,590[3]
8	Fuel Channel				Qualified By GEH (GNF) proprietary method					
9	Shroud Head and Separators Assembly (Incl. Shroud Head Bolts)	B	psi	7,926	Shroud Head Bolt	B	P_m	psi	7,909	16,900
10	Jet Pump Assembly	D	psi	54,427	Riser Brace	D	$P_m + P_b$	psi	54,427	60,840

No	Component	CLTP			ARTS/MELLLA					
		Service Level	Unit	Value	Location	Service Level	Stress Category /Other	Unit	Value [1]	Allowable Value [2]
11	Access Hole Cover (Top Hat Design)	B	psi	10,012	Cover	B	$P_m + P_b$	psi	10,012	20,580
12.a	Core Spray Line	B	psi	19,890	Elbow	B	$P_m + P_b$	psi	19,890	23,850
12.b	Core Spray Sparger	B	psi	6,560	Tee Junction	B	P_m	psi	6,560	21,450
13	Feedwater Sparger	B	N/A	0.88	Sparger pipe to Endplate Weld	B	Fatigue Usage	N/A	0.88	1
14	In-Core Housing and Guide Tube	B	psi	25,160	In-core housing @ RPV Penetration	B	$P_m + P_b$	psi	25,160	25,400
15	Core Differential Pressure and Liquid Control Line	C	psi	17,015[4]	Unknown	B	$P_m + P_b$	psi	17,015[4]	36,900
16	Low Pressure Coolant Injection (LPCI) Coupling	C	psi	27,600	Support Ring	C	$P_m + P_b$	psi	27,600	31,400
17	Steam Dryer	D	kips	75.15	Lifting Rod	D	Buckling	kips	75.15	88.99

Notes:

[1] Stresses/loads values reported are for the limiting loading condition, with the least margin of safety.
[2] AVs are consistent with the original design basis.
[3] For OFS, Calculated and Allowable loads provided are in vertical downward direction.
[4] For the Core Differential and Liquid Control Line, the calculated stress shown is based on Absolute summation of upset loads. Actual stress based on SRSS methodology will be less.

10.0 ANTICIPATED TRANSIENT WITHOUT SCRAM

10.1 Approach/Methodology

The basis for the current ATWS requirements is 10 CFR 50.62. This regulation includes requirements for an ATWS-RPT, an Alternate Rod Insertion (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate. The purpose of the ATWS analysis is to demonstrate that these systems are adequate for operation in the MELLLA region. This is accomplished by performing a plant-specific analysis in accordance with the approved licensing methodology (Reference 40) to demonstrate that ATWS acceptance criteria are met for operation in the MELLLA region.

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

Three ATWS events for CGS were re-evaluated at the MELLLA point (100% of CLTP and 80.7% of RCF) with ARI assumed to fail, thus requiring the operator to initiate SLCS injection for shutdown. These events were: (1) Closure of all MSIVs (MSIVC), (2) Pressure Regulator Failure Open (PRFO) to Maximum Steam Demand Flow, and (3) Loss of Offsite Power (LOOP).

The MSIVC and PRFO events result in reactor isolation and a large power increase without scram. These events are the most limiting for fuel integrity and RPV integrity.

The LOOP event does not result in reduction in the number of Residual Heat Removal (RHR) cooling loops, this event is not potentially limiting for suppression pool or containment integrity.

The Inadvertent Opening of a Relief Valve (IORV) event was also considered, but found to be non-limiting. As a result of the sequence of events for the IORV event, it is non-limiting with respect to the ATWS acceptance criteria. Peak suppression pool temperature and containment pressure are limited because the main condenser remains available for most of the event. RPV and fuel integrities are not challenged because the vessel is shutdown (via boron injection) by the time the MSIVs isolate.

Because ATWS events are beyond design basis events and involve more than one failure, boiling transition is not the applicable acceptance criterion. For ATWS, the 10 CFR 50.46 criteria for fuel integrity have been adopted and peak cladding temperatures are calculated to be well below 2200°F. Therefore, boiling transition is not a fuel integrity criterion. An inadvertent two-pump trip would result in a power decrease as flow is reduced to natural circulation. There would be no boiling transition consequences. An automatic scram may not be generated unless the core is unstable. The stability protection hardware would scram the reactor to protect the fuel in these situations.

The subject of ATWS with instability has been covered generically for the BWR fleet in References 41 and 42. Reference 41 states that for ATWS with instability, the fuel integrity criterion is that fuel damage be limited so as not to significantly distort the core, impede core cooling, or prevent safe shutdown. The potentially limiting non-isolation ATWS event with respect to fuel integrity has been determined in Reference 41 to be a turbine trip with full bypass capacity. The full bypass capacity is more limiting than when only partial bypass is available because the full bypass capability eliminates the interference that SRV cycling will have with the instability oscillations. This event also results in a large FW temperature reduction, which also aggravates the potential instability. CGS has a much smaller bypass capacity than that assumed in the generic analysis and thus, is bounded by the generic study. Another event than can lead to instability is a two-pump trip. This event would have a similar behavior without as much feedwater temperature decrease. Non-isolation ATWS events do not put a demand on the reactor vessel as there is no pressurization and no energy is transferred to the suppression pool. Therefore, vessel and containment integrity criteria are met.

If one of these limiting non-isolation events occurs with a core instability and without a scram, then emergency operating procedures require operator action to reduce water level to below the feedwater sparger. This reduces the core subcooling, oscillation magnitude and mitigates the effect on fuel cladding heat up to meet the acceptance criteria.

The following ATWS acceptance criteria were used to determine acceptability of the CGS operation in the MELLLA region:

1. Fuel integrity:
   - Maximum clad temperature < 2200° F
   - Maximum local clad oxidation < 17%
2. RPV integrity:
   - Peak RPV pressure < 1500 psig (ASME service level C)
3. Containment integrity:
   - Peak suppression pool bulk temperature < 204.5°F
   - Peak containment pressure < 45 psig

The adequacy of the margin to the SLCS relief valve lifting as described in NRC Information Notice 2001-13 (Reference 43) was also assessed.

10.2 Input Assumptions

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

Analytical Assumptions	Bases/Justifications
The reactor is operating at 3486 MWt (100% of CLTP)	ATWS analyses are performed at nominal rated core power, consistent with generic ATWS evaluation bases
Both beginning-of-cycle (BOC) and end-of-cycle (EOC) nuclear dynamic parameters were used in the	Consistency with generic ATWS evaluation bases

Analytical Assumptions	Bases/Justifications
calculations	
Dynamic void reactivity are based on CGS Cycle 20 data	ATWS analyses are performed conservatively compared to a nominal basis, which bounds cycle to cycle variation
Four SRV OOS, specified as the valves with the lowest setpoints	Consistency with the CGS current licensing basis
The relief mode of the dual mode SRV is used in the analysis to limit peak vessel pressure	Consistency with generic ATWS evaluation bases
MSIV closure starts at event initiation (time zero) for the MSIVC event	Consistency with generic ATWS evaluation bases
The PRFO event is initiated by the failure of the pressure regulator in the open position.	Consistency with generic ATWS evaluation bases

10.3 Analyses Results

Table 10-2 presents the results for the MSIVC and PRFO events. As shown, the peak vessel bottom pressure for this event is 1364 psig, which is below the ATWS vessel overpressure protection criterion of 1500 psig.

The highest calculated peak suppression pool temperature is 180°F, which is below the ATWS limit of 204.5°F. The highest calculated peak containment pressure is less than 10.0 psig, which is below the ATWS limit of 45 psig. Thus, the containment criteria for ATWS are met.

Analyses have also been performed for one pump operation with 44% boron-10 enrichment. The one pump operation increases the SLCS transport delay due to the reduced volumetric flow in the system. As a result, the peak pool temperature was determined to be 187°F, which is well below the temperature limit of 204.5°F. The peak containment pressure was determined to be less than 12 psig, well below the 45 psig limit. Other acceptance criteria are not affected by one SLCS pump operation as the peak values occur before SLCS initiation.

Coolable core geometry is ensured by meeting the 2200°F PCT, and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. The limiting PCT is determined to be 1572°F, which is significantly less than the ATWS limit. The fuel cladding oxidation is insignificant and less than the 17% local limit.

The maximum SLCS pump discharge pressure depends primarily on the SRV setpoints. The maximum SLCS pump discharge pressure during the limiting ATWS event using one SLCS 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 SLCS initiation.

The relief valves used for the SLCS at CGS have a setpoint of 1400 psig and a drift tolerance of −28 psig, resulting in a lower setpoint tolerance of 1372 psig. There is 162.5-psid margin between the maximum SLCS discharge pressure of 1209.5 psig and the lower setpoint of 1372 psig. A margin of 30-psid from the relief valve lower setpoint is needed to adequately accommodate the SLCS pump pressure pulsation. Therefore, the margin from the lower setpoint is adequate to prevent the SLCS relief valve from lifting during SLCS operation to meet the guidelines published in NRC Information Notice 2001-13 (Reference 43).

10.4 Conclusions

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

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

Parameter	Current Analysis
Dome Pressure (psia)	1035
MELLLA Core Flow (Mlbm/hr / % rated)	87.6 / 80.7
Core Thermal Power (MWt / %CLTP)	3486 / 100.0
Steam / Feed Flow (Mlbm/hr / %NBR)	15.013 / 100
Sodium Pentaborate Solution Concentration in the SLCS Storage Tank (% by weight)	13.6
Boron-10 Enrichment (atom %)	19.8
SLCS Injection Location	HPCS
Number of SLCS Pumps Operating	2
SLCS Injection Rate (gpm)	82.4
SLCS Liquid Transport Time (sec)	321
Initial Suppression Pool Liquid Volume (ft^3)	112197
Initial Suppression Pool Temperature (°F)	90
Number of RHR Heat Exchanger Cooling Loops	2
RHR Heat Exchanger Design Effectiveness per Loop (BTU/sec·°F)	289.0
Number of RHR Heat Exchanger Loops Available for LOOP Event	2
RHR Heat Exchanger Design Effectiveness during LOOP (BTU/sec·°F)	289.0
RHR Service Water Temperature (°F)	90
Transient time at which the RHR suppression pool cooling is established (seconds)	660
High Dome Pressure ATWS-RPT Setpoint (psig)	1170
DSRV Capacity – per valve (lbm/hr) / Reference Pressure (psig) / Accumulation (%)	876500 / 1165 / 3
Dual Safety Relief Valve (DSRV) Configuration	18 DS/RV (4 OOS)

Table 10-2 Summary of ATWS Calculation Results

Acceptance Criteria	Criteria Limit	Limiting Results			
		MSIVC BOC	MSIVC EOC	PRFO BOC	PRFO EOC
Peak Vessel Pressure (psig)	1500	1345	1349	1364	1358
Peak Cladding Temperature (°F)*	2200	<1572	<1572	<1572	≤1572
Peak Local Cladding Oxidation (%)	17	< 17	< 17	< 17	< 17
Peak Suppression Pool Temperature (°F)*	204.5	177	180	177	179
Peak Containment Pressure (psig)	45	< 10	< 10	< 10	< 10

* Not specifically calculated. Analysis evaluation determined a bounding value of 1572 °F. PRFO event at EOC is the limiting case.

11.0 STEAM DRYER AND SEPARATOR PERFORMANCE

The ability of the steam dryer and separator to perform their design functions during MELLLA operation was evaluated. The CGS plant-specific evaluation concluded that the performance of the steam dryer and separator remains acceptable (moisture content $\leq$ 0.1 weight %, carryunder is acceptable and dryer skirt remains covered at L4, the low water level alarm) in the MELLLA region.

MELLLA decreases the core flow rate, resulting in an increase in separator inlet quality for constant reactor thermal power. These factors, in addition to core radial power distribution, influence steam separator-dryer performance. The CGS steam separator/dryer performance was evaluated on a plant-specific basis to determine the influence of MELLLA on the steam dryer and separator operating conditions; (a) the entrained steam (i.e., carryunder) in the water returning from the separators to the reactor annulus region, (b) the moisture content in the steam leaving the RPV into the main steam lines and (c) the margin to dryer skirt uncovery.

12.0 HIGH ENERGY LINE BREAK

The following HELBs were evaluated for the effects of MELLLA:

- Main Steam Line Break (MSLB) in the main steam tunnel.
- Feedwater Line Break (FWLB) in the main steam tunnel.
- Reactor Core Isolation Cooling (RCIC) line breaks (various locations).
- Reactor Water Cleanup (RWCU) line breaks (various locations).

The effect of increased subcooling due to MELLLA was evaluated based on the HELB mass / energy release profiles assumed in the current CGS design basis. Analyses were performed at rated conditions, and MELLLA conditions at minimum Reactor Recirculation System (RRS) pump speed with consideration of FFWTR/FWHOOS for the break locations listed above, taking into account the changes in enthalpy and pressure at each operating condition.

With consideration of flashed steam that maximizes subcompartment pressurization, the mass and energy release profiles assumed in the current CGS design basis HELB analyses for the FWLB line break in the main steam tunnel remain bounding at the full power and normal feedwater temperature for the MELLLA conditions listed above.

The mass and energy releases at the MELLLA state points for the MSLB in the main steam tunnel and the RCIC line break were found to be unchanged from the HELB mass / energy release profiles assumed in the current CGS design basis.

The mass and energy release profiles assumed in the current CGS design basis HELB analyses for the Reactor Water Clean-Up (RWCU) line breaks are bounding for the MELLLA conditions listed above.

The RWCU HELB analysis was performed using the GOTHIC model for the ARTS/MELLLA evaluation. This analysis was originally performed using the RELAP model. The results of the evaluation showed that there was good agreement between the original RELAP model and the GOTHIC replica. The only significant difference occurred at the beginning of the transient where RELAP chokes at a higher mass flow rate. Further review showed that RELAP maintained a higher pressure at the break. Both choke points are correct for the pressures calculated. This discrepancy had little effect on the total release.

The results for the total amount of energy released show that all of the GOTHIC models are bounded by the RELAP results. The GOTHIC benchmark shows a 2.8% decrease in energy released, a 3.0% decrease for the new high temperature conditions, and a 6.7% decrease for the low temperature conditions compared to the RELAP model. While the high temperature, high-pressure model showed a slight increase to the GOTHIC benchmark case, it is insignificant and remains well within the bounds of the original design basis.

CGS has evaluated the effects of the MELLLA operating condition on the RWCU HELB and concluded the results are acceptable with respect to the existing design criteria.

13.0 TESTING

Required pre-operational tests (i.e., PRNMS firmware upgrade) will be performed in preparation for operation at the MELLLA conditions with the ARTS improvements. Routine measurements of reactor parameters (e.g., Average Planar Linear Heat Generation Rate (APLHGR), LHGR, and MCPR) will be taken within a lower power test condition in the MELLLA region. Core thermal power and fuel thermal margins will be calculated using accepted methods to ensure current licensing and operational practice are maintained.

Measured parameters and calculated core thermal power and fuel thermal margins will be utilized to project those values at the RTP test condition. The core performance parameters will be confirmed to be within limits to ensure a careful monitored approach to RTP in the MELLLA region.

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

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

Steam separator and dryer performance will be evaluated by measuring the main steam line moisture content. The evaluation will be conducted near the RTP / MELLLA boundary corner. Other test condition P/F operating points may be tested as deemed appropriate prior to the RTP / MELLLA boundary corner test to demonstrate the test methodology or to determine the steam moisture content at the P/F conditions.

14.0 REFERENCES

1. GE Nuclear Energy, "Power Uprate with Extended Load Line Limit Safety Analysis for WNP-2," NEDC-32141P, Revision 0, June 1993.

2. GESTAR II General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-16, October 2007.

3. Global Nuclear Fuel, "Supplemental Reload Licensing Report for CGS Reload 19, Cycle 20," 0000-0098-0322-SRLR, Revision 0, March 2009.

4. CGS Final Safety Analysis Report, July 2009.

5. GE Nuclear Energy, "GE Instrument Setpoint Methodology," NEDC-31336P-A, September 1996.

6. NRC Regulatory Issue Summary, RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," August 24, 2006.

7. GE Atomic Power Equipment Department, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, Class II, J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, 1966.

8. GE Nuclear Energy, "Engineering Evaluation of Average Power Range Monitor, Rod Block Monitor, Technical Specifications (ARTS) Improvement Program," NEDE-30908P, April 1983.

9. GE Nuclear Energy, "Licensing Topical Report: Nuclear Measurement Analysis and Control Power Range Neutron Monitor Retrofit Plus Option III Stability Trip Function," Volumes 1 and 2, NEDC-32410P-A, October 1995.

10. NRC Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," Revision 2, May 31, 2005.

11. GE Nuclear Energy, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," NEDO-31960-A and NEDO-31960-A Supplement 1, November 1995.

12. GE Nuclear Energy, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," NEDO-32465-A, Revision 0, August 1996.

13. GE Nuclear Energy, "Plant-Specific Regional Mode DIVOM Procedure Guideline," GE-NE-0000-0028-9714-R1, June 2005.

14. GE Nuclear Energy, "Reactor Long-Term Stability Solution Option III: Licensing Basis Hot Channel Oscillation Magnitude for Columbia Generating Station with GEH PRNM System," 0000-0105-3839-R0, August 2009.

15. GE to BWR Owners' Group Detect and Suppress II Committee, "Backup Stability Protection (BSP) for Inoperable Option III Solution," OG 02-0119-260, July 17, 2002.

16. GEH Nuclear Energy, "ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions," NEDE-33213P-A, Revision 0, April 2009.

17. GE Nuclear Energy, "WNP-2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-32115P, Revision 2, March 2004.

18. GE Nuclear Energy, "Columbia Generating Station GE14 ECCS-LOCA Evaluation," 0000-0090-6853-R0, February 2009.

19. GE Nuclear Energy, "General Electric Model for LOCA Analysis in Accordance with 10 CFR 50 Appendix K," NEDO-20566A, September 1986.

20. Letter, C.O. Thomas (NRC) to J.F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," June 1, 1984.

21. GE Nuclear Energy, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," NEDE-23785-1-PA, Revision 1, October 1984.

22. GE Nuclear Energy, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident; Volume I, GESTR-LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," NEDC-23785-2-P, May 1984.

23. GE Nuclear Energy, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," NEDC-32950P, January 2000.

24. GE Nuclear Energy, "GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, March 2002.

25. GE Nuclear Energy, "The GE Pressure Suppression Containment Analytical Model," NEDO-10320, April 1971.

26. GE Nuclear Energy, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NEDO-20533, June 1974.

27. GE Nuclear Energy, "Mark II Containment Dynamic Forcing Functions Information Report," NEDO-21061, Revision 4, November 1981.

28. NUREG-0487, U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," October 1978, Supplement 1, September 1980, and Supplement 2, February 1981.

29. NUREG-0808, U.S. Nuclear Regulatory Commission, "Mark II Containment Program Load Evaluation and Acceptance Criteria," August 1981.

30. "Plant Design Assessment Report for SRV and LOCA Loads, Washington Public Power Supply System," FSAR Appendix G, Amendment 38, January 1983.

31. GE Atomic Power Equipment Department, "Maximum Two-Phase Vessel Blowdown from Pipes," APED-4827, F. J. Moody, April 20, 1965.

32. GE Nuclear Energy, "Mark II Pressure Suppression Containment Systems, An Analytical Model of Pool Swell Phenomena," NEDE-21544-P, December 1976.

33. GE Nuclear Energy, "4T Condensation Oscillation (4TCO) Test Program, Final Test Report," NEDE-24811-P, May 1980.

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

35. Numerical Applications, Inc., "Gothic Modeling of Columbia Generating Station RPV Annulus Pressurization during RRC Break," NAI-1471-01, Revision 0, September 21, 2009.

36. NRC Generic Letter 94-03, "Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," July 25, 1994.

37. NRC Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 2, May 1976.

38. GE Nuclear Energy, "Hanford-2 Reactor Internals Vibration Measurements," NEDE-30995P, Revision 0, May 1985.

39. ASME Section III, 1986, Division 1, Appendix I, Figure I-9.2.2, Design Fatigue Curve for Austenitic Steels.

40. GE Nuclear Energy, "Qualification of the One Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 – Volume 4)," NEDC-24154P-A, February 2000.

41. GE Nuclear Energy, "ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability," NEDO-32047-A, June 1995.

42. GE Nuclear Energy, "Mitigation of BWR Core Thermal Hydraulic Instabilities in ATWS," NEDO-32164, December 1992.

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

ATTACHMENT A

CGS ARTS-MELLLA Instrument Limits Calculation-RBM

GEH Number: 0000-0101-2139-R0
Revision Number: 0
DRF Number: 0000-0101-2133
Class I
March 2010

Instrument Limits Calculation

Energy Northwest

Columbia Generating Station

Rod Block Monitor

(NUMAC ARTS-MELLLA)

Contents:

This document is a supplement analysis data sheet to Reference 1. Included in this document in sequential order are:

1. The setpoint functions for the system
2. The setpoint function analyses inputs and the source reference of the inputs
3. The devices in the setpoint function instrument loop
4. The component analysis inputs and input sources
5. The calculated results
6. Input comments and result recommendations
7. References

System: Rod Block Monitor (RBM)

The following setpoint functions are included in this document:

1. Low Power Trip Setpoint (LTSP)
2. Intermediate Power Trip Setpoint (ITSP)
3. High Power Trip Setpoint (HTSP)
4. Low Power Setpoint (LPSP)
5. Intermediate Power Setpoint (IPSP)
6. High Power Setpoint (HPSP)

1. Function: RBM Rod Withdrawal Blocks

[[		

-
-]]

1.1 [[

				]]

2. Components:

[[

			]]

3. Summary Results:

[[

				]]

4. Comments and Recommendations:

1. [[

2.

3.

4.

5.

6.

7.

8.

9.

10.

11.

12.

13.

14.

15.

16.

17.

]]

5. References:

1. [[

2.

3.

    3.1.

    3.2.

4.

5.

6.

    6.1.

    6.2.

    6.3.

    6.4.

7.

    7.1.

    7.2.

    7.3.

    7.4.

]]

List of Regulatory 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
	one-time	continuing compliance	
Raise the Standby Liquid Control tank Boron enrichment level to be in compliance with revised analysis requirements.	X		Prior to startup from outage that installs the PRNM modification, (currently planned for spring 2011).
Incorporate Nominal Trip Setpoint values, and the methodology for determining these values, into the Licensee Controlled Specifications	X		Prior to startup from outage that installs the PRNM modification, (currently planned for spring 2011).
Implement administrative controls to: - control access APRM / OPRM panel and channel bypass switch - provide for manual bypass of the APRM / OPRM channels or protective functions - manage number of inoperable LPRMs - limit number of bypassed LPRMs between APRM gain calibrations - ensure minimum number of operable OPRM cells	X		Prior to startup from outage that installs the PRNM modification, (currently planned for spring 2011).
Document the Human Factors Engineering review in the final design package.	X		Prior to startup from outage that installs the PRNM modification, (currently planned for spring 2011).