LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM AND ARTS / MELLLA IMPLEMENTATION
Enclosure 3 – Attachment 2

NEDO-33507, Revision 1

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

January 2012

(non-proprietary version)

HITACHI

GE Hitachi Nuclear Energy

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ENERGY NORTHWEST
Columbia Generating Station
APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA)

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

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Revision Summary

Revision	Change Summary
0	Initial issue.
1	Updated Section 1.2.2 title to include Rod Block Monitor. Deleted discussion and reference to RIS 2006-17. Updated Attachment A proprietary markings and deleted select references.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

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:

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

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 and Rod Block Monitor 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.

1.3 Average Power Range Monitor Improvements

The functions of the APRM are integrated within the Nuclear Measurement Analysis and Control (NUMAC™) 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™ 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.

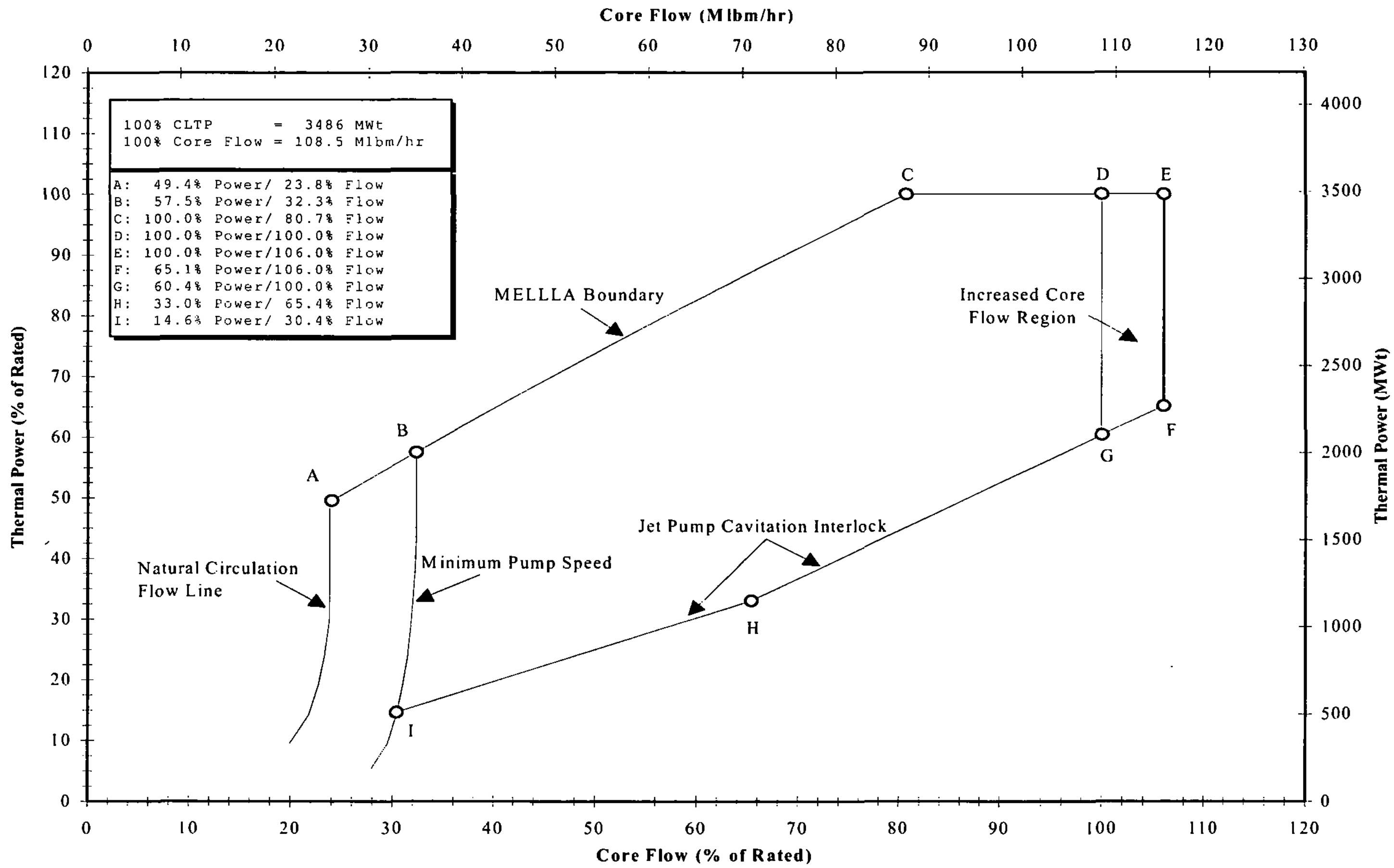

Figure 1-1 MELLLA Operating Range Power/Flow Map

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.

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  ]] 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. [[

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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. [[

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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, [[

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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. [[

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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

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[1]

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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.

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[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 ΔMCPR IMCPR		$BCCD_2$ Channel ΔMCPR 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

	Trip Level Setting (Note a)	
Function	**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.

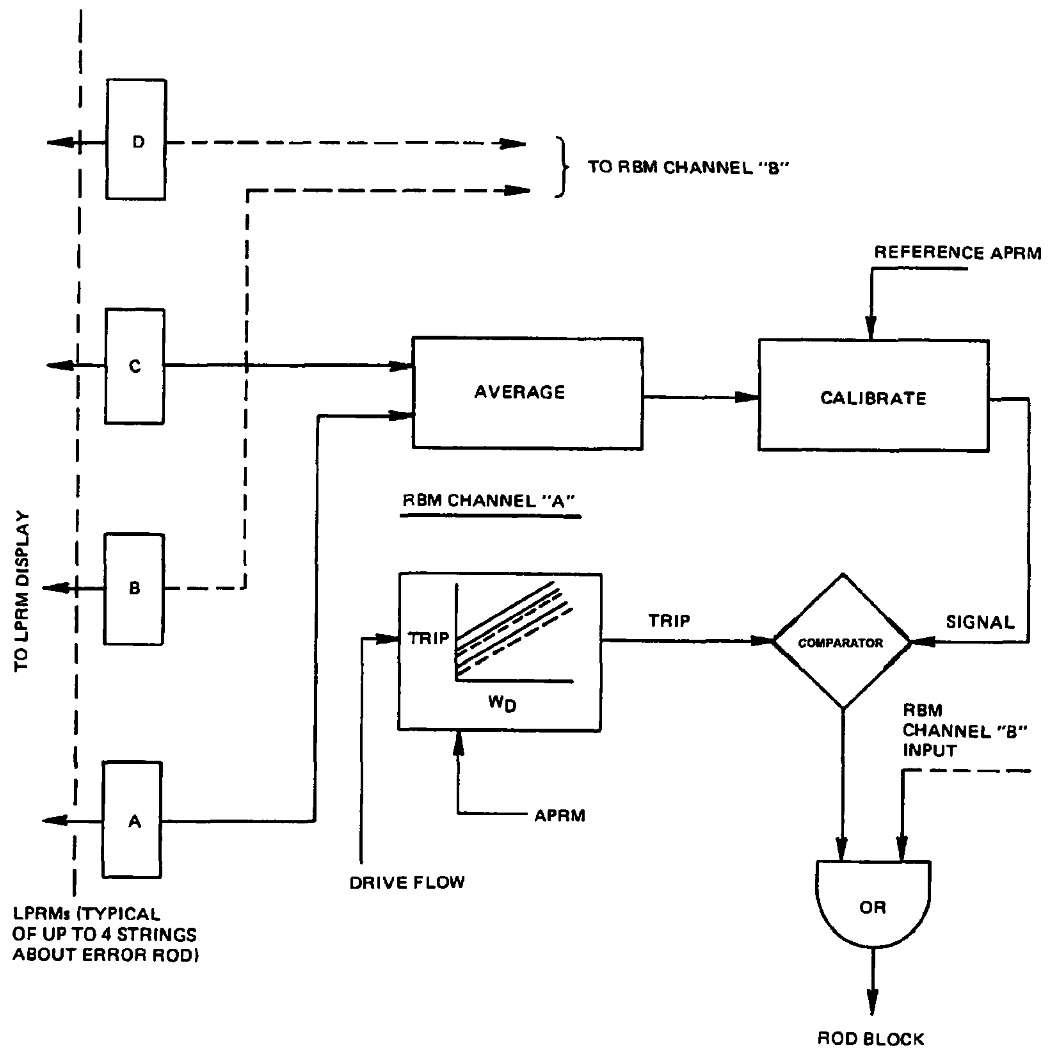

Figure 4-1 Conceptual Illustration of Current Flow-Dependent RBM System with AC/BD LPRM Assignment

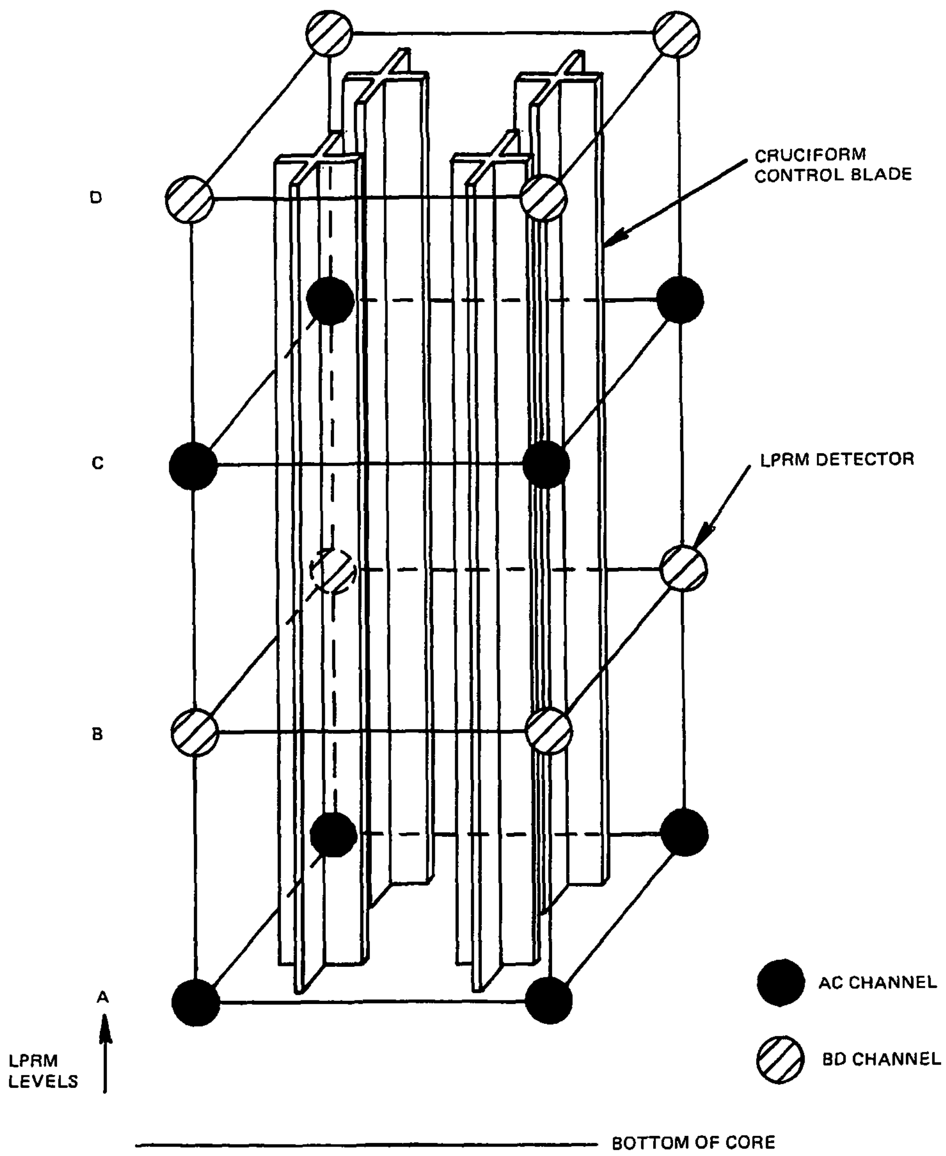

Figure 4-2 RBM Current AC/BD LPRM Assignment

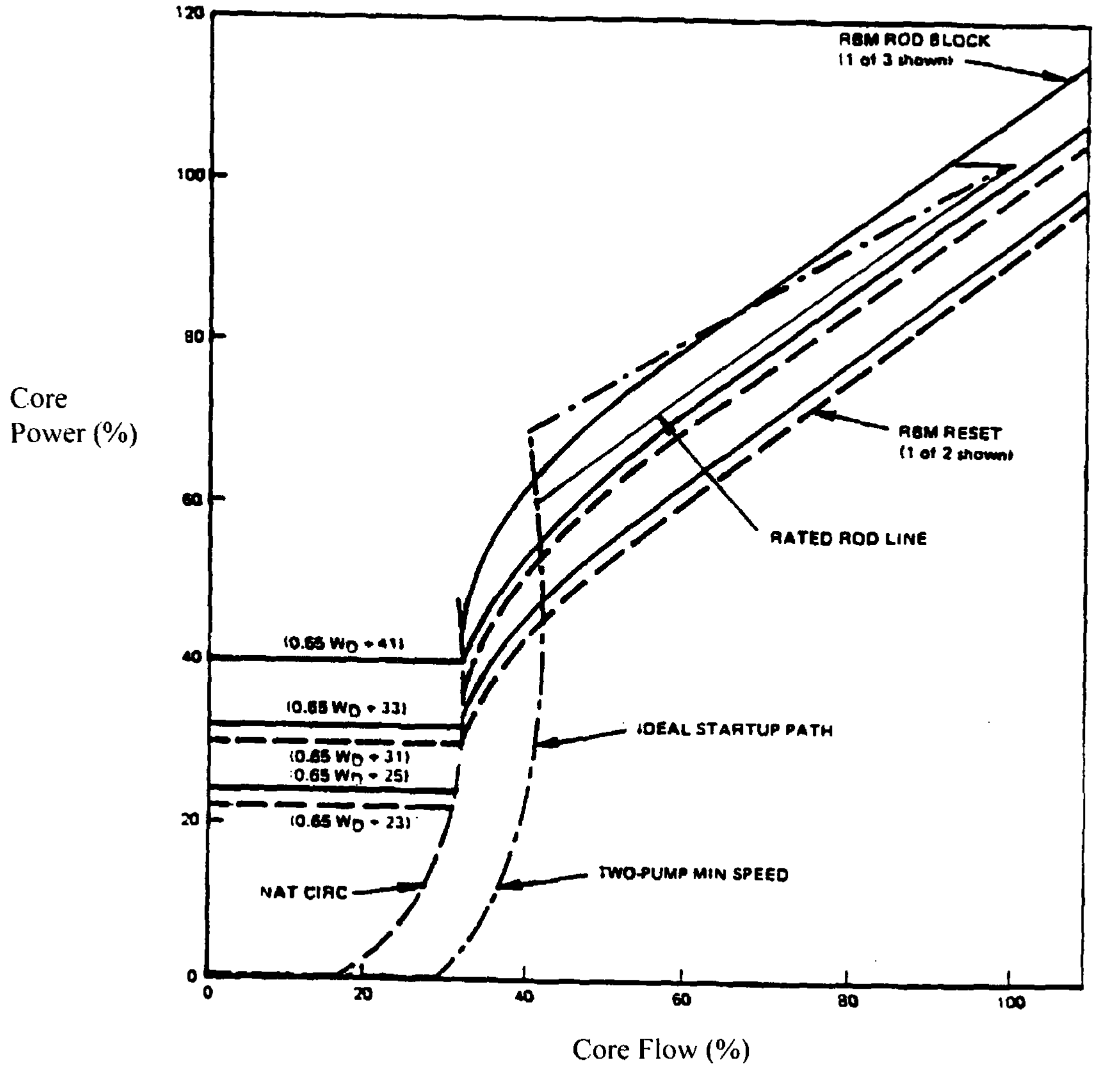

Figure 4-3 Typical RBM System Configuration Limits (Typical for 106% Setpoint)

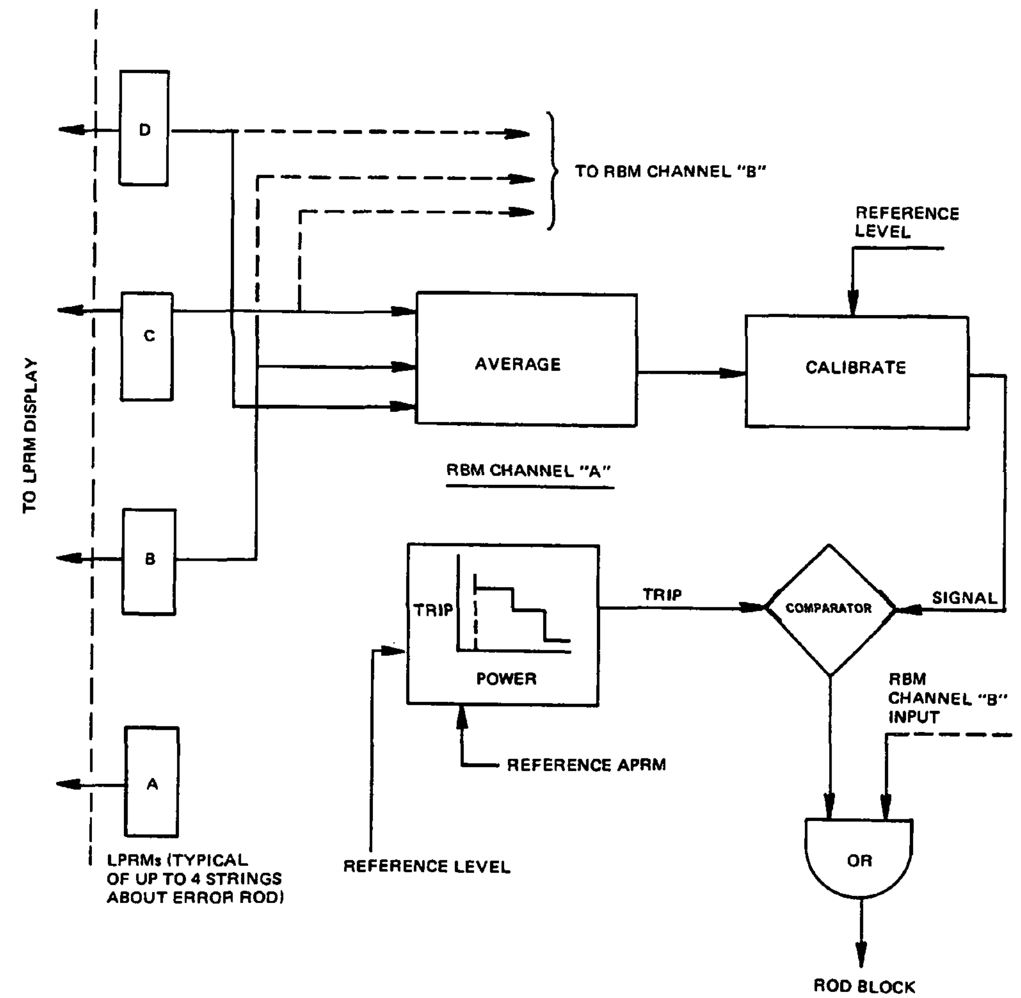

Figure 4-4 New Power-Dependent RBM System with $BCCD_1$/$BCCD_2$ LPRM Assignment

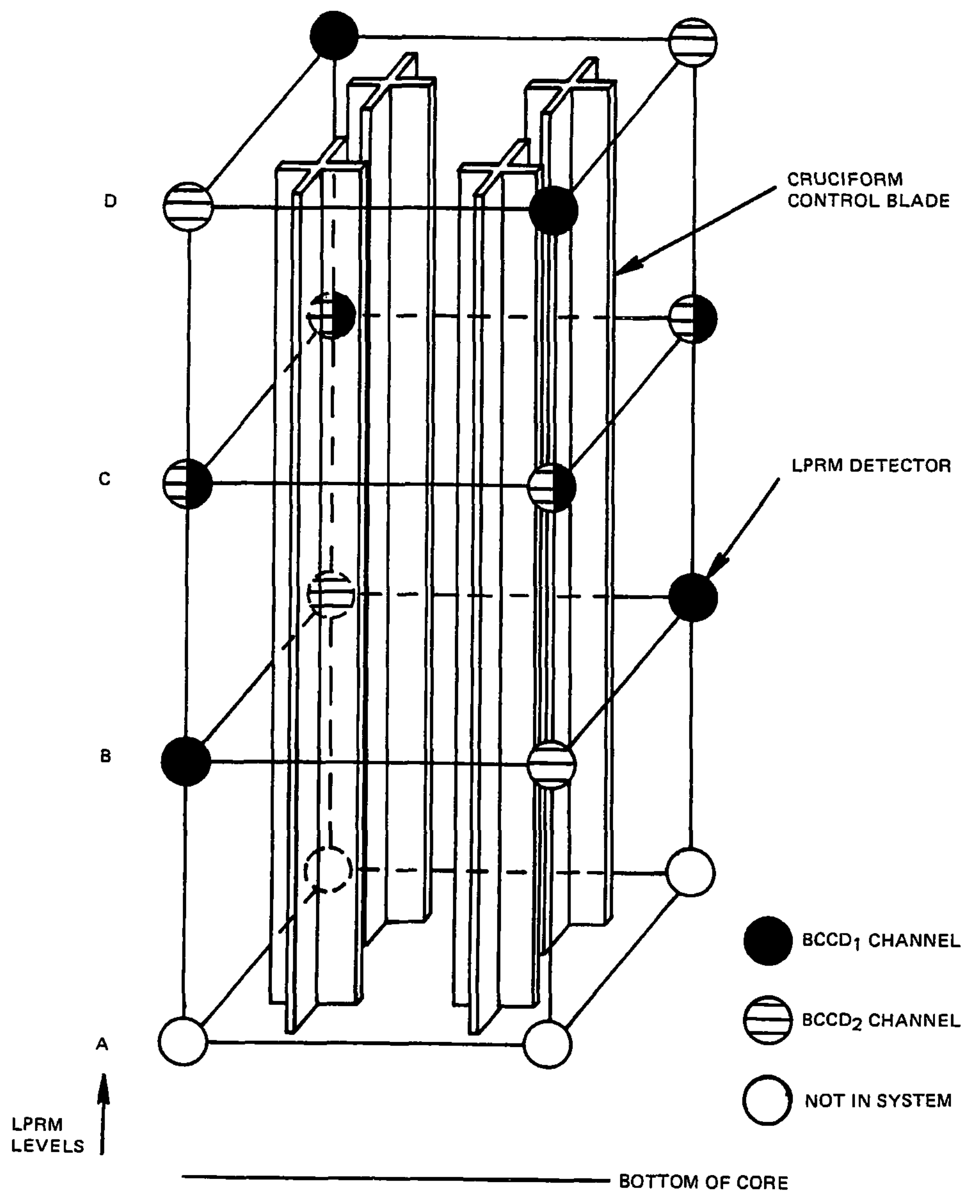

Figure 4-5 New RBM $BCCD_1$/$BCCD_2$ LPRM Assignment

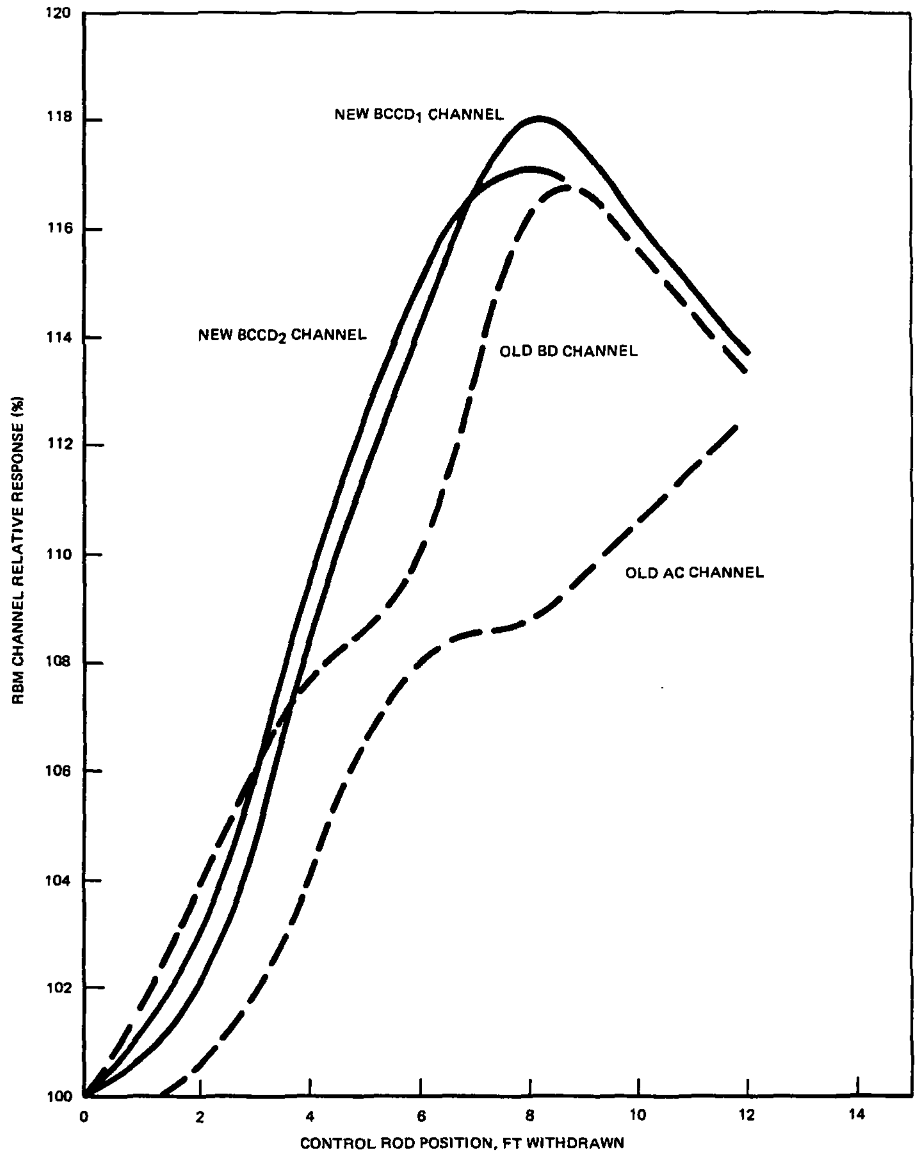

Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs)

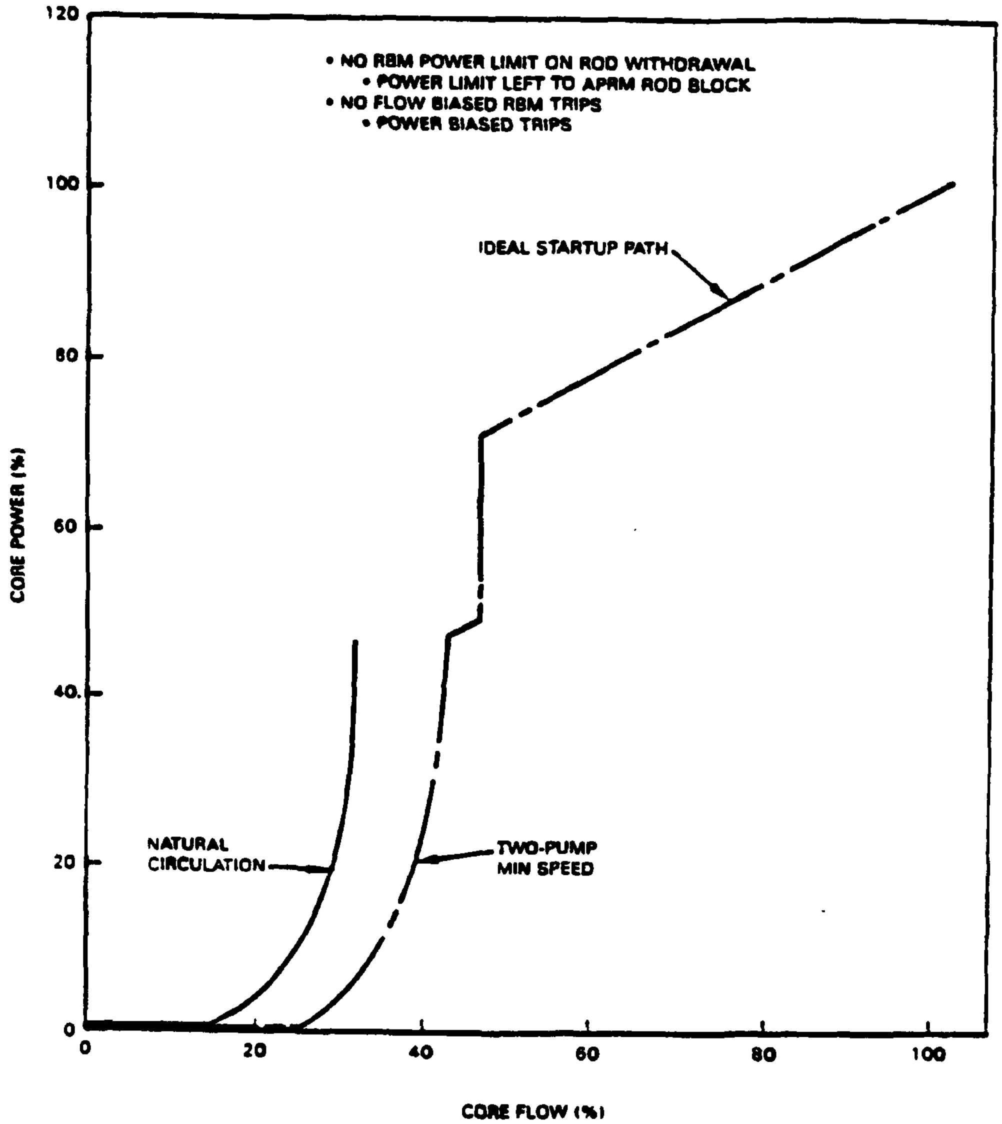

Figure 4-7 New RBM System Core Power Limit (Typical)

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Figure 4-8 Design Basis RWE MCPR Requirement Versus RBM Setpoint

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Figure 4-9 Design Basis MCPR Requirement for RWE (ARTS)

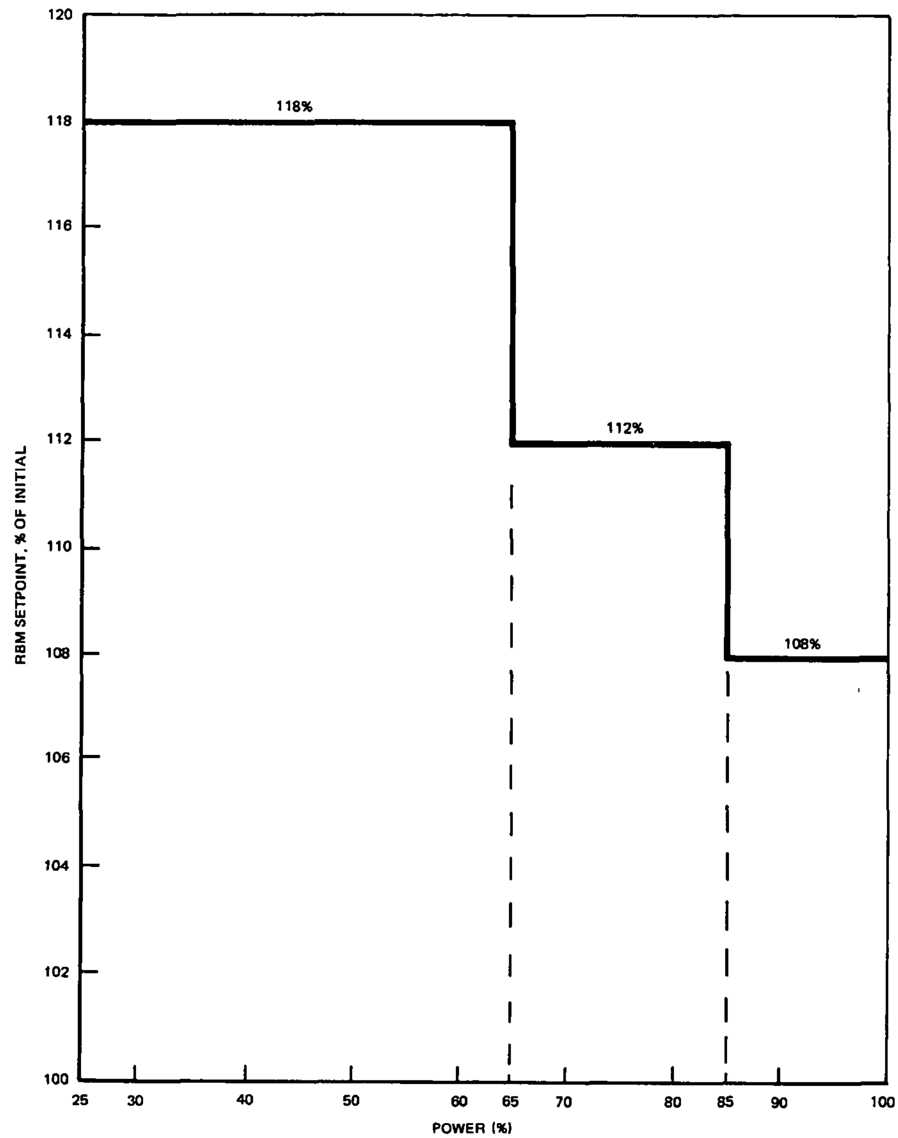

Figure 4-10 RBM Setpoint Versus Power (without Filter)

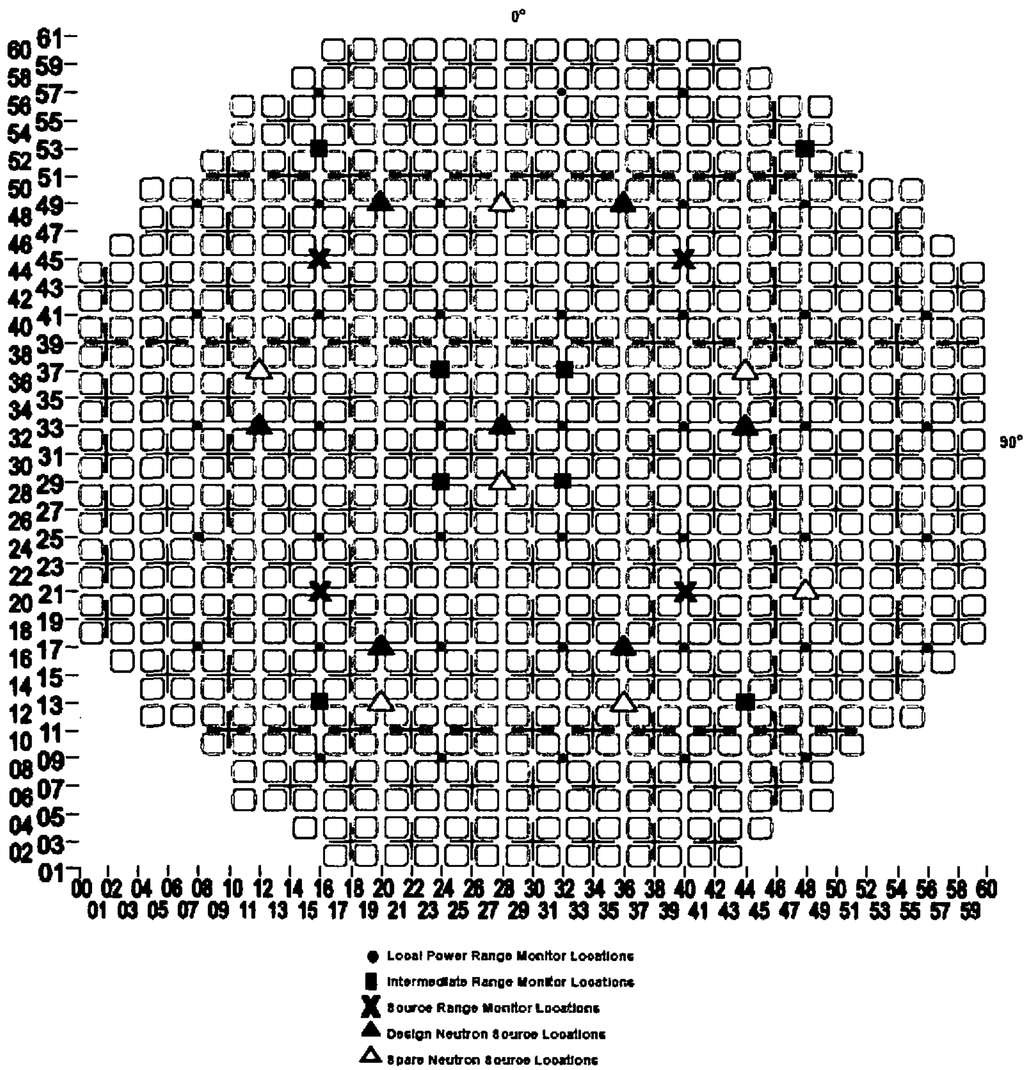

Figure 4-11 CGS Neutron Monitoring System

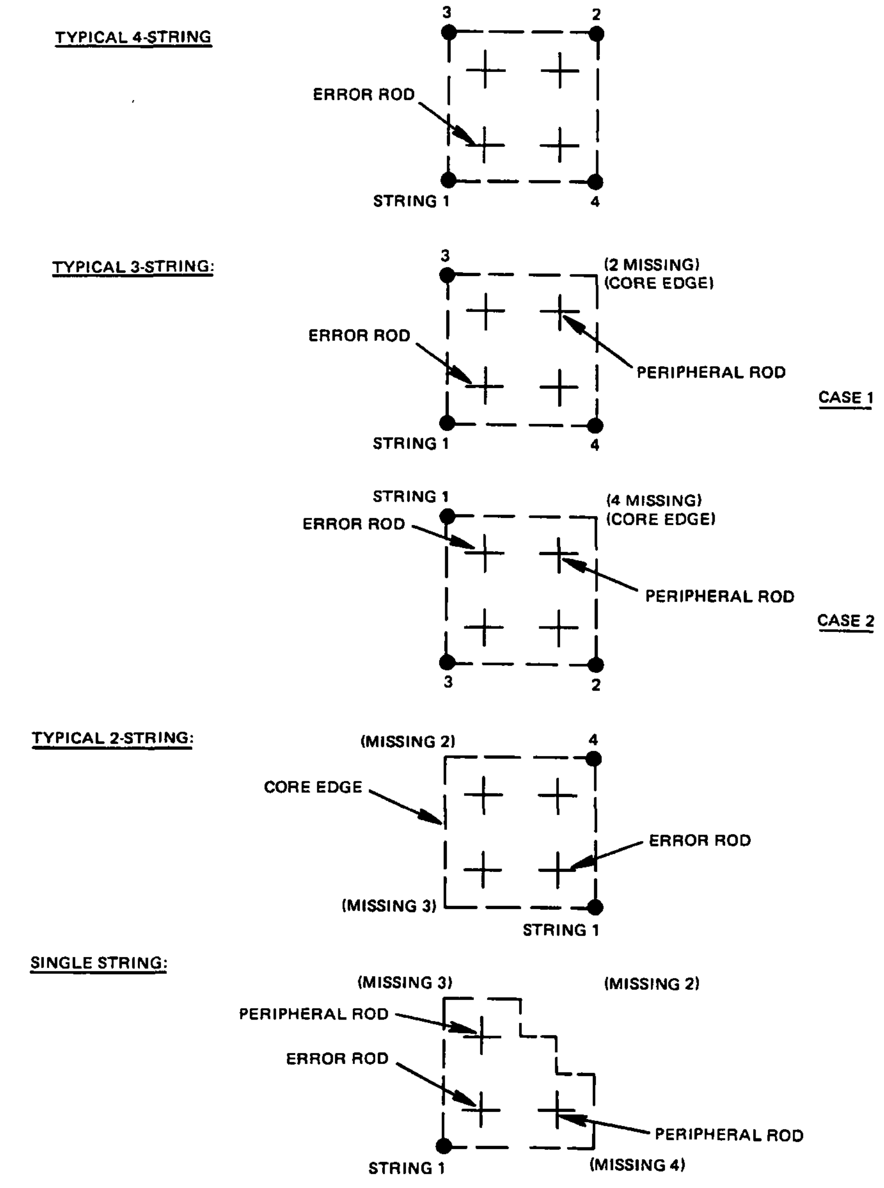

Figure 4-12 Rod Block Monitor Rod Group Geometries

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Figure 4-13 Results of LPRM Failure Rate Sensitivity Studies

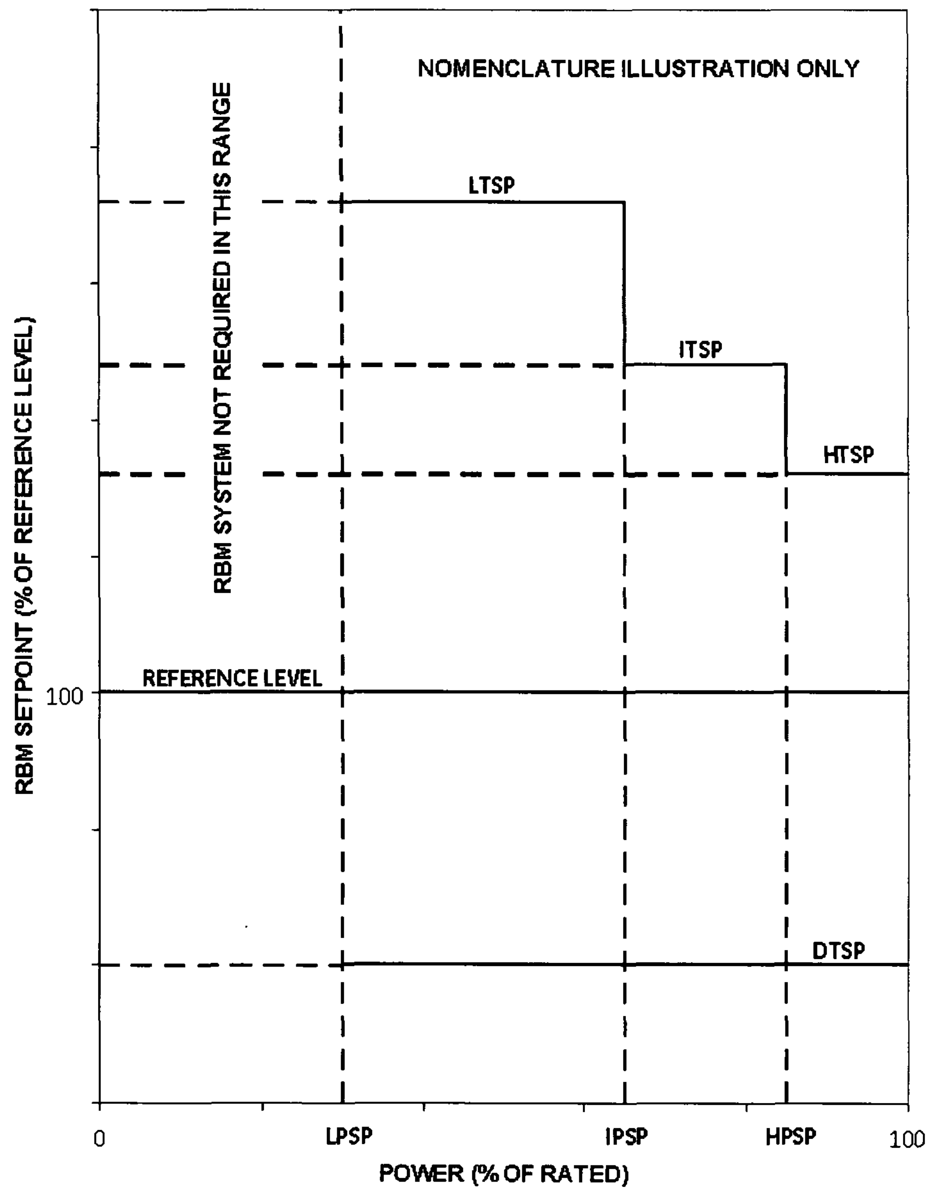


Figure 4-14 Power-Dependent RBM Trip Nomenclature

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

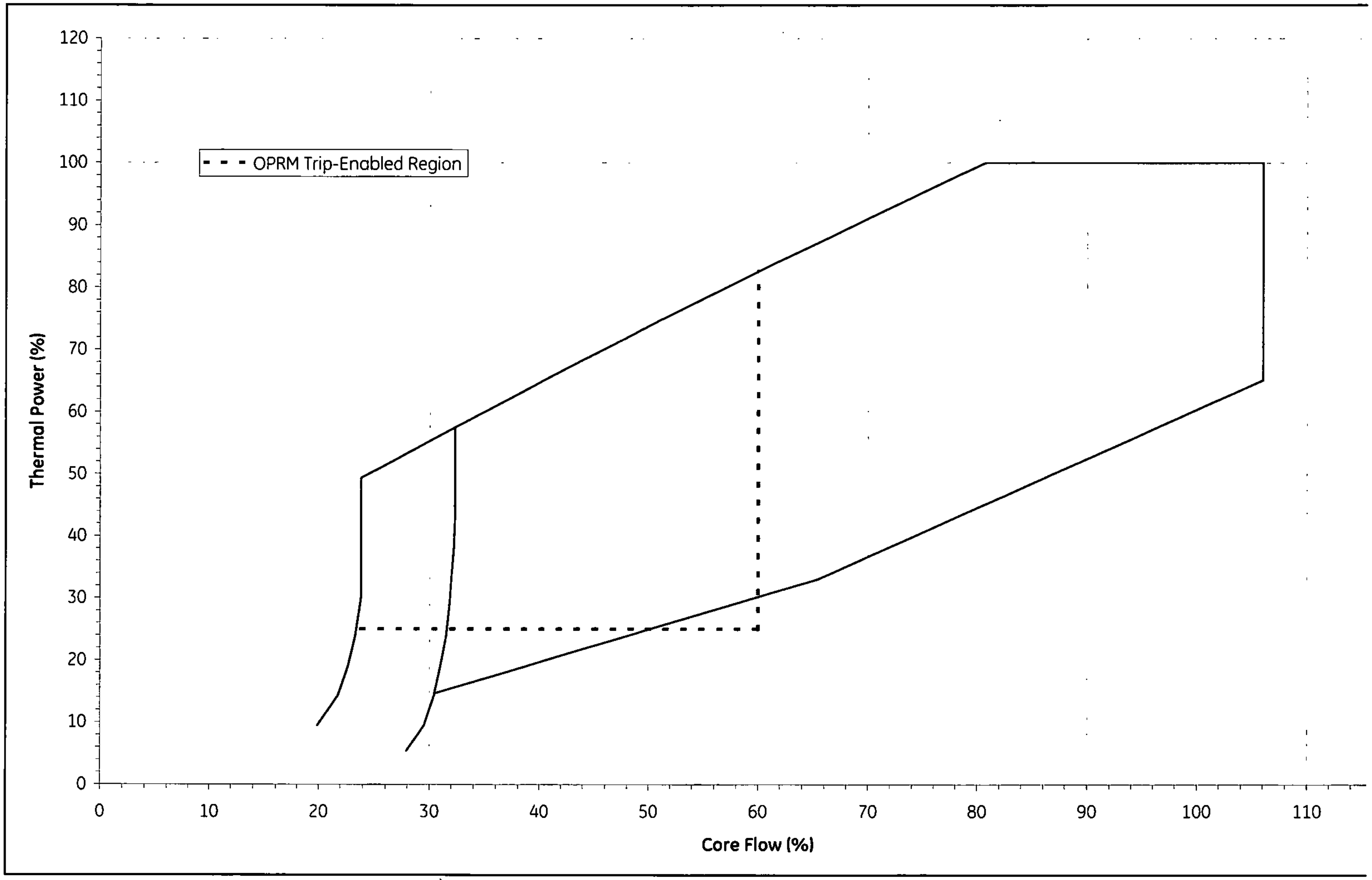


Figure 6-2 Demonstration of Proposed BSP Regions for NFWT

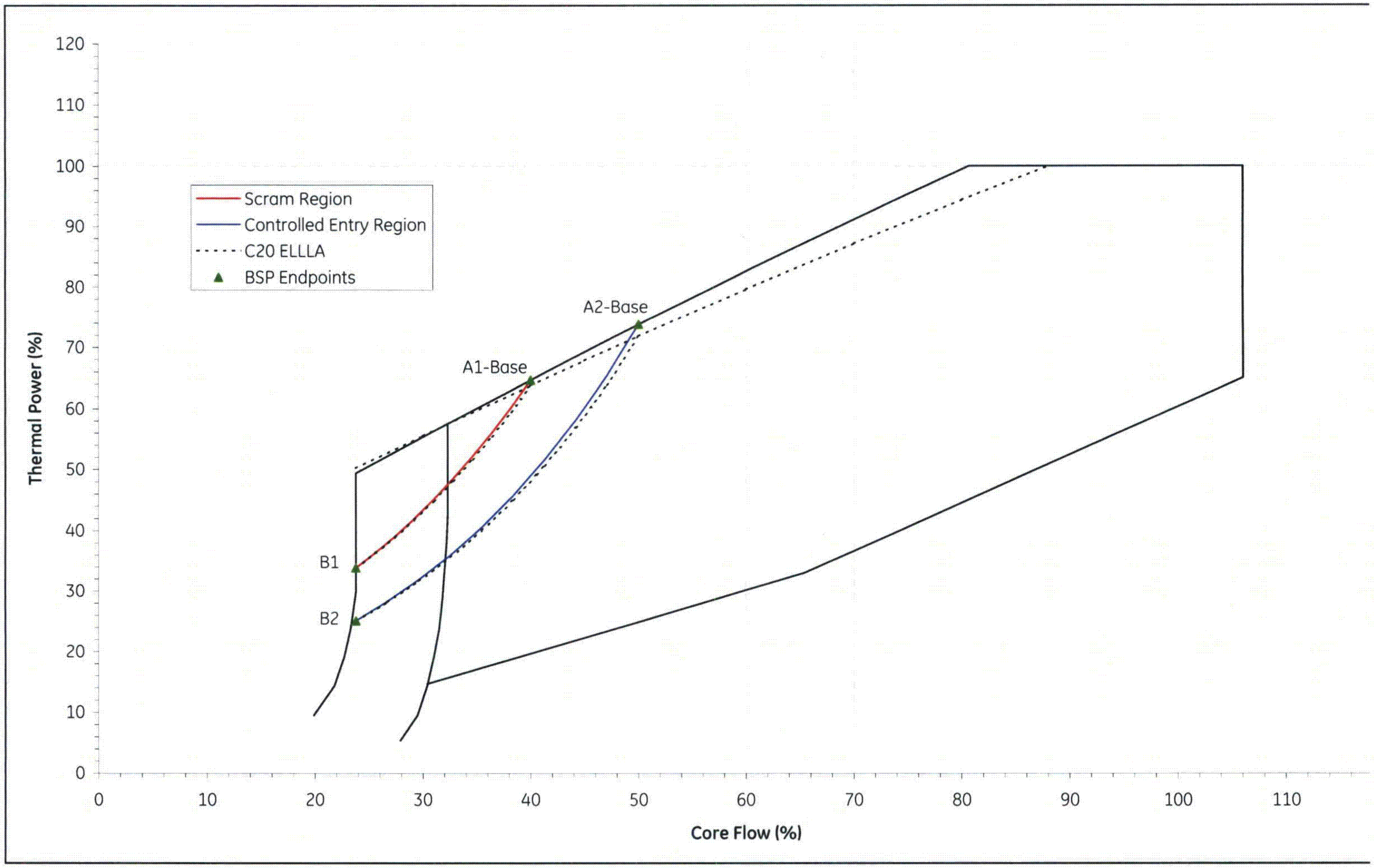
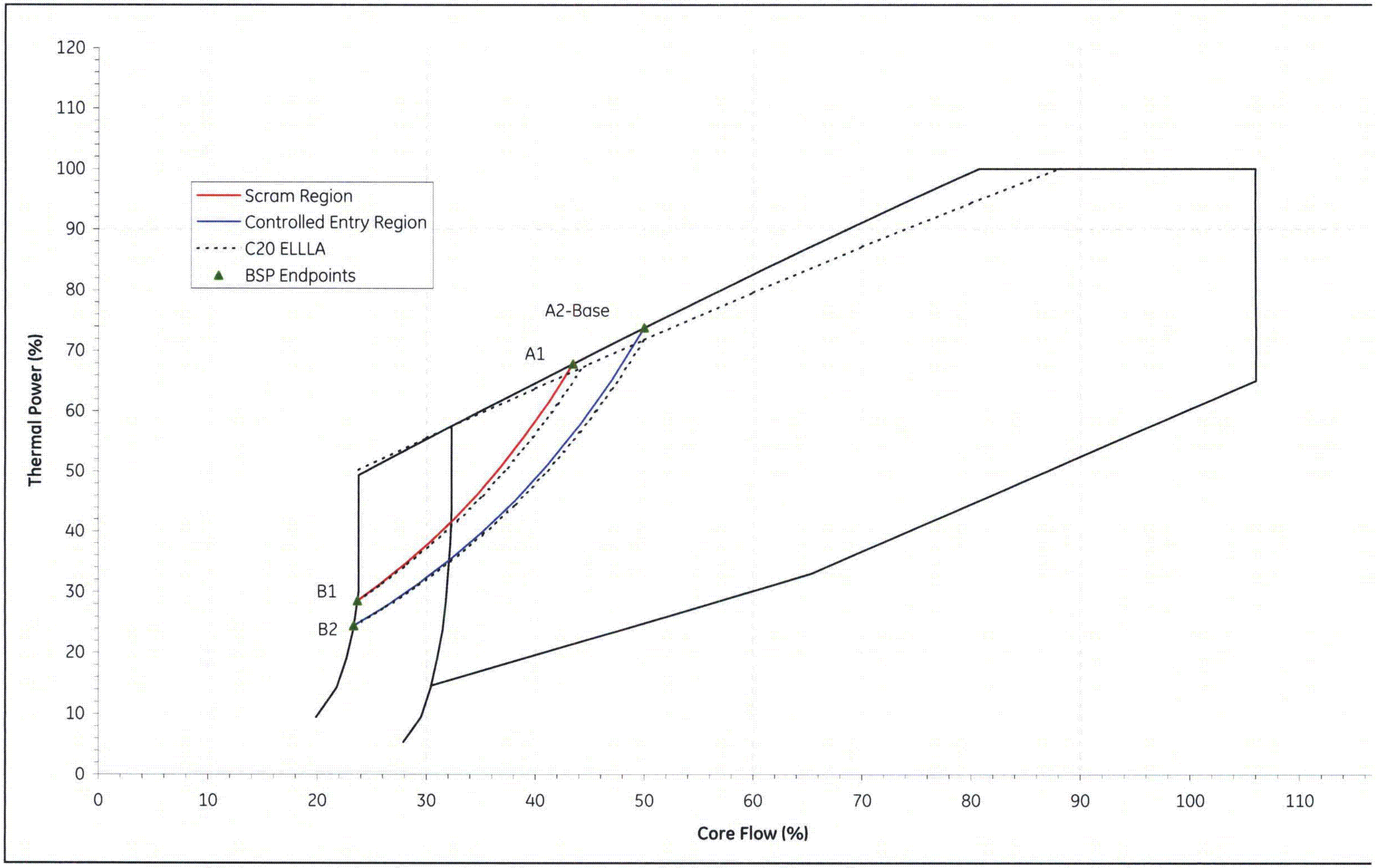


Figure 6-3 Demonstration of Proposed BSP Regions for RFWT



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 [[

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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

		CLTP			ARTS/MELLLA					
No	**Component**	**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

		CLTP				ARTS/MELLLA				
No	**Component**	**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 ≤ 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.

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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

Setpoint Characteristics:	Definition	Reference(s)
Event Protection:	Limiting event for the setpoint: The RBM is designed to prevent fuel damage during a Rod Withdrawal Error (RWE) event during high power operation.	Ref. 2, Section 3.19 Ref. 4 Bases B 3.3.2.1
Function After Earthquake	☐ Required ☒ Not Required	Ref. 2 Section 3.19.2 Ref. 4 Bases B3.3.2.1, Comment 9
Setpoint Direction: • Low Power Trip Setpoint (LTSP) • Intermediate Power Trip Setpoint (ITSP) • High Power Trip Setpoint (HTSP) • Low Power Setpoint (LPSP) • Intermediate Power Setpoint (IPSP) • High Power Setpoint (HPSP)	☒ Increasing ☐ Decreasing ☒ Increasing ☐ Decreasing ☒ Increasing ☐ Decreasing ☒ Increasing ☐ Decreasing ☒ Increasing ☐ Decreasing ☒ Increasing ☐ Decreasing	Ref. 4 Bases Section 3.3.2.1
Single or Multiple Channel	☒ Single ☐ Multiple	Ref. 4 Bases Section 3.3.2.1, Ref. 6.2 Section 4.1.6
LER Calculation Basis if Multiple Channel	Standard (Conservative) LER Calculation ☒, or Configuration Specific LER Calculation ☐	Ref. 1, Ref. 2
Trip Logic for Configuration Specific LER Calculation	n/a	

Plant Data:	Value	Sigma if not 2	Reference(s)
Power Primary Element (LPRM Detector) (% Power) $APEA_{Accuracy}$ $APEA_{PowerSupplyEffect}$ DPEA • Trip Setpoints • Power Setpoints	 ± 1%; bias 0.49% negligible negligible ± 0.2% / 7 days; bias 0.33% / 7 days		Ref. 2, Comment 6

1. Function: RBM Rod Withdrawal Blocks (cont'd)

Plant Data:	Value	Sigma if not 2	Reference(s)
Power Process Measurement Accuracy (PMA) • Tracking - Trip Setpoints • Tracking - Power Setpoints • Noise - Trip Setpoints • Noise - Power Setpoints	 ± 1% ± 1.11% ± 2.0% 0.0%	 3	

Components (or Devices) in Setpoint Function Instrument Loop:

- LPRM Detector
- NUMAC Chassis: Instrument Loop Power Electronics (LPRM, APRM, RBM, Trip Circuit)

1.1 RBM Low Power Trip Setpoint (LTSP)

Current Function Limits:	Value/Equation			Reference(s)	
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% Reference Level) unfiltered	filtered		
Analytical Limit	$0.58\ W_d + 38$	127.0	125.8		
Tech Spec Allowable Value	$0.58\ W_d + 35.0$	Results provided in Section 3			
Nominal Trip Setpoint	$0.58\ W_d + 32.0$	Results provided in Section 3			
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3	

1.2 RBM Intermediate Power Trip Setpoint (ITSP)

Current Function Limits:	Value/Equation			Reference(s)	
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% Reference Level) unfiltered	filtered		
Analytical Limit	$0.58\ W_d + 46$	122.0	121.0		
Tech Spec Allowable Value	$0.58\ W_d + 43.0$	Results provided in Section 3			
Nominal Trip Setpoint	$0.58\ W_d + 40.0$	Results provided in Section 3			
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3	

1.3 RBM High Power Trip Setpoint (HTSP)

Current Function Limits:	Value/Equation			Reference(s)	
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% Reference Level) unfiltered	filtered		
Analytical Limit	$0.58\ W_d + 54$	117.0	116.0		
Tech Spec Allowable Value	$0.58\ W_d + 51.0$	Results provided in Section 3			
Nominal Trip Setpoint	$0.58\ W_d + 48$	Results provided in Section 3			
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3	

1.4 RBM Low Power Setpoint (LPSP)

Current Function Limits:	Value/Equation			Reference(s)
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% RTP) unfiltered	filtered	
Analytical Limit	n/a	30	30	
Tech Spec Allowable Value	n/a	Results provided in Section 3		
Nominal Trip Setpoint	n/a	Results provided in Section 3		
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3

1.5 RBM Intermediate Power Setpoint (IPSP)

Current Function Limits:	Value/Equation			Reference(s)
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% RTP) unfiltered	filtered	
Analytical Limit	n/a	65	65	
Tech Spec Allowable Value	n/a	Results provided in Section 3		
Nominal Trip Setpoint	n/a	Results provided in Section 3		
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3

1.6 RBM High Power Setpoint (HPSP)

Current Function Limits:	Value/Equation			Reference(s)
	Present Value/Equation (% RTP)	ARTS-MELLLA Condition (% RTP) unfiltered	filtered	
Analytical Limit	n/a	85	85	
Tech Spec Allowable Value	n/a	Results provided in Section 3		
Nominal Trip Setpoint	n/a	Results provided in Section 3		
Operational Limit	n/a	n/a		Ref. 1, Ref. 2, Comment 3

2. Components:

2.1 Power Electronics (LPRM, APRM, RBM, Trip Circuit)

Component Information:	Value/Equation	Reference(s)
Plant Instrument ID No.	Undefined	Comment 2
Instrument vendor	GE / Reuter-Stokes	Ref. 6.1
Model ID No. (including Range Code)	NUMAC	Ref. 6.1
Plant Location(s)	Control Bldg	Ref. 6.1 Section 4.2.1 and Appendix C
Process Element	LPRMs: NA250/NA300	Ref. 6.2 Sections 1.5 & 3.2,

Inputs:

Vendor Specifications	Value / Equation	Sigma if not 2	Reference(s)
Top of Scale	FS = 125%	n/a	Ref. 6.2 Sections 4.3.2 & 4.7.2
Bottom of Scale	0%	n/a	Ref. 6.2 Sections 4.3.2 & 4.7.2
Upper Range Limit	n/a	n/a	Ref. 6.2 Sections 4.3.2 & 4.7.2
Accuracy • LPRM Detector • LPRM Electronics	See Section 1 ± 0.943% (% local power)		Ref. 1 & Ref. 2
Temperature Effect	included in accuracy		
Seismic Effect	included in accuracy		Ref. 6.4 Section 4.1.1, Comment 4
Radiation Effect	included in accuracy		Ref. 6.1 Section 5.2, Comment 4
Humidity Effect	included in accuracy		Ref 6.1 Section 5.2, Comment 4
Power Supply Effect (Detector)	See Section 1		
RFI/EMI Effect	included in accuracy		Ref. 6.4 Sections 4.1.1, and 4.2.5,

			Comment 4
Insulation Resistance Effect	Negligible		Comment 4
Over-pressure Effect	n/a		Comment 5
Static Pressure Effect	n/a		Comment 5

2.1 Power Electronics (LPRM, APRM, RBM, Trip Circuit) (cont'd)

Plant Data:	**Value**	**Reference(s)**
Calibration Temp Range	70 to 104 ºF	Ref. 3.2 Sec. 6.5, Comment 15
Normal Temperature Range	40 to 104 ºF	Ref. 3.2 Sec. 6.5
Trip Temp Range	40 to 104 ºF	Ref. 3.2 Sec. 6.5, Comment 16
Humidity Operating Range	10 to 60% RH	Ref. 3.2 Sec. 6.5
Plant Radiation value	n/a	Ref. 3.2 Sec. 6.5
Plant seismic value	0.7g	Ref. 3.2 Section 6.5, Comment 9
Power Supply Variation value	Negligible	Comment 4
RFI/EMI value	n/a	Comment 4
Over-pressure value	n/a	Comment 5
Static Pressure value	n/a	Comment 5

Drift:	**Value**		**Sigma if not 2**	**Reference(s)**
Current Calib. Interval	7 days	☐Includes extra 25%		Ref. 4 Table 3.3.1.1-1 SR 3.3.1.1.2
Desired Calib. Interval	7 days	☐Includes extra 25%		Ref. 4 Table 3.3.1.1-1 SR 3.3.1.1.3
Drift Source	☒Vendor Trip Setpts	☒Calculated Power Setpts		Ref. 1, Ref. 2
Drift Value (Trip Setpoints)	± 0.3% FS / 4 hours (% RBM power)			Ref. 6.2 Section 4.7.2.9, Comment 8
Drift Value (Power Setpoints) (% power)	± 0.5% FS / 700 hours	± 0.5% SP / 8.75 days		Ref. 6.3 Section 4.3.3.3.5, Comment 14

2.1 Power Electronics (LPRM, APRM, RBM, Trip Circuit) (cont'd)

Calibration:	**Value / equation**	**Sigma if not 3**	**Reference(s)**
	Included in APRM calibration		
As Left Tolerance	Trip setpoints: 0 Power setpoints: AGAF		Comment 7
Leave Alone Tolerance	Trip setpoints: = ALT Power setpoints: = ALT		Comment 7
Input Calibration Tool:	n/a		Comment 7
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			
Input Calibration Standard:	n/a		Comment 7
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			

Output Calibration Tool:	n/a		Comment 7
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			
Output Calibration Standard:	n/a		Comment 7
Accuracy			
Resolution / Readability			
Minor Division			
Upper Range			
Temperature Effect			

Application Specific Input:	**Value**	**Sigma if not 2**	**Reference(s)**
Minimum no. of LPRMs per RBM Channel (Trip Setpoints)	4 of 8		Ref. 6.2 Section 4.7.9.4.2, 4.8.2
Minimum no. of LPRMs per APRM Channel (Power Setpoints)	20 of 43		Ref. 6.1 Section 4.1.5, and Section 3.1.1. Comment 13
APRM Gain Adjustment Factor (AGAF)	± 2% RTP	3	

3. Summary Results:

Calculated Values

Setpoint Function	Analytic Limit (from Section 1) (%RTP)		Allowable Value (%RTP)		Nominal Trip Setpoint (%RTP)		Meets LER Avoid-ance Criteria	Meets Spurious Trip Avoid-ance Criteria
	Unfiltered	Filtered	Unfiltered	Filtered	Unfiltered	Filtered		
Low Power Setpoint (LPSP)	30	30	28.0	28.0	26.0	26.0	Y	n/a
Intermediate Power Setpoint (IPSP)	65	65	63.0	63.0	61.0	61.0	Y	n/a
High Power Setpoint (HPSP)	85	85	83.0	83.0	81.0	81.0	Y	n/a

Setpoint Function	Analytic Limit (from Section 1) (% Reference Level)		Allowable Value (% Reference Level)		Nominal Trip Setpoint (% Reference Level)		Meets LER Avoid-ance Criteria	Meets Spurious Trip Avoid-ance Criteria
	Unfiltered	Filtered	Unfiltered	Filtered	Unfiltered	Filtered		
Low Power Trip Setpoint (LTSP)	127.0	125.8	124.6	123.4	124.2	123.0	Y	n/a
Intermediate Power Trip Setpoint (ITSP)	122.0	121.0	119.6	118.6	119.2	118.2	Y	n/a
High Power Trip Setpoint (HTSP)	117.0	116.0	114.6	113.6	114.2	113.2	Y	n/a

Application Specific Setpoint Adjustments

Setpoint Function				
Low Power Setpoint (LPSP) - setting adjustment	NTSP 26.0% RTP (from above)	Deadband 1.1% RTP	Actual Instrument Setting 24.9% RTP	Ref. 6.2 Section 4.7.9.1.2 and Section 4.8.2.4, Comment 10

4. Comments and Recommendations:

1. Unless specifically identified as "bias" errors in this document, all instrument uncertainty errors will be considered to be random in nature, even when the "±" symbol is not shown.
2. Some plant specific information has not been provided or is not currently available in the current plant setpoint document, but is considered unnecessary because the effects of this information are included within the instrument accuracy values or are not necessary for setpoint evaluation.
3. STA evaluations are not performed for rod blocks or permissives per GEH setpoint methodology (Reference 1 and Reference 2), such as the RBM Rod Blocks. Therefore, the Operational Limits are not applicable.
4. Seismic effect, radiation effect, humidity effect, power supply effect, Radio Frequency Interference/ Electromagnetic Interference (RFI/EMI) effect, and insulation resistance effect errors are marked "negligible" or "included in accuracy" and are considered to have negligible impact on the manufacturer's accuracy terms when they are not identified separately.
5. Per Reference 1 and Reference 2, overpressure effects are only applicable to pressure measurement devices (e.g., differential pressure transmitters), and static pressure effects are only applicable to differential pressure measurement devices. These effects are marked "n/a" for other devices.
6. [[ ]] (Reference 2 Section 4.5.3)
7. The APRM subsystem is calibrated on-line weekly (Reference 4, SR 3.3.1.1) using the AGAF process, where the gain of the APRMs is adjusted to read the Core Thermal Power (CTP) determined by the Process Computer (P/C), within a specified As Left Tolerance. [[ ]] Thus, the only calibration error to consider for the APRM electronics sub-loop is the As Left Tolerance specified by the AGAF process.
8. The Power Electronics Drift for the RBM Trip setpoints uses the 4-hour drift error specification. The only drift error would be the drift in the several hours after control rod selection and nulling, and before the control rod is motion. This is estimated to be a few hours, so the 4-hour drift interval is used.
9. The RBM Rod Block limits control rod withdrawal if localized neutron flux exceeds a pre-determined setpoint during control rod manipulations. However, the RBM system is not essential for the safety of the plant. Hence, the RBM rod withdrawal block setpoint does not perform a protective function. Therefore, the Seismic Effect for the RBM does not need to be considered.
10. As described in the Technical Specifications (Reference 4 Section 3.3.2.1), the LPSP is considered as an automatic "enable" feature when thermal power is above the LPSP, and the AV and NTSP are calculated accordingly. The enable feature occurs as Reactor power increases past the LPSP. The vendor documents for the RBM equipment treat the LPSP as an automatic "bypass" feature (Reference 6.2, Section 4.8.2.4) when below the LPSP. The bypass feature occurs as Reactor Power decreases below the LPSP. These two descriptions are not interchangeable/equivalent; there is a need in the equipment logic for an instrument setting "deadband". Therefore, the equipment instrument setting for the LPSP NTSP must include the 1.1% Rated Thermal power deadband (i.e., hysteresis of 1.0% and an accuracy of 0.1%). The deadband does not apply to the AV. The equipment instrument setting is equal to the NTSP for the other RBM setpoint functions.

4. Comments and Recommendations (cont'd):

11. For the RBM Downscale Trip Setpoint (DTSP), no credit is taken for it in the RWE analyses. Choice of this setpoint is an operational issue to be decided by the plant. There is no AL for this setpoint. A value of 95% is recommended, but it can be lowered if operational problems are encountered.
12. Per Reference 1 and Reference 2, the difference between the AL and AV and the difference between the AL and NTSP are independent of the number for the AL. This applies for all of the Power and Trip setpoint functions.
13. Reference 6.1 specifies that up to 23 LPRMs per APRM channel can be bypassed. Based on a total of 43 LPRMs, this is the basis for a minimum of 20 LPRMs per LPRM channel.
14. A conservative value for the design drift value of ± 0.5%SP/8.75 days is applied based on the equipment surveillance interval of 7 days plus 25%.
15. Calibration temperature range is conservatively chosen to be between 70 and 104 °F. Reference 3.2 provides a calibration temperature of 70 °F; a max temperature of 104 °F, which corresponds to the maximum normal temperature is assumed.
16. The Trip temperature range was chosen to be between 40 to 104 °F, which is equal to the normal temperature range. This is because the RBM is used for transient states, and not for accident trips. The temperature range is expected to be normal when the trip is required.
17. Transfer functions used in this calculation:

RBM Power Electronics:	Output is proportional to the average of the inputs, and multiplied by a gain adjustment, calculated relative to a constant arbitrary reference equivalent to 100% RTP.
APRM Power Electronics:	Output is proportional to the average of the inputs.

5. References:

1. NEDC-32889P, Rev. 3, "General Electric Methodology for Instrumentation Technical Specification and Setpoint Analysis", November 2002.

2. NEDC-31336P-A, Class 3, "General Electric Instrument Setpoint Methodology", September 1996.

3. Current applicable CGS setpoint calculations:

   3.1. Deleted.

   3.2. Energy Northwest setpoint calculation E/I-02-93-1286, "APRM–CH–A" dated 6/20/2003.

4. Columbia Generating Station Technical Specifications and Bases, as revised through Amendment 212.

5. Deleted.

6. Vendor Specifications:

   6.1. GE 24A5221TC, "PRNM Requirements Specification", Data Sheet, Rev. 4, 12/7/2009.

   6.2. GE 24A5221, "NUMAC Power Range Neutron Monitor (PRNM)", Requirements Specification, Rev. 17, 7/21/2008.

   6.3. GE 25A5916, "NUMAC Average Power Range Monitor (APRM)", Performance Specification, Rev. 5, 2/28/2005.

   6.4. GE 23A5082, "NUMAC Requirements Specification", Design Specification, Rev. 1, 8/9/1995.