ATTACHMENT (5)

NEDO-33286, REV. 0

APRM/RBM/TECHNICAL SPECIFICATIONS/MAXIMUM EXTENDED LOAD LINE LIMIT ANALYSIS (ARTS/MELLLA) NON-PROPRIETARY VERSION

GE Energy
Nuclear

3901 Castle Hayne Rd
Wilmington, NC 28401

NEDO-33286
Revision 0
Class I
DRF 0000-0056-7605
March 2007

Nine Mile Point Nuclear Station Unit 2

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

NON-PROPRIETARY NOTICE

This is a non-proprietary version of the document NEDC-33286P Revision 0, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between Constellation Energy Nine Mile Point, LLC and GE, Purchase Order 7702225 Rev 0, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than Constellation Generation Group, or for any purpose other than that for which it is intended, is not authorized; and, with respect to any unauthorized use, GE makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

TABLE OF CONTENTS

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

10.4 Conclusions 10-3
11.0 STEAM DRYER AND SEPARATOR PERFORMANCE 11-1
12.0 HIGH ENERGY LINE BREAK 12-1
13.0 TESTING 13-1
14.0 REFERENCES 14-1
ATTACHMENT A A-1

LIST OF TABLES

Table 1-1 Computer Codes Used for ARTS/MELLLA Analyses* 1-9
Table 2-1 Analyses Presented In This Report 2-2
Table 2-2 Applicability of Analyses 2-2
Table 3-1 Base Conditions for ARTS/MELLLA Rated Transient Analyses 3-8
Table 3-2 Base Conditions for ARTS/MELLLA Off-rated Transient Analyses – Normal Feedwater Temperature and Reduced Feedwater Temperature 3-8
Table 3-3 MELLLA Transient Analysis Results at RTP Conditions, Cycle 11 3-9
Table 3-4 ARTS Transient Analysis Results – Generic K(P) Confirmation Above Pbypass 3-10
Table 3-5 ARTS Transient Analysis Results – MCPR(P) Below Pbypass 3-12
Table 3-6 ARTS Transient Analysis Results – Generic LHGRFAC(P) Confirmation Above Pbypass 3-12
Table 3-7 ARTS Transient Analysis Results – LHGRFAC(P) Below Pbypass 3-13
Table 4-1 Rod Block Monitor System Improvements 4-12
Table 4-2 Rod Withdrawal Error Analysis Results 4-13
Table 4-3 RWE Analysis Results For Peripheral Rod Groups (108% Setpoint) 4-14
Table 4-4 RBM Signal Filter Setpoint Adjustment 4-14
Table 4-5 RBM System Setup 4-15
Table 4-6 RBM Setup Setpoint Definitions 4-16
Table 5-1 NMP2 Cycle 11 Sensitivity of Overpressure Analysis Results to Initial Flow 5-1
Table 6-1 Option III Setpoint Demonstration 6-3
Table 7-1 ECCS-LOCA Analysis Bases for NMP2 ARTS/MELLLA 7-3
Table 7-2 ECCS-LOCA Peak Cladding Temperature for NMP2 ARTS/MELLLA 7-4
Table 8-1 Cases Analyzed For Short-Term Containment Response 8-7
Table 8-2 Summary of Sensitivity Study Results for Peak Drywell Pressure and Temperature and Initial Drywell Pressurization Rate 8-7
Table 9-1 Flow-Induced Loads on Shroud and Jet Pumps for NMP2 9-6
Table 9-2 Maximum Acoustic Loads on Shroud and Jet Pumps 9-6
Table 9-3 Maximum Acoustic Loads on Shroud Support (MELLLA) 9-6
Table 9-4 RPV Internals Structural Evaluation Results 9-7
Table 10-1 Operating Conditions and Equipment Performance Characteristics for ATWS Analyses 10-4
Table 10-2 Summary of ATWS Calculation Results 10-4

LIST OF FIGURES

Figure 1-1 MELLLA Operating Range Power/Flow Map 1-11
Figure 3-1 Power-Dependent MCPR Limits, MCPR(P) / K(P) 3-14
Figure 3-2 Power-Dependent LHGR Multiplier, LHGRFAC(P) 3-16
Figure 3-3 Flow-Dependent MCPR Limits, MCPR(F) 3-18
Figure 3-4 Flow-Dependent LHGR Multiplier, LHGRFAC(F) 3-19
Figure 4-1 Conceptual Illustration of Current Flow-Dependent RBM System with AC/BD LPRM Assignment 4-17
Figure 4-2 RBM Current AC/BD LPRM Assignment 4-18
Figure 4-3 Current RBM System Configuration Limits (Typical for 106% Setpoint) 4-19
Figure 4-4 New Power-Dependent RBM System with $BCCD_1/BCCD_2$ LPRM Assignment 4-20
Figure 4-5 New RBM $BCCD_1/BCCD_2$ LPRM Assignment 4-21
Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs) 4-22
Figure 4-7 New RBM System Core Power Limit (Typical) 4-23
Figure 4-8 Design Basis RWE MCPR Requirement Versus RBM Setpoint 4-24
Figure 4-9 Design Basis MCPR Requirement for RWE (ARTS) 4-25
Figure 4-10 RBM Setpoint Versus Power (without Filter) 4-26
Figure 4-11 NMP2 Neutron Monitoring System 4-27
Figure 4-12 Rod Block Monitor Rod Group Geometries 4-28
Figure 4-13 Results of LPRM Failure Rate Sensitivity Studies 4-29
Figure 4-14 Power-Dependent RBM Trip Nomenclature 4-30
Figure 6-1 MELLLA OPRM Trip Enabled Region 6-4
Figure 6-2 Demonstration of Proposed BSP Regions 6-5

ACRONYMS

Term	Definition
ΔCPR	Delta Critical Power Ratio
ΔW	Difference between two loop and single loop effective drive flow at the same core flow
ABA	Amplitude Based Algorithm
ADS	Automatic Depressurization System
AL	Analytical Limit
A/M	ARTS/MELLLA
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
ARTS	APRM/RBM/Technical Specifications
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
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 the Oscillation Magnitude
DPEA	Time Dependent part of Primary Element Accuracy
DS/RV	Dual Safety / Relief Valve
DTPF	Design Total Peaking Factor
ECCS	Emergency Core Cooling System
ELLLA	Extended Load Line Limit Analysis
EOC	End-of-Cycle

Term	Definition
FCV	Flow Control Valve
FLE	Fuel Loading Error
FIV	Flow-Induced Vibration
FWCF	Feedwater Controller Failure
FFWTR	Final Feedwater Temperature Reduction
FWTR	Feedwater Temperature Reduction
FWHOOS	Feedwater Heater Out-of-Service
FWLB	Feedwater Line Break
FRTP	Fraction of Rated Thermal Power
FWTR	Feedwater Temperature Reduction
GE	General Electric
GESTR	GE Stress and Thermal Analysis of Fuel Rods
GEXL	GE Critical Boiling Length
GRBA	Growth Rate Based Algorithm
GSF	Generic Shape Function
HELB	High Energy Line Break
HFCL	High Flow Control Line
HPCS	High Pressure Core Spray
HPCSDG	High Pressure Core Spray Diesel Generator
IBA	Intermediate Break Accident
ICA	Interim Corrective Action
ICF	Increased Core Flow
ICGT	Incore Guide Tube
ICPR	Initial Critical Power Ratio
IORV	Inadvertent Opening of a Relief Valve
IRLS	Idle Recirculation Loop Start-up
ISA	Instrumentation, Systems, and Automation Society
JR	Jet Reaction
LFWH	Loss of Feedwater Heating
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
LRNBP	Load Rejection with No Bypass
LSSS	Limiting Safety System Settings

Term	Definition
MCHFR	Minimum Critical Heat Flux Ratio
MCPR	Minimum Critical Power Ratio
MELLLA	Maximum Extended Load Line Limit Analysis
MFCV	Minimum Flow Control Valve
MFLPD	Maximum Fraction of Limiting Power Density
MLHGR	Maximum Linear Heat Generation Rate
MOP	Mechanical Over-Power
MSIV	Main Steam Line Isolation Valve
MSIVC	Main Steam Line Isolation Valve Closure
MSIVF	Main Steamline Isolation Valve Closure with a Flux Scram
MSLB	Main Steam Line Break
NMP2	Nine Mile Point Nuclear Station Unit 2
N/R	Not Reported
NCL	Natural Circulation Line
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NTSP	Nominal Trip Setpoint
NUMAC™	Nuclear Measurement Analysis and Control
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OOS	Out-of-Service
OPRM	Oscillation Power Range Monitor
PBDA	Period Based Detection Algorithm
PCT	Peak Cladding Temperature
P/F	Power / Flow as in Power/Flow Map
PLHGR	Peak Linear Heat Generation Rate
PLU	Power Load Unbalance
PMA	Process Measurement Accuracy
PRNM	Power Range Neutron Monitor
PRNMS	Power Range Neutron Monitoring System
PRFO	Pressure Regulator Failure Open
PS	Pool Swell
RBM	Rod Block Monitor
RCF	Rated Core Flow
RCIC	Reactor Core Isolation Cooling (System)
RDLB	Recirculation Discharge Line Break
RFI	Recirculation Flow Increase

Term	Definition
RG	Regulatory Guide
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 (System)
RWE	Rod Withdrawal Error
SBA	Small Break Accident
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
SRV	Safety-Relief Valve (for NMP2 this is the same as DS/RV)
STP	Simulated Thermal Power
TBPOOS	Turbine Bypass Out-of-Service
TCV	Turbine Control Valve
TLO	Two Loop Operation
TOP	Thermal Over-Power
TS	Technical Specification
TTNBP	Turbine Trip with No Bypass
USAR	Updated Safety Analysis Report
VPF	Vane Passing Frequency
V&V	Verification and Validation
W_T	% 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 Nine Mile Point Nuclear Station Unit 2 (NMP2) 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 modified to include the operating region bounded by the rod line which passes through the 100% of current licensed thermal power (CLTP) / 80% 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 Analyses (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 NMP2 in the region above the rated rod line.

1.1 Background

NMP2 has performed a Stretch Power Uprate, which increased the CLTP to 3467 MWt or 104.3% 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 NMP2 operation at 3467 MWt.

NMP2 originally included minimum critical heat flux ratio (MCHFR) as the thermal margin criterion. This MCHFR basis included operating, overpower, and safety limit values that along with a design power peaking factor, translate to the rated power load line, 108% load line, and 120% load line, respectively (thus, the APRM flow-biased rod block and scram protection functions). Therefore, these APRM flow-biased setpoint values originate with a deterministic overpower analysis. Later, with the change to the MCPR thermal margin basis under which NMP2 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 NMP2 Updated Safety Analysis Report (USAR) 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 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: 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: Hope Creek, LaSalle Units 1 and 2, Dresden Units 2 and 3, Quad Cities Units 1 and 2, and Vermont Yankee. Susquehanna plant has a full ARTS submittal and Fitzpatrick has a partial ARTS submittal currently under review with the NRC.

1.2 ARTS/MELLLA Bases

1.2.1 Analytical Bases

The power/flow operating map (Figure 1-1) includes operating domain changes for ARTS/MELLLA consistent with approved operating domain improvements for other BWRs. The NMP2 MELLLA operating domain is defined by the following upper boundary:

- The MELLLA boundary line, extended up to the existing maximum CLTP of 3467 MWt. The MELLLA boundary is defined as the line that passes through the 100% of CLTP / 80.0% of RCF state point.
- The CLTP of 3467 MWt.
- The currently analyzed ICF condition of 105.0% 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:

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

where: A = 22.191

B = 0.89714

C = -0.0011905

K = 1.158 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 power/flow point for Single Loop Operation (SLO) operation remains unchanged from its current absolute value of 2427 MWt (70% of RTP) for MELLLA. For NMP2, SLO is not extended into MELLLA region.

When compared to the current power/flow 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 NMP2 in the region above the ELLLA and up to the MELLLA boundary line. The scope of the analyses performed covers the initial application for NMP2 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 5.

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 the current Cycle 11 core design using GE14 and GE11 fuel (Reference 2). 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 3 were reviewed for the MELLLA region based on existing thermal analysis limits at plants similar to NMP2 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 / 105% of RCF). The operating limit is calculated on a cycle specific basis in accordance with Reference 5 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 NMP2 meets the licensing criteria in the MELLLA operating domain.

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

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

The APRM Flow-Biased Simulated Thermal Power (STP) scram line is conservatively not credited in any NMP2 safety analyses. In addition, the APRM Flow-Biased STP rod block line is conservatively not credited in any NMP2 safety analyses, although it is part of the NMP2 design configuration. This section discusses the setpoint changes for these systems for operational flexibility purposes and provides the inputs to the NMP2 TS changes.

For the current, ELLLA operating domain, power/flow map, the APRM Flow-Biased STP scram line analytical limit (AL) for two loop operation (TLO) is defined as: 0.58 Wd + 65%, and for SLO, 0.58 (Wd – 5) + 65%, of RTP. The APRM Flow-Biased STP Scram clamp is at 118% 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 line is currently set at: for TLO, 0.58 Wd + 59%, and for SLO, 0.58 (Wd – 5) + 59% of RTP. NMP2 does not have an APRM Flow-Biased STP Rod Block clamp. A Rod Block clamp setpoint of 112% will be implemented for ARTS-MELLLA corresponding to the current maximum value at 100% flow.

With the current power/flow map, the operational margin between the APRM Flow-Biased STP rod block line and the ELLLA Boundary line is significantly reduced, in comparison to the operational margin originally available with respect to the 100% rod line. With the proposed MELLLA power/flow map expansion, the upper boundary of the licensed operating domain is extended to approximately the 115.8% rod line. To accommodate this expanded operating domain and to restore the original margin between the MELLLA boundary line and the APRM Flow-Biased STP rod block line, the following ALs are redefined:

Analytical Limit		TLO	SLO
APRM Flow Biased STP High Scram	Flow-Biased Equation *	0.64(Wd -ΔW) + 66.8% = 0.64 Wd + 66.8%	0.58(Wd - ΔW) + 65% = 0.58 Wd + 62.1%
	Flow-Biased Clamp	No change	No change
APRM Flow Biased STP Rod Block	Flow-Biased Equation *	0.64(Wd - ΔW) + 60.8% = 0.64 Wd + 60.8%	0.58(Wd - ΔW) + 59% = 0.58 Wd + 56.1%
	Flow-Biased Clamp	112%	112%

* 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 following allowable values (AVs) are determined from the above changed ALs:

Allowable Value		TLO	SLO
APRM Flow Biased STP High Scram	Flow-Biased Equation	0.64(W_d - ΔW) + 63.8% = 0.64 Wd + 63.8%	0.58(W_d - ΔW) + 62% = 0.58 W_d + 59.1%
	Flow-Biased Clamp	No change	No change
APRM Flow Biased STP Rod Block	Flow-Biased Equation	0.64(W_d - ΔW) + 54.8% = 0.64 W_d + 54.8%	0.58(W_d - ΔW) + 53% = 0.58 W_d + 50.1%
	Flow-Biased Clamp	109.5%	109.5%

The RBM Upscale Flow-Biased rod block line limits are currently set at: the TS AVs of: TLO, 0.66 W_d + 47% and SLO, 0.66 W_d + 43.7%, of RTP; AL values are: TLO, 0.66 W_d + 50% and SLO, 0.66 W_d + 46.7%, of RTP. ARTS change the form of the RBM from a flow-biased to a power-biased function. In Section 4.3, Rod Withdrawal Error Analysis, the evaluation of the RWE event was performed taking credit for the mitigating effect of the power-dependent RBM. The power-dependent RBM ALs and AVs are presented in Table 4-5.

The AV revisions were performed using the GE instrument setpoint methodology (NEDC 31336P-A, September 1996). Attachment A provides the GE setpoint calculation for the power-based RBM setpoint function.

The RBM trip setpoints are determined by use of NRC approved setpoint methodology. Using the GE setpoint methodology based on Instrumentation, Systems, and Automation Society (ISA) setpoint calculation method 2, the RBM Allowable Values are determined from the Analytical Limit, 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 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 since 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. Since the trip setpoint is a numerical value stored in the digital hardware and not subject to drift, the as-found and as-left tolerance values for the setpoint are the same as the setpoint (i.e. there is no tolerance band). The surveillance procedures also perform a channel functional test, which assures the RBM is functioning properly.

The suggested notes in Regulatory Issue Summary (RIS) 2006-17 (Reference 36), which are intended to assure that the trip setpoints are verified to be within predefined limits, so appropriate actions can be taken if found to be outside the predetermined limits, cannot be applied to the RBM. The suggested notes cannot be applied to the RBM since the trip setpoints are keyed into the RBM module via a keyboard and are displayed and stored as digital values in computer memory. As such, the RBM trip setpoint is therefore not subject to drift, is not calibrated in the traditional sense, and does not have as-found and as-left tolerance bands. The calibration of the RBM verifies the trip setpoint is as it was set. No range of values is acceptable, only the exact keyed in values, thus assuring that the NMP2 RBM trip will be within safety limits and that the limiting condition for operation will be met.

The RBM does have an LSSS due to RWE analysis. However, it is exempt from the requirements of RIS 2006-17, based on this discussion.

The APRM Flow-Biased instrumentation addressed in this report does not have Limiting Safety System Settings (LSSS) and is therefore not impacted by RIS 2006-17.

1.3 Average Power Range Monitor Improvements

The functions of the APRM are integrated within the NUMAC™ Power Range Neutron Monitoring system (PRNMS) (Reference 4). The safety related functions 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 power/flow map.
3. Provide an indication of the core average power level of the reactor in the power range.

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

NMP2 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 Fraction of Rated Thermal Power (FRTP) is greater than 1, the flow-referenced APRM trips must be lowered (setdown) or the APRM gain must be increased (NMP2 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 37.

The NMP2 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 5.
3. Peak cladding temperature and maximum cladding oxidation fraction following a LOCA shall remain within the limits defined in 10 CFR 50.46.

The safety analyses used to evaluate the OLMCPR are documented in Section 3.0 of this report. These analyses ensure that the SLMCPR and the fuel thermal-mechanical design bases are satisfied. These analyses also establish the power-dependent and flow-dependent MCPR and LHGR curves for NMP2. 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 safety limit MCPR during withdrawal of a single control rod.
2. Providing a signal to permit operator evaluation of the change in local relative power during the movement of a single control rod.

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

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 (3) Y	NEDE-24011P Rev. 0 SER NEDE-30130-P-A (2) NEDC-32992P NEDO-32465-A NEDO-32465-A
Reactor Internal Pressure Differences	LAMB TRACG ISCOR	07 02 09	(4) (5) Y (1)	NEDE-20566P-A 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 9 09 03	Y Y Y (1) Y	NEDE-30130-P-A (6) NEDE-24154P-A NEDC-24154P-A, Vol 4, NEDE-24011-P Rev 0 SER NEDC-32084P-A, Rev 2
Containment System Response	ISCOR M3CPT LAMB	09 05 08	Y(1) Y (4)	NEDE-24011-P Rev 0 SER NUREG-0661 NEDE-20566P-A
Annulus Pressurization Loads	ISCOR	09	Y (1)	NEDE-24011-P Rev. 0 SER
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 ISCOR TASC	11 09 (11) 04 09 03	Y Y (10) Y (1) Y	NEDE-30130-P-A (6) NEDC-24154P-A, Vol 4, Sup 1 NEDE-24011-P Rev 0 SER NEDC-32084P-A Rev 2

* The application of these codes to the ARTS/MELLLA analyses complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. The application of the codes also complies with the SERs for the extended power uprate programs.

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 since 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.

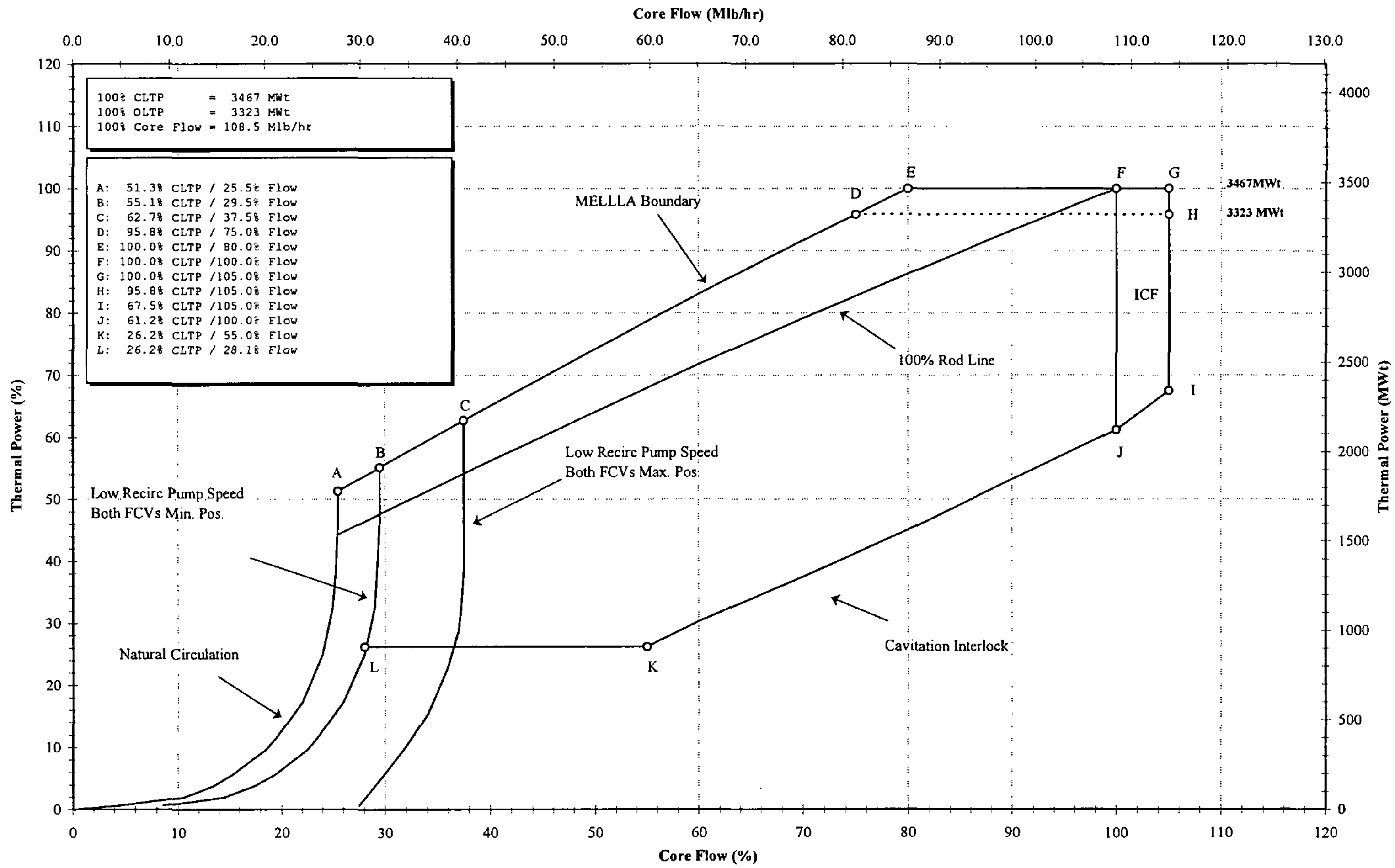


Figure 1-1 MELLLA Operating Range Power/Flow Map

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 NMP2 Cycle 11 core design was utilized. These tasks will be revalidated as part of the subsequent cycle-specific reload licensing analyses in accordance with Reference 5. The remainder of the ARTS/MELLLA scope of work is applicable to NMP2, unless there is a plant configuration change that affects the analysis.

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

Table 2-1 Analyses Presented In This Report

Section	Item	Result
3.0	Fuel Thermal Limits	Acceptable - Bounded by Limits Presented in Section 3.0
4.0	Rod Block Monitor System Improvement	Acceptable for Cycle 11 Core
5.0	Vessel Overpressure Protection	Acceptable - Below ASME Limit
6.0	Thermal-Hydraulic Stability	Acceptable for Cycle 11 Core
7.0	LOCA Analysis	Acceptable for Cycle 11 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	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 and plant specific adjustment based on LRNBP result below PLU Power Level	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-specific review
Flow-dependent MCPR and LHGR limits	Generic	Cycle-independent unless change in plant configuration from licensing analysis basis. Cycle dependent RWE analysis performed with the applicable setpoints.
RBM power-dependent setpoints	Generic, with plant-specific confirmation for initial application	Cycle-independent unless change in plant configuration from licensing analysis basis

3.0 FUEL THERMAL LIMITS

The potentially limiting AOOs and accident analyses were evaluated to support NMP2 operation in the MELLLA region with ARTS off-rated limits. The power/flow state points chosen for the review of AOOs are presented in Table 3-1 and Table 3-2. These state points include the MELLLA region and the current licensed operating domain for NMP2. 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 11 reload licensing analyses (Reference 2) and the NMP2 Updated Safety Analysis Report (USAR) (Reference 3) were examined for operation in the ARTS/MELLLA region (including off-rated power and flow conditions). The following events were considered potentially limiting in the ARTS/MELLLA region and were reviewed as part of the ARTS program development:

1. Generator Load Rejection with No Bypass (LRNBP) event;
2. Turbine Trip with No Bypass (TTNBP) event;
3. Feedwater Controller Failure (FWCF) maximum demand event;
4. Loss of Feedwater Heating (LFWH) event;
5. Rod Withdrawal Error (RWE) event;
6. Fuel Loading Error (FLE) event;
7. Idle Recirculation Loop Start-up (IRLS) event; and
8. Recirculation Flow Increase (RFI) event.

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 analytical methods and input assumptions used for the NMP2 evaluations were consistent with the bases used in Reference 5. The LFWH, FLE, and IRLS events were not re-evaluated for the following reasons.

- The LFWH event is not limiting for NMP2 and the effect of MELLLA on the LFWH severity is sufficiently small that the LFWH remains non-limiting for MELLLA. The LFWH evaluation at 87% flow for NMP2 Cycle 11 showed that there is a large margin in OLMCPR for the LFWH event compared to the LRNBP event (1.21 for the LFWH versus 1.44 for the LRNBP GE14 results at End-of-Cycle (EOC)). At 80% initial core flow, the OLMCPR from the LFWH would still show a large margin to the LRNBP, TTNBP, and FWCF events as shown in Reference 3. [[

]] Finally, considering that the LFWH event becomes less limiting as the power decreases (less feedwater to be affected by loss of heating), the LFWH event was not considered in the determination or validation of the off-rated limits. However, the LFWH event is analyzed on a cycle-specific basis.

- [[ ]] Therefore, this event was also not considered in the determination of the off-rated limits.
- [[ ]] 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.
- The Rod Withdrawal Error is discussed in Section 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 (Pbypass) where reactor scram on turbine stop valve closure or turbine control valve fast closure is bypassed. The analytical value of Pbypass for NMP2 is 30% of RTP. The second power range is between Pbypass and 25% of RTP. No thermal monitoring is required below 25% of RTP, per NMP2 Technical Specifications.

The power dependent MCPR multiplier, K(P), was originally developed for application to all plants in the high power range (between rated power and Pbypass). The values for K(P) increased at lower powers based on the FWCF transient severity trends. As power is reduced from the rated condition in this power range, the generator load rejection with no bypass and turbine trip with no bypass become less severe since 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.

Subsequently, it was identified that the turbine control system performance assumptions used when developing the generic power dependent limits above Pbypass did not correspond to the actual turbine control system performance. This issue was documented to the NRC staff in Reference 40. Specifically, the generic power dependent limits assumed a fast closure of the Turbine Control Valve (TCV) and associated direct scram would occur for all Generator Load Rejections above Pbypass. In reality, the Power Load Unbalance (PLU) device will only initiate a fast closure above a certain power level designated as the PLU power level for this report. For powers between the PLU power level and Pbypass, a Generator Load Rejection will result in the slow closure of the TCVs causing pressure to increase until the high pressure or high neutron flux scram setpoint is reached, terminating the transient. The PLU power level varies from plant to plant. Therefore, plant specific power dependent limits are developed between PLU power level and Pbypass. For NMP2, the PLU power level was identified as 48 % of RTP.

Between Pbypass and 25% power, NMP2 specific evaluations were performed to establish the plant-unique MCPR and LHGR limits in the low power range (below Pbypass). These plant-specific limits include sufficient conservatism to remain valid for future NMP2 reloads of GE14 fuel, except that the power-dependent MCPR limits below Pbypass and flow dependent MCPR limits must be adjusted in accordance with Section 3.3.5 if the SLMCPR exceeds 1.07.

Generic flow-dependent MCPR and LHGR limits are applied to NMP2. These generic limits include sufficient conservatism to remain valid for future NMP2 reloads of GE fuel, utilizing the GEXL-PLUS correlation and the GEMINI analysis methods as defined in Reference 5, providing the core flow corresponding to the maximum two recirculation pump runout is $\leq$ 112.0% of RCF. The flow-dependent MCPR limits must be adjusted in accordance with Section 3.3.5 if the SLMCPR exceeds 1.07.

3.2 Input Assumptions

The maximum power/flow state condition for the operating region analysis is the rated power and maximum flow point (100%P / 105%F). Figure 1-1 shows the power/flow map used in the AOO analyses. Plant heat balance, core coolant hydraulics, and nuclear dynamic parameters corresponding to the rated and off-rated conditions were used for the analysis and reflect the NMP2 Cycle 11 core configuration (Reference 2). 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 Pbypass power ranges), the ARTS transient analyses are based on the CLTP of 3467 MWt. AOO analyses were performed with the approved reload licensing methodology (Reference 5).

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

Analytical Assumptions	Bases/Justifications
Initial core flow range of 80% to 105% flow for thermal limits transients at 100% of RTP	Bounding power/flow state points for MELLLA
Conservative End-of-Cycle 11 nuclear dynamic parameters	Consistent with NMP2 current licensing bases
The lowest two opening setpoint safety-relief valves (SRV) declared Out-of-Service (OOS)	Consistent with NMP2 current licensing bases
SLMCPR = 1.07	Consistent with NMP2 current licensing bases
[[ ]]	Consistent bases of the ARTS program

3.3 Analyses Results

The limits associated with operation in the MELLLA region are presented in Table 3-3. The MELLLA region will be incorporated into subsequent cycle specific reload licensing analyses in accordance with Reference 5. The analyses presented in Table 3-3 are based on End-of-Cycle exposures. [[ ]]

3.3.1 Power-Dependent MCPR Limit

As stated previously, the generic evaluations indicate that the power-dependent severity trends are to be examined in two power ranges, above and below Pbypass.

In the high power range (between rated power and Pbypass), the trend for the power-dependent MCPR responses for the FWCF with the turbine bypass in service has been shown to be more severe than all other fast pressurization transient severity trends. 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 at offrated power. However, as described above, in the operating domain between Pbypass and the point at which the PLU system is enabled, the response to a generator load rejection would be a slow closure of the TCVs. For NMP2, this will result in a scram initiated on high reactor pressure. Therefore, between the PLU power level and Pbypass, the load rejection may be more severe. Plant specific analyses were performed to confirm the generic limits above the PLU power level and to generate plant specific limits between the PLU power level and Pbypass.

The results used to verify the generic MCPR(P) limits analyses are summarized in Table 3-4a through Table 3-4c. As previously stated, the MCPR(P) is derived from the generic K(P) multiplied by the rated power OLMCPR. For power levels above Pbypass, the formula for calculating the generic K(P) is given in Figure 3-1a and Figure 3-1b. A comparison of the plant-specific calculated values with the generic power-dependent MCPR limits verifies the applicability of the generic limits to NMP2 above the PLU enabling power. Below the PLU enabling power, plant specific limits were developed and included in Figure 3-1a and Figure 3-1b.

Below Pbypass, 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 a significant sensitivity to the initial core flow for transients initiated below Pbypass. Therefore, the power-dependent limits are determined for power levels above 25% and below Pbypass based on a core flow of 75%. This core flow bounds the core flow range below 30% power based on the NMP2 Power–Flow 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 1.07 (Technical Specification 2.1). The NMP2 specific analyses results used to establish the MCPR(P) and the MCPR(P) limits for application at power levels below Pbypass are given in Table 3-5a and Table 3-5b.

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,

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

For GE fuel designs, both 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, NMP2-specific transient analyses were performed to demonstrate the applicability of the generic LHGR(P) limits. The transient and initial condition selection are identical to that previously described for MCPR(P). The applicable results of these analyses for power levels above Pbypass are shown in Table 3-6. For NMP2, the equation for LHGRFAC(P) above Pbypass is:

$$LHGRFAC(P) = 1.0 + 0.00523\ (P - 100\%)$$

The generic LHGRFAC(P) above Pbypass are shown in Figure 3-2a and Figure 3-2b.

As previously discussed, a significant sensitivity to initial core flow exists below Pbypass. Therefore, below Pbypass the power-dependent LHGR multipliers are based on a core flow of 75%. To prevent the situation where the limits are more restrictive after increasing power above Pbypass, the extrapolation of the generic above Pbypass limits are taken as the upper bound for the below Pbypass limits. Appropriate LHGRFAC(P) multipliers are selected based on plant-specific transient analyses with suitable margin to ensure applicability to future NMP2 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 results of plant-specific transient analyses below Pbypass are presented in Table 3-7a and Table 3-7b.

Power Levels (% of CLTP)	Core Flow (% of rated)	Generic LHGRFAC(P)
25% < P < 30%	Flow < 75%	0.634 + 0.00523 (P – 30%)
P < 25%	Not applicable	No thermal limits monitoring required

The plant-specific LHGRFAC(P) below Pbypass is shown in Figure 3-2a and Figure 3-2b.

3.3.3 Flow-Dependent Minimum Critical Power Ratio Limit

Flow dependent MCPR limits, MCPR(F), are necessary to assure that the safety limit MCPR 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. This event was also used to determine the current MCPR flow multiplier K(F). [[

]] The bounding generic flow dependent MCPR limits are shown in Figure 3-3. To verify the applicability of the original ARTS generic flow dependent MCPR limits, RFI and IRLS events were re-performed in a generically applicable manner for the GE14 fuel introduction. 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 it 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 MCPR Safety Limit is less than or equal to 1.07.

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) multipliers are shown in Figure 3-4.

3.3.5 Safety Limit Minimum Critical Power Ratio Adjustment Procedure

The MCPR limits, provided in Figure 3-1 and Figure 3-3 assume a dual-loop SLMCPR of 1.07. The off-rated MCPR(P) is defined by Figure 3-1. Only adjustment of the P < Pbypass portion of the MCPR(P) curve may be required because, at P > Pbypass, the K(P) applies the rated power OLMCPR adjustment to the MCPR(P). The off-rated MCPR(F) is defined by Figure 3-3. When necessary, adjustment to the entire MCPR(F) limit is required.

Should a future cycle SLMCPR exceed 1.07, the MCPR(F) and below Pbypass MCPR(P) limits must be increased by the following factor:

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

Should a future cycle SLMCPR be less than 1.07, the MCPR(F) and below-Pbypass 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 Figure 3-3. The off-rated MCPR(P) is defined by Figure 3-1a and Figure 3-1b. Only adjustment of the P < Pbypass portion of the MCPR(P) curve is required because, at P > Pbypass, 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 5. At any power/flow 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 > 1.07 or SLO, as applicable.

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

	Rated	80%F MELLLA	105%F ICF
Power (MWt / % of RTP)	3467.0 / 100.0	3467.0 / 100.0	3467.0 / 100.0
Flow (Mlb/hr / % rated)	108.5 / 100.0	86.8 / 80.0	113.9 / 105.0
Steam Flow (Mlb/hr)	15.002	14.996	15.003
FW Temperature (°F)	425.1	425.1	425.1
Core Inlet Enthalpy (Btu/lb)	529.2	524.1	530.2
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/105%F	85%P/62%F	60%P/105%F	48%P/105%F	30%P/105%F
Power (MWt)	2947.0	2947.0	2080.2	1664.2	1040.1
Flow (Mlb/hr)	113.9	67.5	113.9	113.9	113.9
Steam Flow (Mlb/hr)	12.439	12.427	8.381	6.532	3.891
FW Temperature (°F)	407.3	407.3	372.0	351.1	310.5
Core Inlet Enthalpy (Btu/lb)	529.6	517.1	530.0	530.8	533.0
Dome Pressure (psia)	1022	1022	1003	995	985
	30%P/75%F	25%P/75%F			
Power (MWt)	1040.1	866.8			
Flow (Mlb/hr)	81.4	81.4			
Steam Flow (Mlb/hr)	3.884	3.182			
FW Temperature (°F)	310.4	295.7			
Core Inlet Enthalpy (Btu/lb)	529.4	530.7			
Dome Pressure (psia)	985	983			
(b) Reduced Feedwater Temperature	85%P/105%F	85%P/62%F	60%P/105%F	48%P/105%F	30%P/105%F
Power (MWt)	2947.0	2947.0	2080.2	1664.2	1040.1
Flow (Mlb/hr)	113.9	67.5	113.9	113.9	113.9
Steam Flow (Mlb/hr)	10.855	10.845	7.483	5.904	3.594
FW Temperature (°F)	294.6	294.6	273.5	260.9	236.1
Core Inlet Enthalpy (Btu/lb)	519.4	500.9	524.1	526.7	531.0
Dome Pressure (psia)	1028	1028	1012	1006	998
	30%P/75%F	25%P/75%F			
Power (MWt)	1040.1	866.8			
Flow (Mlb/hr)	81.4	81.4			
Steam Flow (Mlb/hr)	3.589	2.962			
FW Temperature (°F)	236.0	227.0			
Core Inlet Enthalpy (Btu/lb)	526.8	528.7			
Dome Pressure (psia)	998	996			

Table 3-3 MELLLA Transient Analysis Results at RTP Conditions, Cycle 11

Initial Power / Flow (% Rated)	Peak Neutron Flux (% Initial)	Peak Heat Flux (% Initial)	ΔCPR Option B GE14/GE11	ΔCPR Option A GE14/GE11	TOP (%) GE14/GE11	MOP (%) GE14/GE11	Peak Steam Line Pressure (psig)	Peak Vessel Pressure (psig)
100 / 105								
LRNBP	547.8	130.5	0.37 / 0.28	0.40 / 0.31	37.3 / 37.3	38.2 / 38.2	1241	1267
TTNBP	519.4	128.1	0.37 / 0.27	0.40 / 0.30	36.2 / 36.2	37.0 / 37.0	1240	1265
FWCF	444.6	128.1	0.33 / 0.25	0.36 / 0.28	31.8 / 31.8	33.1 / 33.1	1211	1233
100 / 80								
LRNBP	513.1	129.6	0.33 / 0.25	0.36 / 0.28	33.1 / 33.1	33.8 / 33.8	1243	1265
TTNBP	479.8	127.2	0.32 / 0.25	0.35 / 0.28	31.7 / 31.7	32.4 / 32.4	1242	1263
FWCF	439.8	127.9	0.30 / 0.23	0.33 / 0.26	29.4 / 29.4	30.3 / 30.2	1217	1236

Table 3-4 ARTS Transient Analysis Results – Generic K(P) Confirmation Above Pbypass

<table>
<tr><th>Power/Flow</th><th>Event</th><th>ΔCPR(P) Option B GE14/GE11</th><th>ΔCPR(P) Option A GE14/GE11</th><th>K(P) Required</th><th>K(P) Generic</th></tr>
<tr><td colspan="6">(a) Equipment in Service</td></tr>
<tr><td>85 / 105</td><td>LRNBP</td><td>0.34 / 0.26</td><td>0.37 / 0.29</td><td rowspan="8">1.033</td><td rowspan="8">1.056</td></tr>
<tr><td>85 / 105</td><td>TTNBP</td><td>0.34 / 0.25</td><td>0.37 / 0.28</td></tr>
<tr><td>85 / 105</td><td>FWCF</td><td>0.31 / 0.24</td><td>0.34 / 0.27</td></tr>
<tr><td>85 / 105</td><td>FWCF(R)</td><td>0.30 / 0.24</td><td>0.33 / 0.27</td></tr>
<tr><td>85 / 62</td><td>LRNBP</td><td>0.35 / 0.24</td><td>0.38 / 0.27</td></tr>
<tr><td>85 / 62</td><td>TTNBP</td><td>0.32 / 0.22</td><td>0.35 / 0.25</td></tr>
<tr><td>85 / 62</td><td>FWCF</td><td>0.28 / 0.19</td><td>0.31 / 0.22</td></tr>
<tr><td>85 / 62</td><td>FWCF(R)</td><td>0.34 / 0.25</td><td>0.37 / 0.28</td></tr>
<tr><td>60 / 105</td><td>LRNBP</td><td>0.29 / 0.23</td><td>0.32 / 0.26</td><td rowspan="4">1.072</td><td rowspan="4">1.150</td></tr>
<tr><td>60 / 105</td><td>TTNBP</td><td>0.28 / 0.22</td><td>0.31 / 0.25</td></tr>
<tr><td>60 / 105</td><td>FWCF</td><td>0.32 / 0.25</td><td>0.35 / 0.28</td></tr>
<tr><td>60 / 105</td><td>FWCF(R)</td><td>0.33 / 0.27</td><td>0.36 / 0.30</td></tr>
<tr><td>48 / 105</td><td>LRNBP PLU</td><td>N/A</td><td>0.71 / 0.60</td><td rowspan="2">1.340</td><td rowspan="4">1.254</td></tr>
<tr><td>48 / 105</td><td>LRNBP PLU (M)</td><td>N/A</td><td>0.72 / 0.60</td></tr>
<tr><td>48 / 105</td><td>FWCF</td><td>0.33 / 0.27</td><td>0.36 / 0.30</td><td rowspan="2">1.095</td></tr>
<tr><td>48 / 105</td><td>FWCF(R)</td><td>0.36 / 0.30</td><td>0.39 / 0.33</td></tr>
<tr><td>30 / 105</td><td>LRNBP PLU</td><td>N/A</td><td>0.63 / 0.53</td><td rowspan="2">1.276</td><td rowspan="4">1.480</td></tr>
<tr><td>30 / 105</td><td>LRNBP PLU(M)</td><td>N/A</td><td>0.64 / 0.53</td></tr>
<tr><td>30 / 105</td><td>FWCF</td><td>0.43 / 0.36</td><td>0.46 / 0.39</td><td>1.198</td></tr>
<tr><td>30 / 105</td><td>FWCF(R)</td><td>0.51 / 0.43</td><td>0.54 / 0.46</td><td></td></tr>
<tr><td colspan="6">(b) Turbine Bypass Out of Service</td></tr>
<tr><td>85 / 105</td><td>FWCF</td><td>0.40 /0.31</td><td>0.43 / 0.34</td><td rowspan="4">1.031</td><td rowspan="4">1.056</td></tr>
<tr><td>85 / 105</td><td>FWCF(R)</td><td>0.37 / 0.30</td><td>0.40 / 0.33</td></tr>
<tr><td>85 / 62</td><td>FWCF</td><td>0.38 / 0.26</td><td>0.41 / 0.29</td></tr>
<tr><td>85 / 62</td><td>FWCF(R)</td><td>0.43 / 0.32</td><td>0.46 / 0.35</td></tr>
<tr><td>60 / 105</td><td>FWCF</td><td>0.42 / 0.33</td><td>0.45 / 0.36</td><td rowspan="2">1.069</td><td rowspan="2">1.150</td></tr>
<tr><td>60 / 105</td><td>FWCF(R)</td><td>0.41 / 0.34</td><td>0.44 / 0.37</td></tr>
<tr><td>48 / 105</td><td>FWCF</td><td>0.43 / 0.35</td><td>0.46 / 0.38</td><td rowspan="2">1.092</td><td rowspan="2">1.254</td></tr>
<tr><td>48 / 105</td><td>FWCF(R)</td><td>0.45 / 0.37</td><td>0.48 / 0.40</td></tr>
<tr><td>30 / 105</td><td>FWCF</td><td>0.54 / 0.45</td><td>0.57 / 0.48</td><td rowspan="2">1.201</td><td rowspan="2">1.480</td></tr>
<tr><td>30 / 105</td><td>FWCF(R)</td><td>0.61 / 0.51</td><td>0.64 / 0.54</td></tr>
</table>

Table 3-4 (Continued)
ARTS Transient Analysis Results – Generic K(P) Confirmation Above Pbypass

Power/Flow	Event	ΔCPR(P) Option B GE14/GE11	ΔCPR(P) Option A GE14/GE11	K(P) Required	K(P) Generic
(c) Recirculation Pump Trip Out of Service					
85 / 105	LRNBP	0.37 / 0.29	0.48 / 0.36	1.042	1.056
85 / 105	TTNBP	0.38 / 0.29	0.49 / 0.36		
85 / 105	FWCF	0.35 / 0.28	0.46 / 0.35		
85 / 105	FWCF(R)	0.34 / 0.27	0.45 / 0.34		
85 / 62	LRNBP	0.35 / 0.25	0.52 / 0.36		
85 / 62	TTNBP	0.33 / 0.23	0.50 / 0.34		
85 / 62	FWCF	0.29 / 0.21	0.46/ 0.32		
85 / 62	FWCF(R)	0.35 / 0.26	0.52 / 0.37		
85 / 62	LRNBP(M)	0.33 / 0.25	0.44 / 0.32		
85 / 62	TTNBP(M)	0.32 / 0.25	0.43 / 0.32		
85 / 62	FWCF(M)	0.29 / 0.22	0.40 / 0.29		
60 / 105	LRNBP	0.32 / 0.26	0.43 / 0.33	1.042	1.150
60 / 105	TTNBP	0.32 / 0.26	0.43 / 0.33		
60 / 105	FWCF	0.35 / 0.29	0.46 / 0.36		
60 / 105	FWCF(R)	0.36 / 0.30	0.47 / 0.37		
48 / 105	FWCF	0.35 / 0.30	0.46 / 0.37	1.062	1.254
48 / 105	FWCF(R)	0.39 / 0.33	0.50 / 0.40		
48 / 105	FWCF	0.46 / 0.39	0.57 / 0.46	1.168	1.480
48 / 105	FWCF(R)	0.54 / 0.46	0.65 / 0.53		

Notes: 100% Power is defined as 3467 MWt
100% Flow is defined as 108.5 Mlbm/hr
LRNBP – Load Rejection with No Bypass
TTNBP – Turbine Trip with No Bypass
FWCF – Feedwater Controller Failure
FWCF(R) – Feedwater Controller Failure at Reduced Feedwater Temperature
LRNBP PLU – Load Rejection Transient below PLU Power level resulting in no direct scram
LRNBP PLU (M) – LRNBP below PLU Power level evaluated at Middle of Cycle (MOC) conditions

Table 3-5 ARTS Transient Analysis Results – MCPR(P) Below Pbypass

Power/Flow	Event	ΔCPR(P) GE14/GE11	MCPR(P)
(a) Equipment in Service – Recirculation Pump Trip OOS			
30 / 75	LRNBP	0.42 / 0.39	2.29
30 / 75	TTNBP	0.42 / 0.39	
30 / 75	FWCF	0.98 / 0.90	
30 / 75	FWCF (R)	1.11 / 1.01	
25 / 75	LRNBP	0.36 / 0.33	2.42
25 / 75	TTNBP	0.35 / 0.33	
25 / 75	FWCF	1.04 / 0.95	
25 / 75	FWCF (R)	1.22 / 1.12	
(b) Turbine Bypass Valve Out of Service			
30 / 75	FWCF	1.28 / 1.10	2.63
30 / 75	FWCF(R)	1.42 / 1.22	
25 / 75	FWCF	1.63 / 1.42	3.09
25 / 75	FWCF (R)	1.84 / 1.58	

Table 3-6 ARTS Transient Analysis Results – Generic LHGRFAC(P) Confirmation Above Pbypass

Power	Event	LHGRFAC(P) (Calculated)	Generic LHGRFAC(P)
85	LRNBP	0.923	0.922
	TTNBP		
	FWCF		
60	LRNBP	0.941	0.791
	TTNBP		
	FWCF		
48	FWCF	0.765	0.728
	LRNBP		
30	FWCF	0.819	0.634
	LRNBP		

Notes:

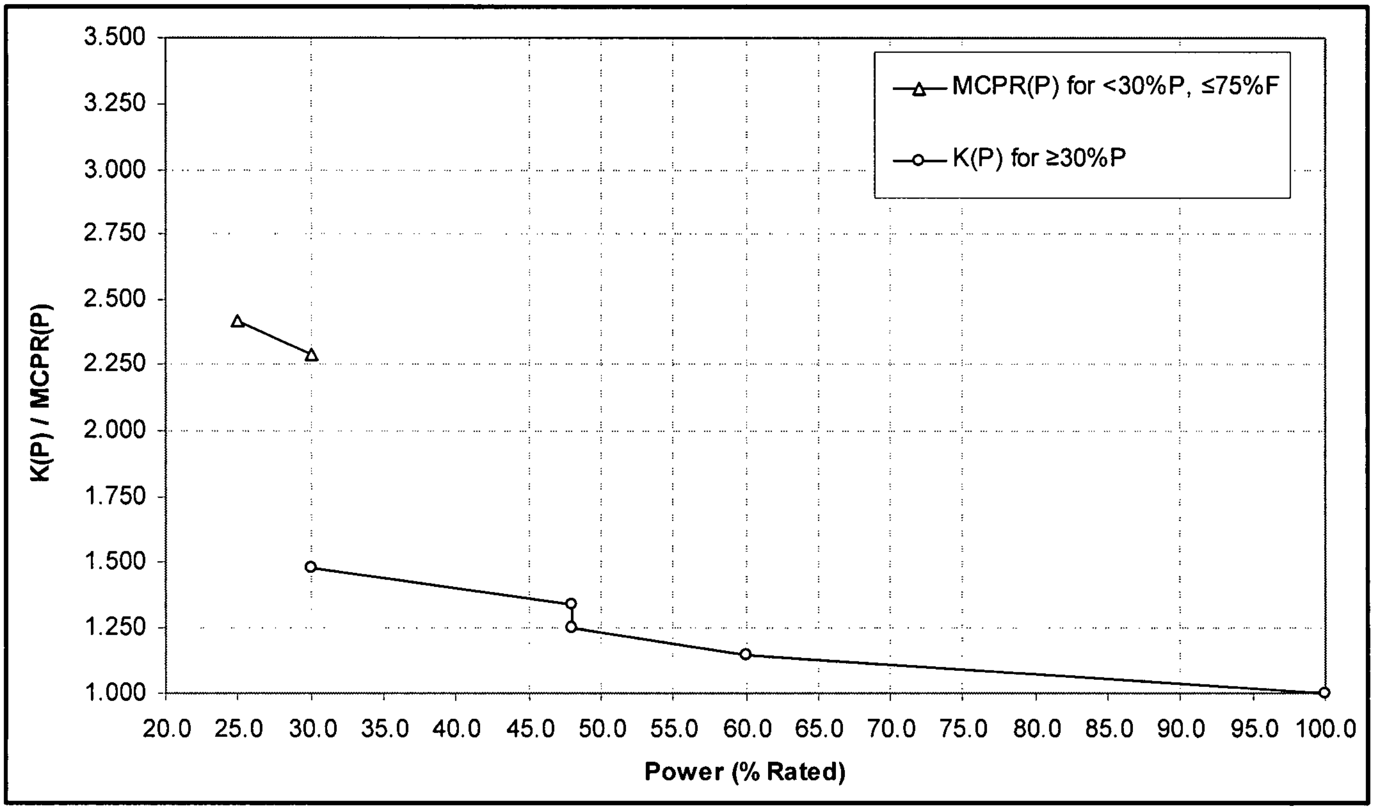
a In order to bound GE11 and GE14 fuel types, the LHGRFAC values were calculated using the highest transient overpower for a given power level and the lowest overpower limits.

b At 60% Power and below, the plastic strain based LHGRFAC does not need to be considered.

c The results include the Turbine Bypass Out-of-Service (TBPOOS) and the Recirculation Pump Trip Out-of-Service (RPTOOS) equipment out of service options.

Table 3-7 ARTS Transient Analysis Results – LHGRFAC(P) Below Pbypass

Initial Power / Flow (%Rated)	Transient	Limiting LHGRFAC(P)
(a) Equipment In Service / Recirculation Pump Trip Out of Service		
30 / 75	LRNBP	0.659
	TTNBP	
	FWCF	
25 / 75	LRNBP	0.674
	TTNBP	
	FWCF	
(b) Turbine Bypass Valve Out of Service		
30 / 75	FWCF	0.583
25 / 75	FWCF	0.549



Operating Limit MCPR(P) = K(P) * Operating Limit MCPR(100)
For P < 25%, No Thermal Limits Required
P-Bypass = 30% Rated Power
MCPR(P) limits are based on a 1.07 SLMCPR

MCPR(P) for <30%P, ≤75%F

POWER	LIMIT
25.0	2.42
30.0	2.29

EQUATIONS

For 25.0% ≤ P < 30.0%: MCPR(P) = 2.29 + 2.60E-02(30.0 – P)

K(P) for ≥30%P

POWER	LIMIT
30.0	1.480
48.0	1.340
48.0 +	1.254
60.0	1.150
100.0	1.000

EQUATIONS

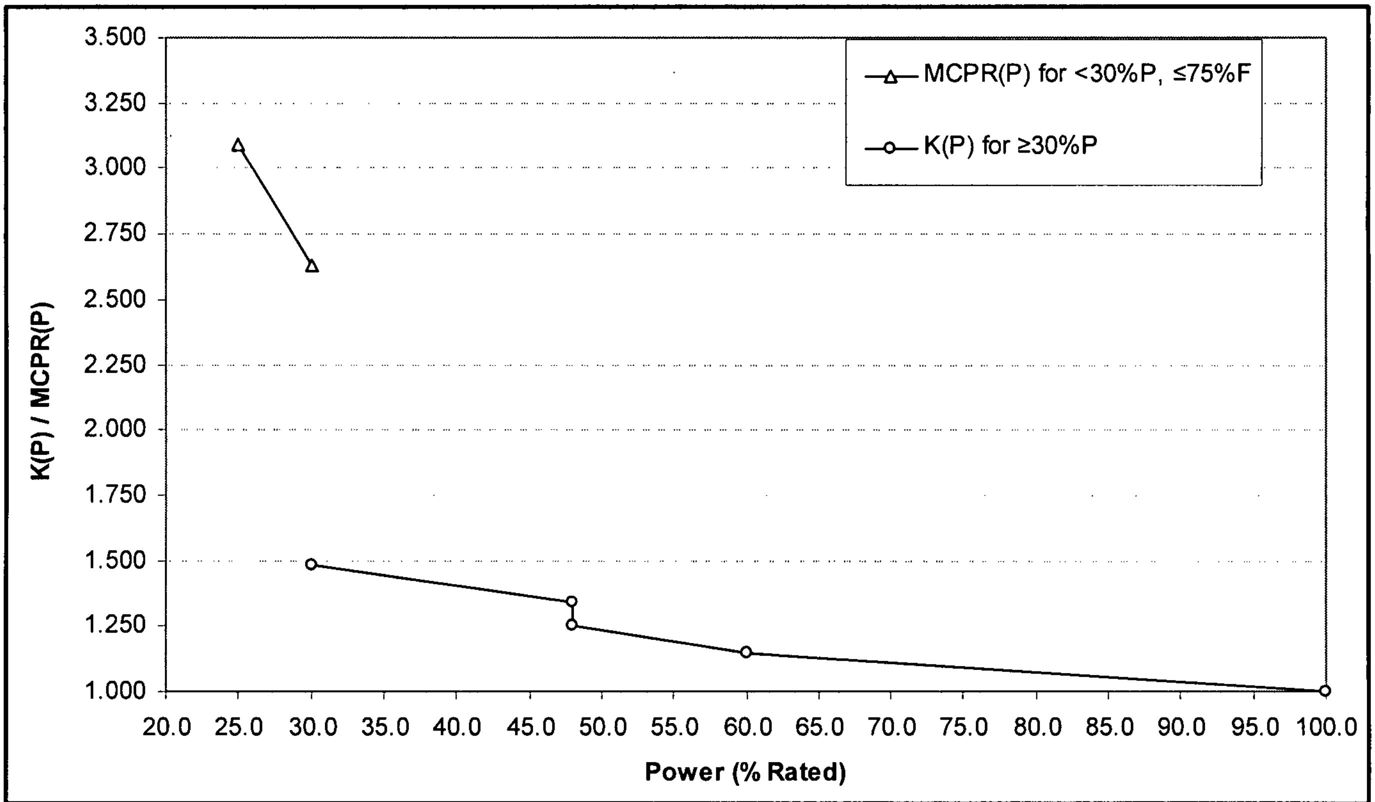
For 30.0% ≤ P ≤ 48.0%: K(P) = 1.340 + 7.78E-03(48.0 – P)

For 48% < P < 60.0%: K(P) = 1.150 + 8.67E-03(60.0 – P)

For 60.0% ≤ P < 100.0%: K(P) = 1.000 + 3.75E-03(100.0 – P)

Figure 3-1 Power-Dependent MCPR Limits, MCPR(P) / K(P)

(a) Equipment In Service and Recirculation Pump Trip Out of Service



Operating Limit MCPR(P) = K(P) * Operating Limit MCPR(100)
For P < 25%, No Thermal Limits Required
P-Bypass = 30% Rated Power
MCPR(P) limits are based on a 1.07 SLMCPR

MCPR(P) for <30%P, ≤75%F

POWER	LIMIT
25.0	3.09
30.0	2.63

EQUATIONS

For 25.0% ≤ P < 30.0%: MCPR(P) = 2.63 + 9.20E-02(30.0 – P)

K(P) for ≥30%P

POWER	LIMIT
30.0	1.480
48.0	1.340
48.0 +	1.254
60.0	1.150
100.0	1.000

EQUATIONS

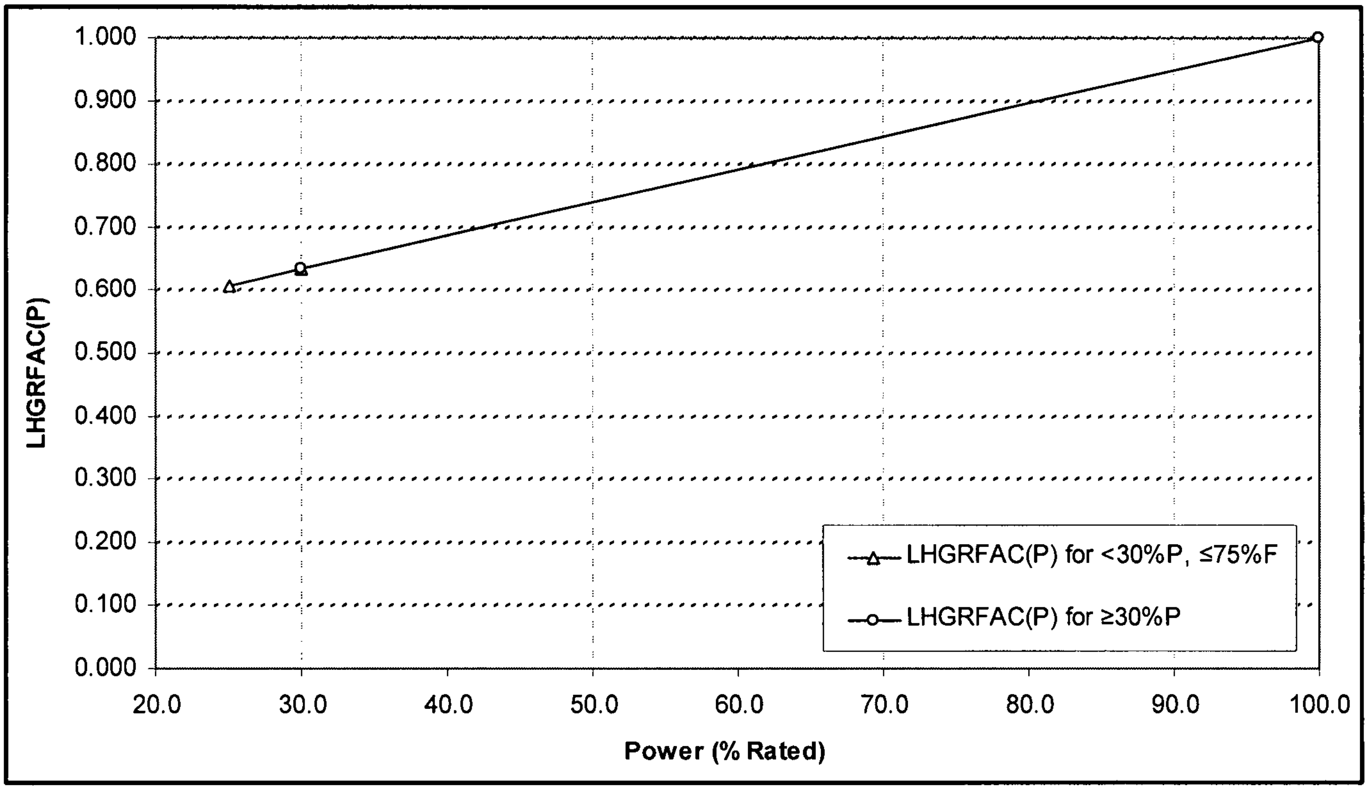
For 30.0% ≤ P ≤ 48.0%: K(P) = 1.340 + 7.78E-03(48.0 – P)

For 48.0% < P < 60.0%: K(P) = 1.150 + 8.67E-03(60.0 – P)

For 60.0% ≤ P < 100.0%: K(P) = 1.000 + 3.75E-03(100.0 – P)

Figure 3-1 Power-Dependent MCPR Limits, MCPR(P) / K(P)

(b) Turbine Bypass Out of Service



LHGR(P) = LHGRFAC(P) * LHGRstd

For P < 25%, No Thermal Limits Required

P-Bypass = 30% Rated Power

LHGRFAC(P) for <30%P, ≤75%F

POWER	LIMIT
25.0	0.608
30.0	0.634

EQUATIONS

For 25.0% ≤ P < 30.0%: LHGRFAC(P) = 0.634 + 5.23E-03(P - 30.0)

LHGRFAC(P) for ≥30%P

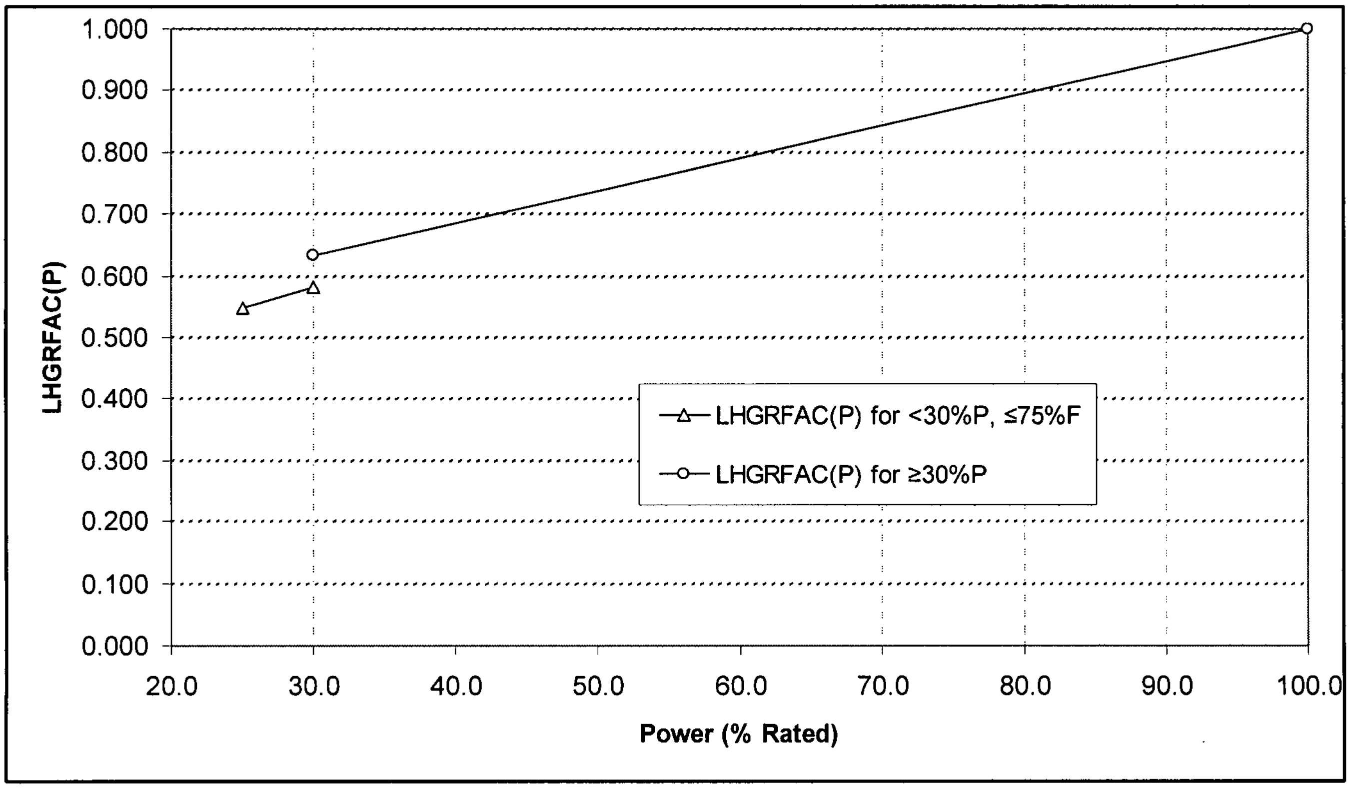
POWER	LIMIT
30.0	0.634
100.0	1.000

EQUATIONS

For 30.0% ≤ P < 100.0%: LHGRFAC(P) = 1.000 + 5.23E-03(P - 100.0)

Figure 3-2 Power-Dependent LHGR Multiplier, LHGRFAC(P)

(a) Equipment in Service and Recirculation Pump Trip Out of Service



LHGR(P) = LHGRFAC(P) * LHGRstd
For P < 25%, No Thermal Limits Required
P-Bypass = 30% Rated Power

LHGRFAC(P) for <30%P, ≤75%F

POWER	LIMIT
25.0	0.549
30.0	0.583

EQUATIONS

For 25.0% ≤ P < 30.0%: LHGRFAC(P) = 0.583 + 6.80E-03(P - 30.0)

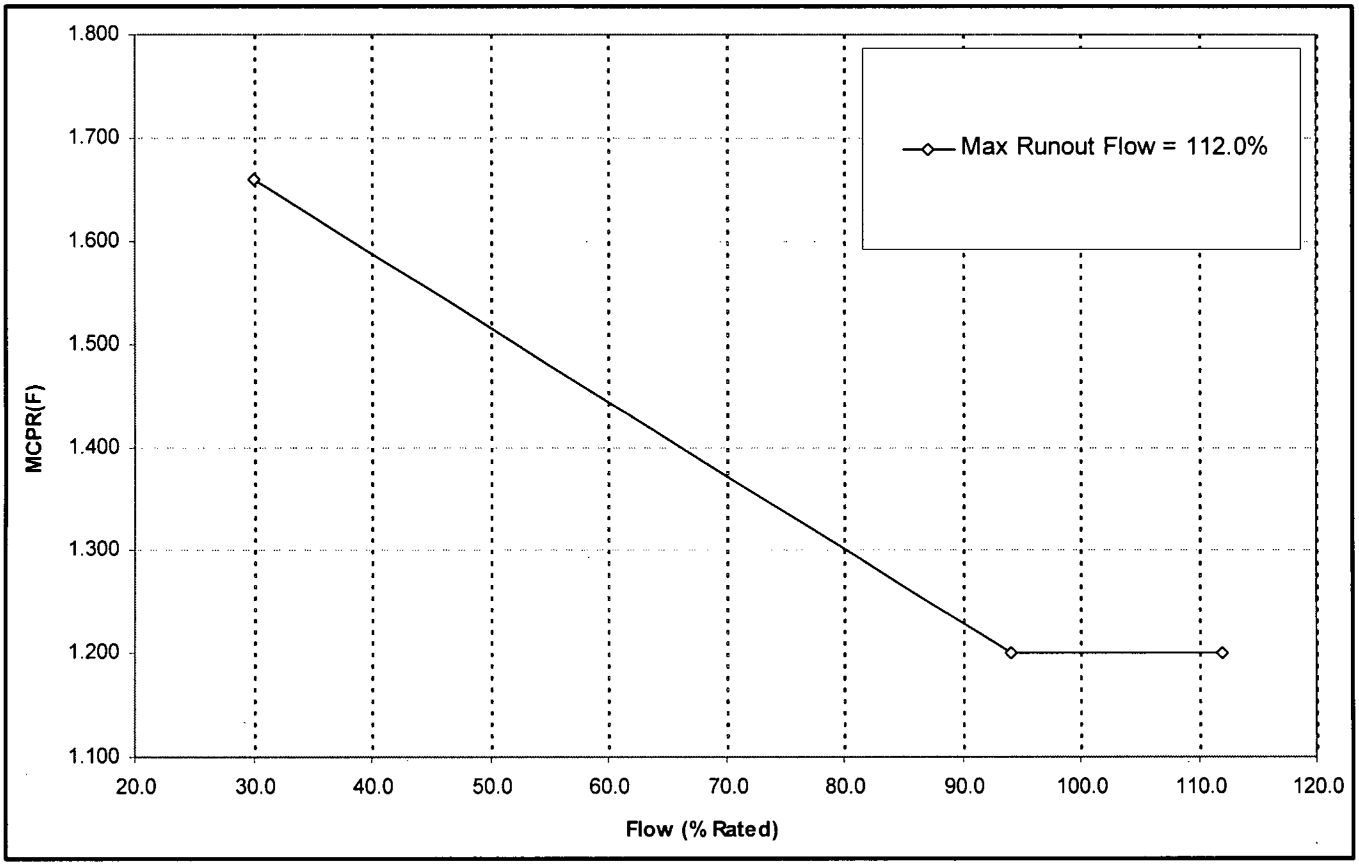
LHGRFAC(P) for ≥30%P

POWER	LIMIT
30.0	0.634
100.0	1.000

EQUATIONS

For 30.0% ≤ P < 100.0%: LHGRFAC(P) = 1.000 + 5.23E-03(P - 100.0)

Figure 3-2 Power-Dependent LHGR Multiplier, LHGRFAC(P)
(b) Turbine Bypass Out of Service



MCPR(F) limits are based on a 1.07 SLMCPR

Max Runout Flow = 112.0%

FLOW	LIMIT
30.0	1.66
94.2	1.20
112.0	1.20

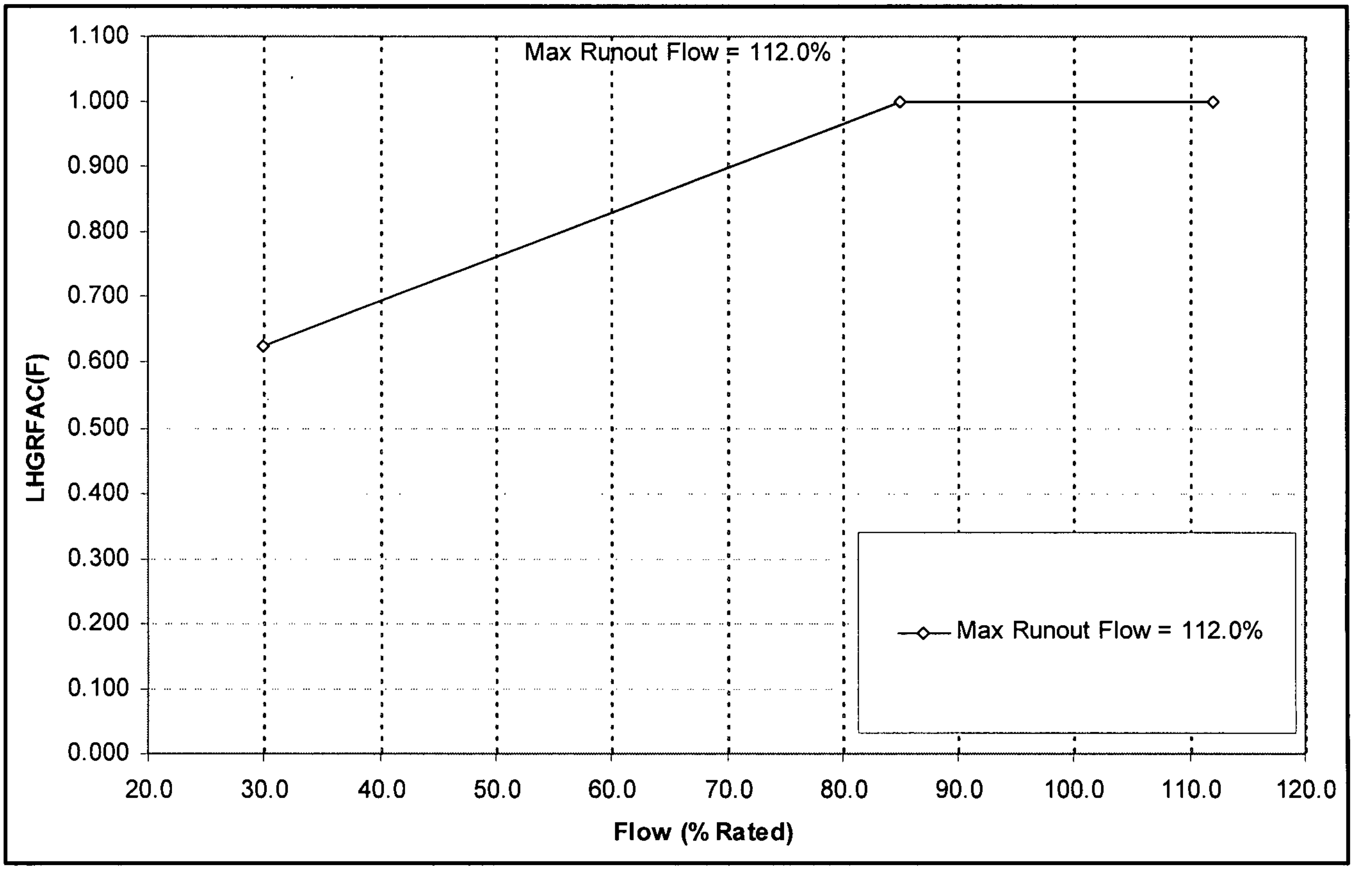
EQUATIONS

MCPR(F) = MAX(1.20, [A(F)*W(C)/100 + B(F)])

A(F) = -0.724

B(F) = 1.882

Figure 3-3 Flow-Dependent MCPR Limits, MCPR(F)



LHGR(F) = LHGRFAC(F) * LHGRstd

Max Runout Flow = 112.0%

FLOW	LIMIT
30.0	0.625
85.0	1.000
112.0	1.000

EQUATIONS

LHGRFAC(F) = MIN(1.0, [A(F)*W(C)/100 + B(F)])

A(F) = 0.681

B(F) = 0.421

Figure 3-4 Flow-Dependent LHGR Multiplier, LHGRFAC(F)

4.0 ROD BLOCK MONITOR SYSTEM IMPROVEMENTS

The function of the Rod Block Monitor (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 Rod Withdrawal Error (RWE) analysis based on the improved RBM system.

4.1 Current Rod Block Monitor System Description

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 NMP2 settings are 110%, 102%, and 94% 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 2% less than the trip point. The operator should then assess the local power and either acknowledge or select a new rod. The highest trip point cannot be bypassed.

A count of the active LPRMs is made automatically and the RBM 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. A 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 NMP2 RBM System is presented in Figure 4-1.

4.1.2 Limitations of Current Rod Block Monitor System

The original NMP2 RBM system electronics were upgraded for the RBM to fulfill its intended function more effectively, for power tracking changes (Reference 4) as part of the GE NUMAC™ PRNM retrofit. The licensing basis for the NUMAC™ PRNM retrofit was based on Reference 38. However, the NUMAC™ PRNM retrofit did not change the functional design of the original NMP2 RBM System as described in Section 4.1.1. Essentially, the current RBM function of the NUMAC™ PRNM installed at NMP2 maintains the original RBM functional design that was based upon mid-1960s technologies. Since the 1960's, there have been significant technological advances in the fields 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 which 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 2% reset band immediately below the two lower trip settings, the next higher RBM trip setting is automatically selected for enforcement. Similarly, manual reset of the lower trip to the next higher trip is allowed when the local power reaches the 2% band as a result of rod withdrawal. In this case, the operator verifies that adequate thermal margins exist before resetting the trips. These reset features are a necessary result of the normalization of the signals to the APRM. If the APRM power is just below the trip, random noise in the signals may cause the trip to be exceeded and no withdrawal will be possible. Since the flow-biased trip settings are roughly parallel the flow control lines, it would be very difficult to increase core power above a RBM trip setting without the reset features. Resets are possible only for the two lower trip settings; the high trip cannot be reset. Since the highest trip setting cannot be reset, another direct consequence of the normalization of the RBM signals to the reference APRM is that control rod withdrawal is not permitted when the reference APRM exceeds the highest RBM trip setting.

Figure 4-3 illustrates an ideal startup path in which rated power is attained without control rod movement after recirculation flow has been increased above the minimum recirculation flow control valve position. 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 NMP2 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 current NMP2 RBM subsystem of the PRNM upgrade to the NUMAC™ platform was licensed and installed in accordance with Section 2.3.3.5.1.3 of Reference 38. The implementation option addressed by this reference section is "*ARTS implementation after the replacement PRNM installation (Non-BWR6 plants that are planning to obtain approval and implement the ARTS improvement at some time after the PRNM project.)*." The current PRNM system hardware installed at NMP2 was specifically designed to allow ARTS implementation without any hardware changes to the LPRM subsystem, APRM subsystem, Oscillating Power Range Monitor (OPRM) subsystem, Recirculation Flow subsystem, or RBM subsystem. The change to the ARTS based RBM consists of upgrading the RBM NUMAC™ chassis functional and display firmware via Electrical Programmable Read Only Memory (EPROM). 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.

A number of alternatives to the current LPRM assignment were studied by GE. 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 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 NMP2 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 NMP2 has been performed and is summarized in Table 4-2. The application of these results is validated for GE fuel and core design for each reload analysis in accordance with Reference 5.

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 power/flow map, two other power/flow 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 5. 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 which 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, error rod position at the rod block trip level was generated as a function of RBM setpoint. The results were tabulated as functions of RBM setpoint. The parameter of interest is the normalized MCPR change, 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. These 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 power/flow 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 safety 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 power/flow 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 power/flow map. The primary parameters affecting an RWE are initial rod pattern and void fraction. Because these parameters are essentially fixed along a given flow control line, [[

]]

The generic ARTS results presented thus far were performed utilizing a 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 (Figure 3-1) for a rated MCPR operating limit of 1.20 assuming a SLMCPR of 1.07. This value was chosen to assure that RWE will not significantly limit plant operation. These RBM setpoints are analytical values and verified to be applicable to NMP2. 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 RBM downscale trip setpoint was selected to detect abnormally low RBM signal conditions. Control rod withdrawal is blocked when the RBM is downscale.

The generic RWE analyses also verified the conformance to the fuel thermal-mechanical limit (i.e., 1% plastic strain) for GE fuel designs. Plant-specific RWE evaluations have been performed for NMP2 using the reference core loading for Cycle 11 and the results show that these criteria are met. Specifically, the RWE MCPR requirement for the Cycle 11 calculations is 1.15, compared against a generic 95/95 requirement of 1.20 for a RBM setpoint of 108%. In addition, calculations will be performed as part of the reload analyses in accordance with Reference 5 to confirm the applicability of the ARTS-based statistical RWE result for subsequent fuel cycles at NMP2. 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 NMP2 core are shown in Figure 4-11. The rod group geometries and error rods studied are shown in Figure 4-12. A single case was selected from the database used to establish the RBM setpoints. This case was re-analyzed with the various geometries of Figure 4-12 substituted for the standard four-string geometry. For this study, the RBM setpoint was fixed at

108%. Results of the study (Table 4-3) show no significant differences between the base (four string) case and the limiting peripheral geometries. [[

]]

[[

]]

4.3.2.2 Local Power Range Monitor Failures

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

4.3.2.3 Effect of Filter on Rod Block Monitor Signal

[[

]]

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 impact of the GEXL-PLUS correlation (applicable to GE11 and GE14 fuel in NMP2 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 Increased Core Flow conditions as well as in the MELLLA region. A comparison of these ΔCPRs with limiting ARTS RWE ΔCPR of 0.13 (corresponding to the RBM setpoints in Table 4-5) indicates that a minimum margin of 0.15 (for GE11 fuel) 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 5.

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 NMP2 NUMAC™ RBM implementation, this filter (T_{c2}) is eliminated because the incoming APRM signal is the STP signal, which already has a 6 second filter on it.

---

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

The first delay, T_{d1}, delays gain adjustment and signal normalization for a preset time interval following rod selection and is necessary to allow the filtered RBM signal to approach its asymptotic value. (No rod withdrawal is possible during this period.) For optimum performance based on experience from plants operating with the ARTS based RBM, this time delay ($_{td1}$) 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 application 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 NMP2 since 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 NMP2 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 Table 4-4. 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 safety limit MCPR is violated. When any control rod in the core will violate this limit upon complete withdrawal, operability of the RBM System is required. The RBM System basis is limited to consideration of single control rod withdrawal errors and does not accommodate multiple errors. 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 $\leq$ 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.

4.6 Rod Block Monitor Modification Compliance to Nuclear Regulatory Commission Regulations and Licensing Topical Reports

Modifications to the RBM firmware will be performed, consistent with the quality requirements as addressed in Reference 38, 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 38). 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 41). 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 benchboard changes have been reviewed and are adequate with the changes (see Section 2.3.3.6.2 of Reference 38).

4.7 Conclusion

The firmware change to the existing NMP2 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	[[
IPSP	65 / 65	
HPSP	85 / 85	]]
LTSP	[[	
ITSP		
HTSP		]]
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 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.

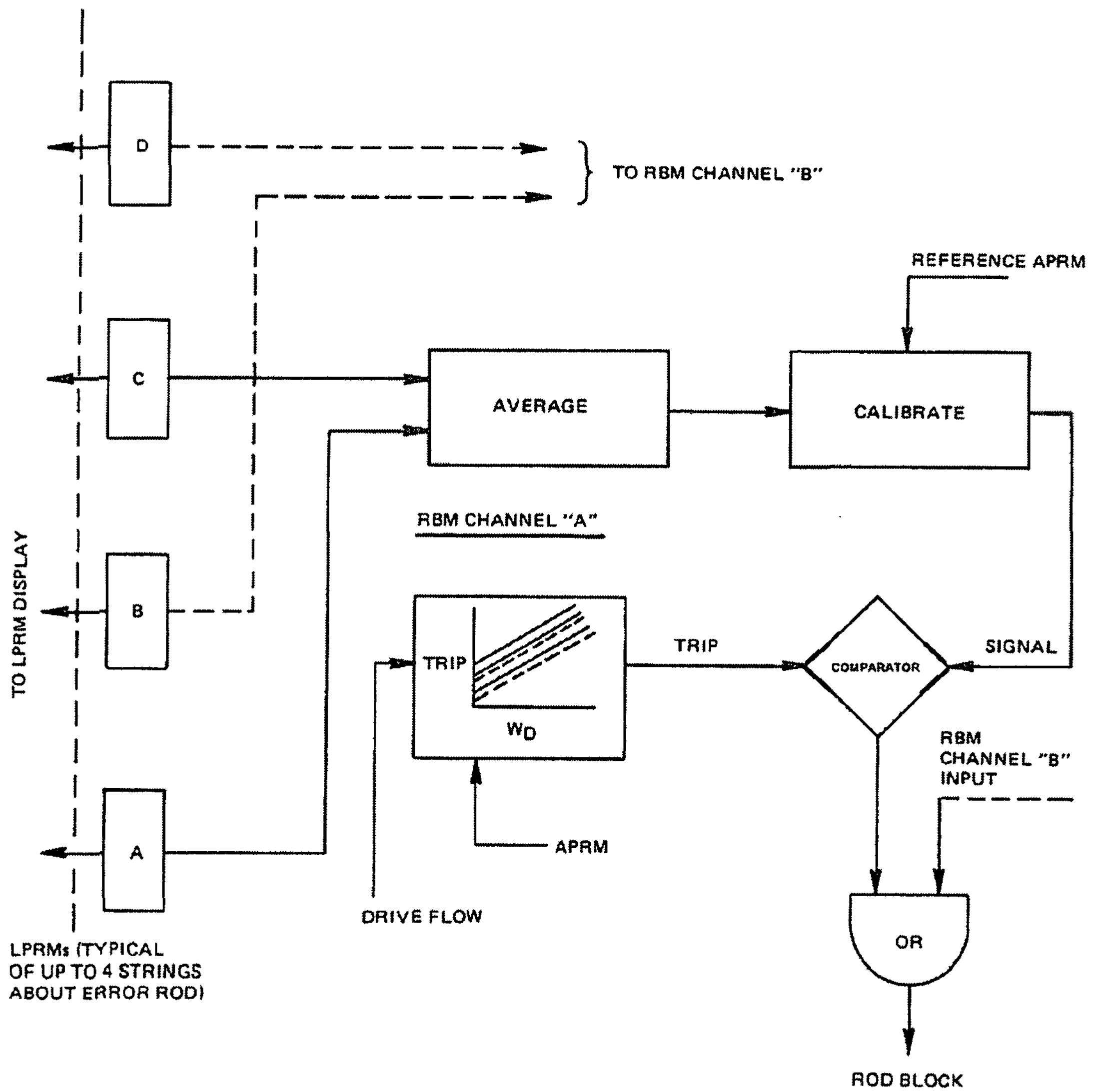


Figure 4-1 Conceptual Illustration of Current Flow-Dependent RBM System with AC/BD LPRM Assignment

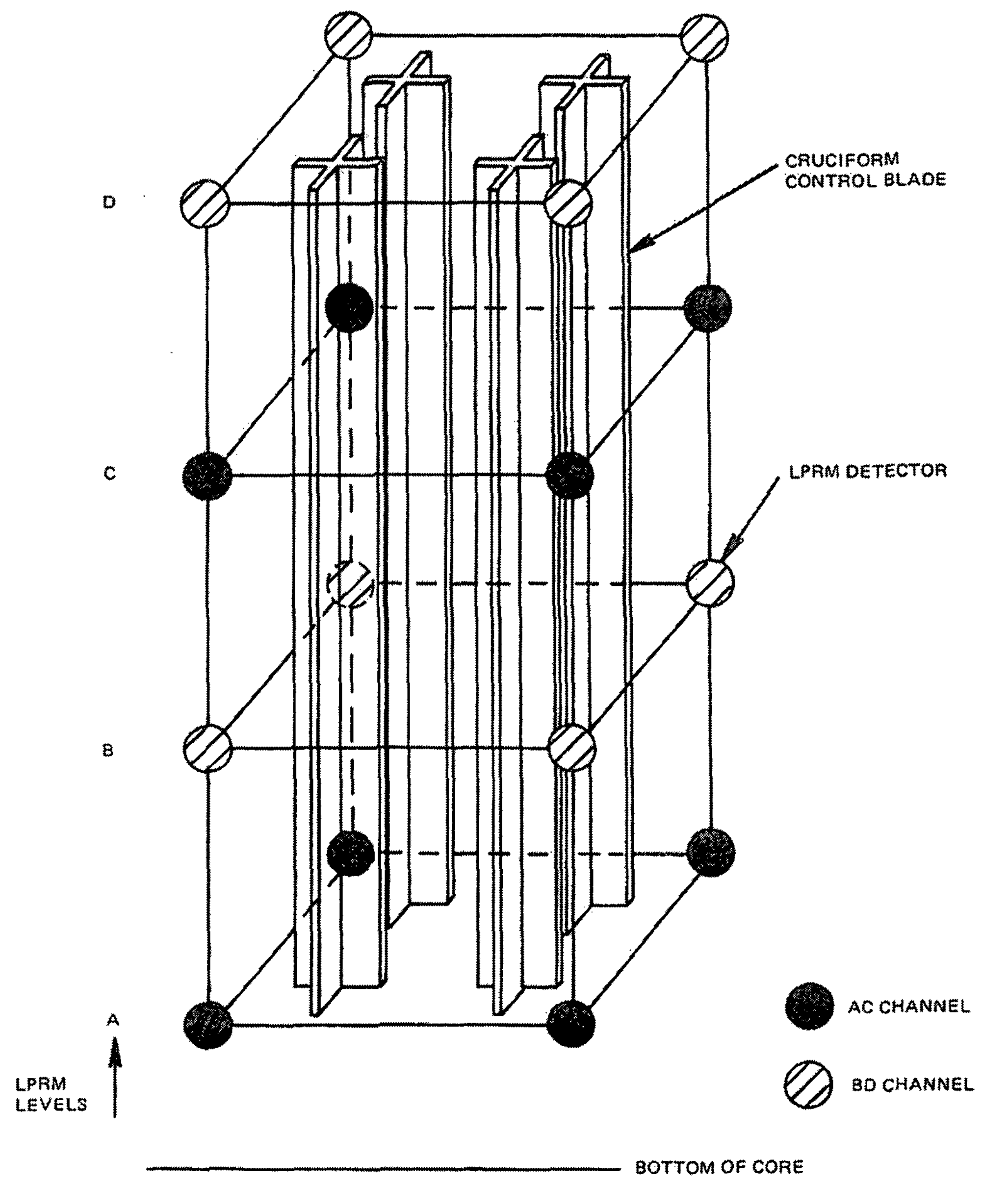


Figure 4-2 RBM Current AC/BD LPRM Assignment

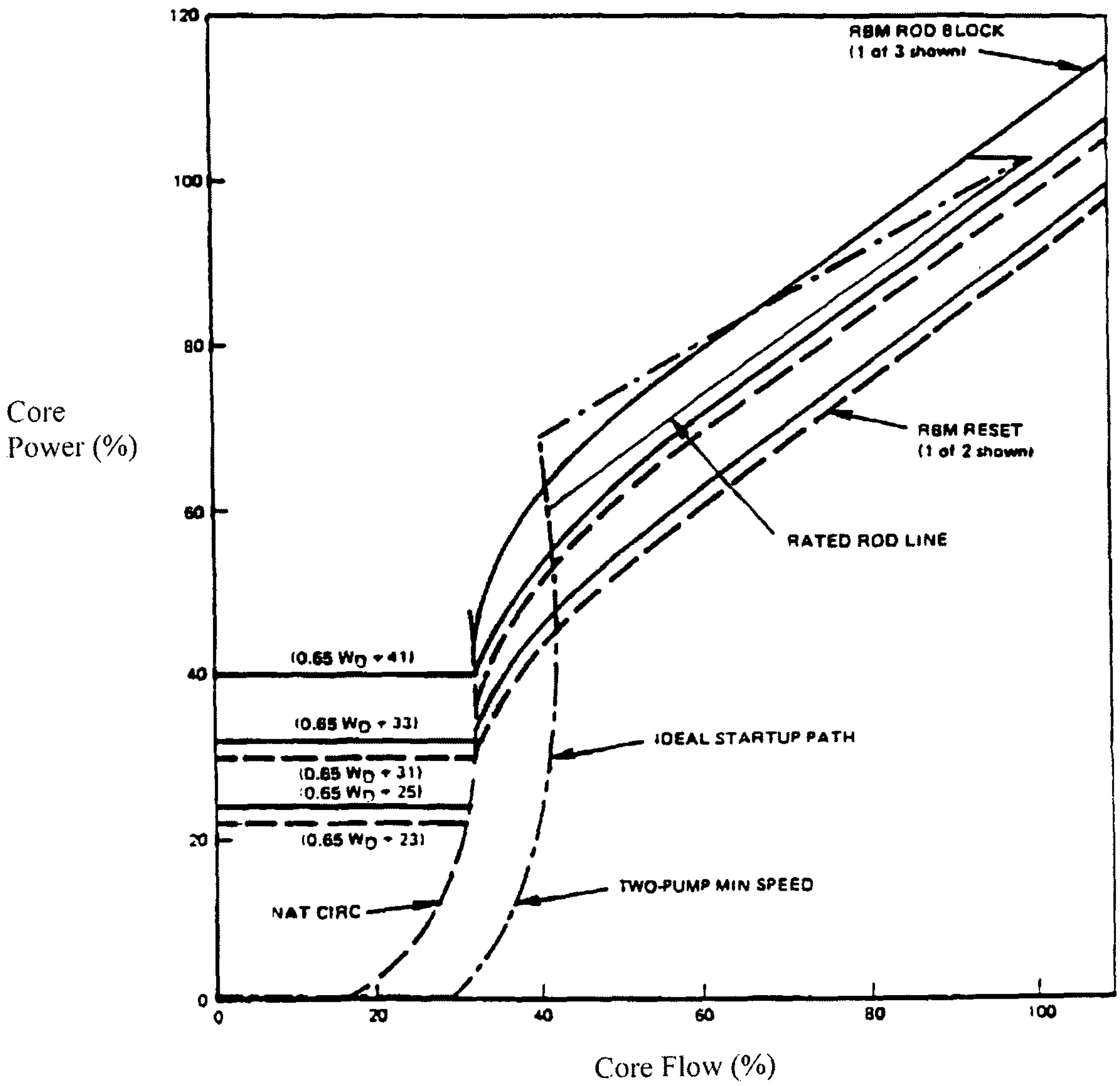


Figure 4-3 Current RBM System Configuration Limits (Typical for 106% Setpoint)

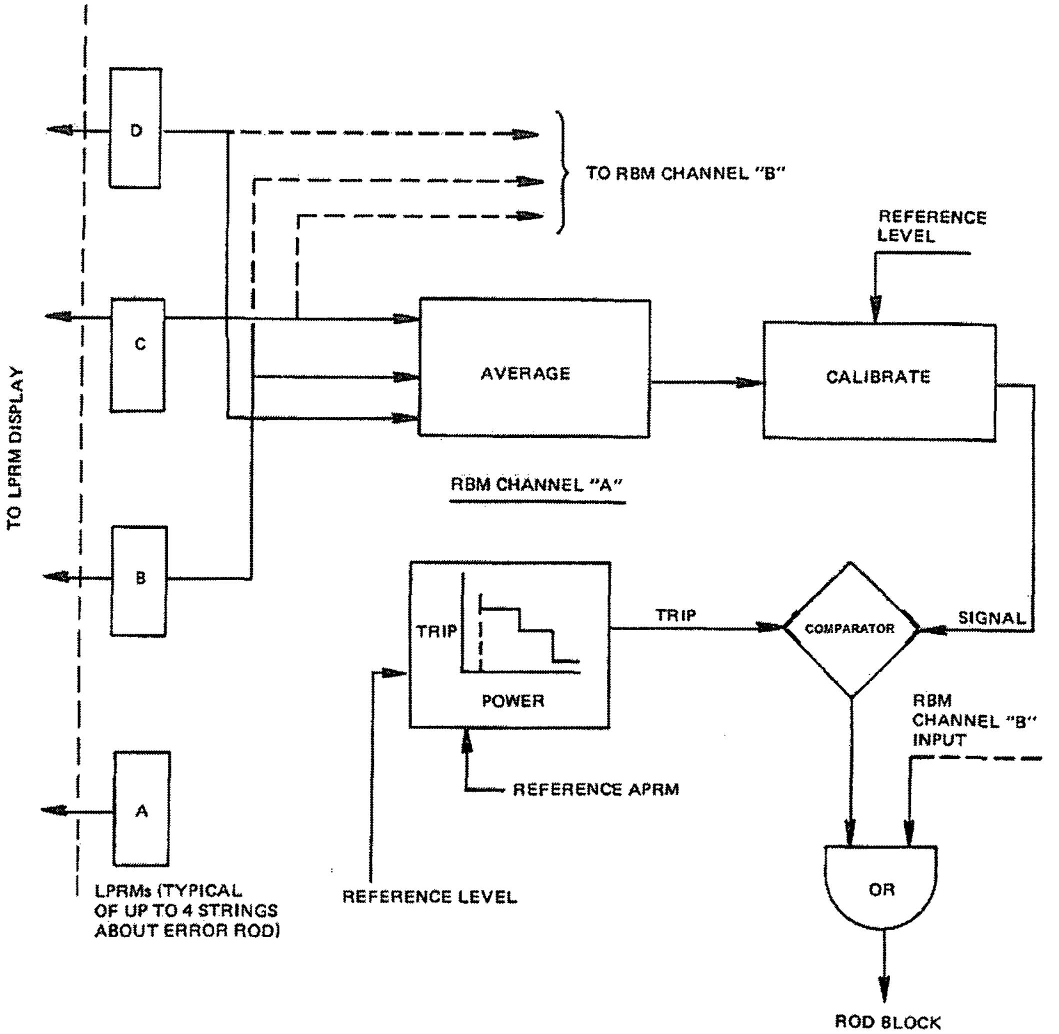


Figure 4-4 New Power-Dependent RBM System with $BCCD_1/BCCD_2$ LPRM Assignment

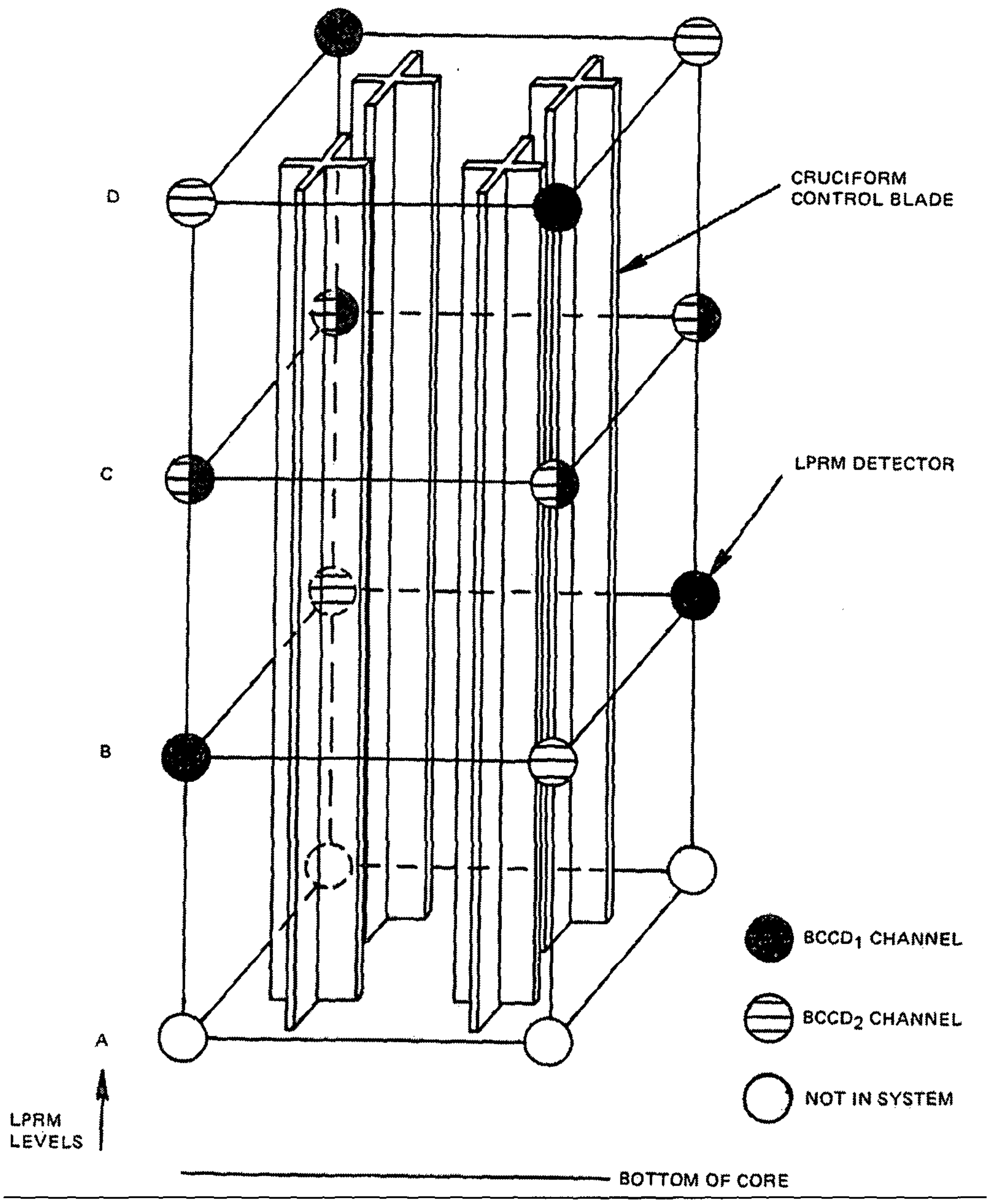


Figure 4-5 New RBM $BCCD_1$/$BCCD_2$ LPRM Assignment

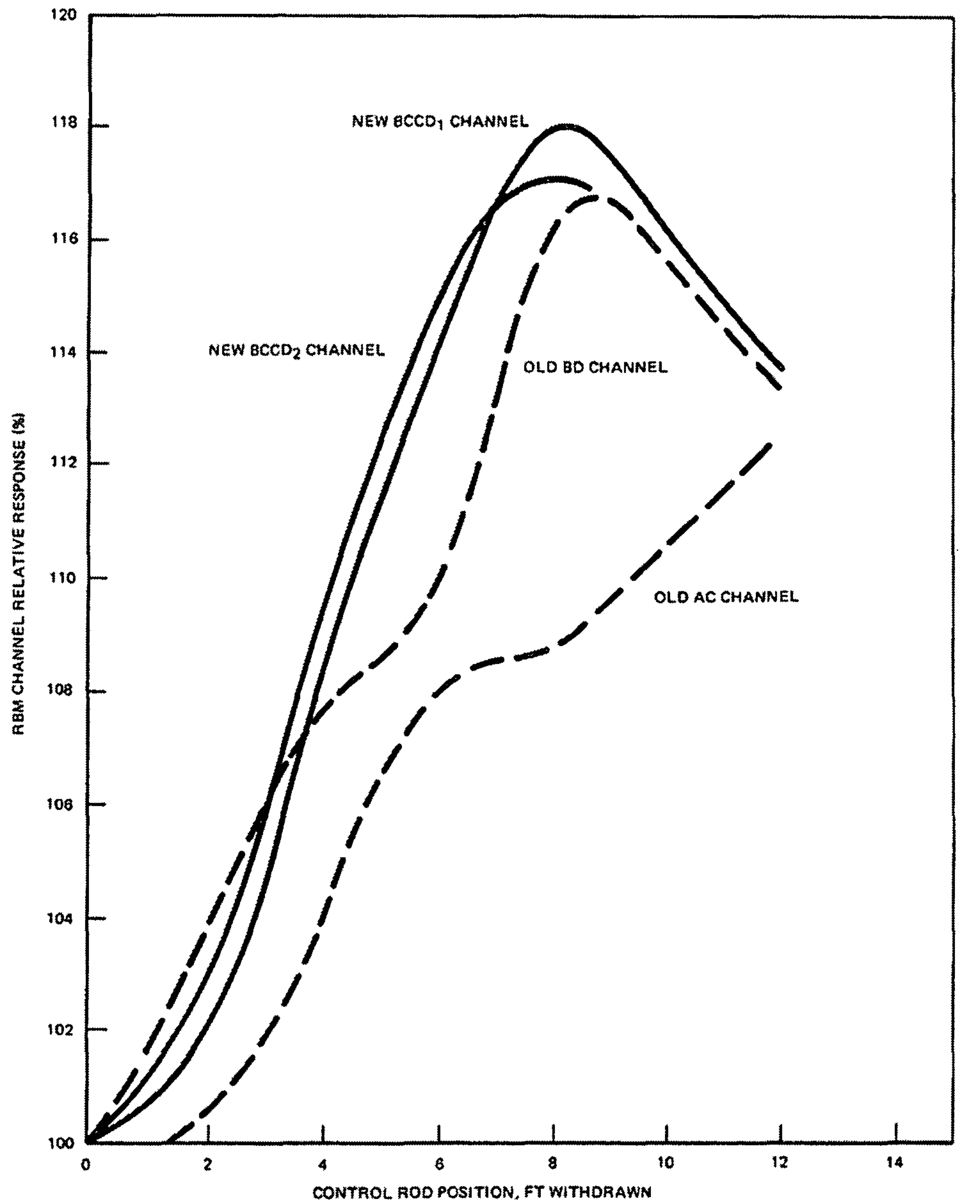


Figure 4-6 Typical RBM Channel Responses, Old Versus New LPRM Assignment (No Failed LPRMs)

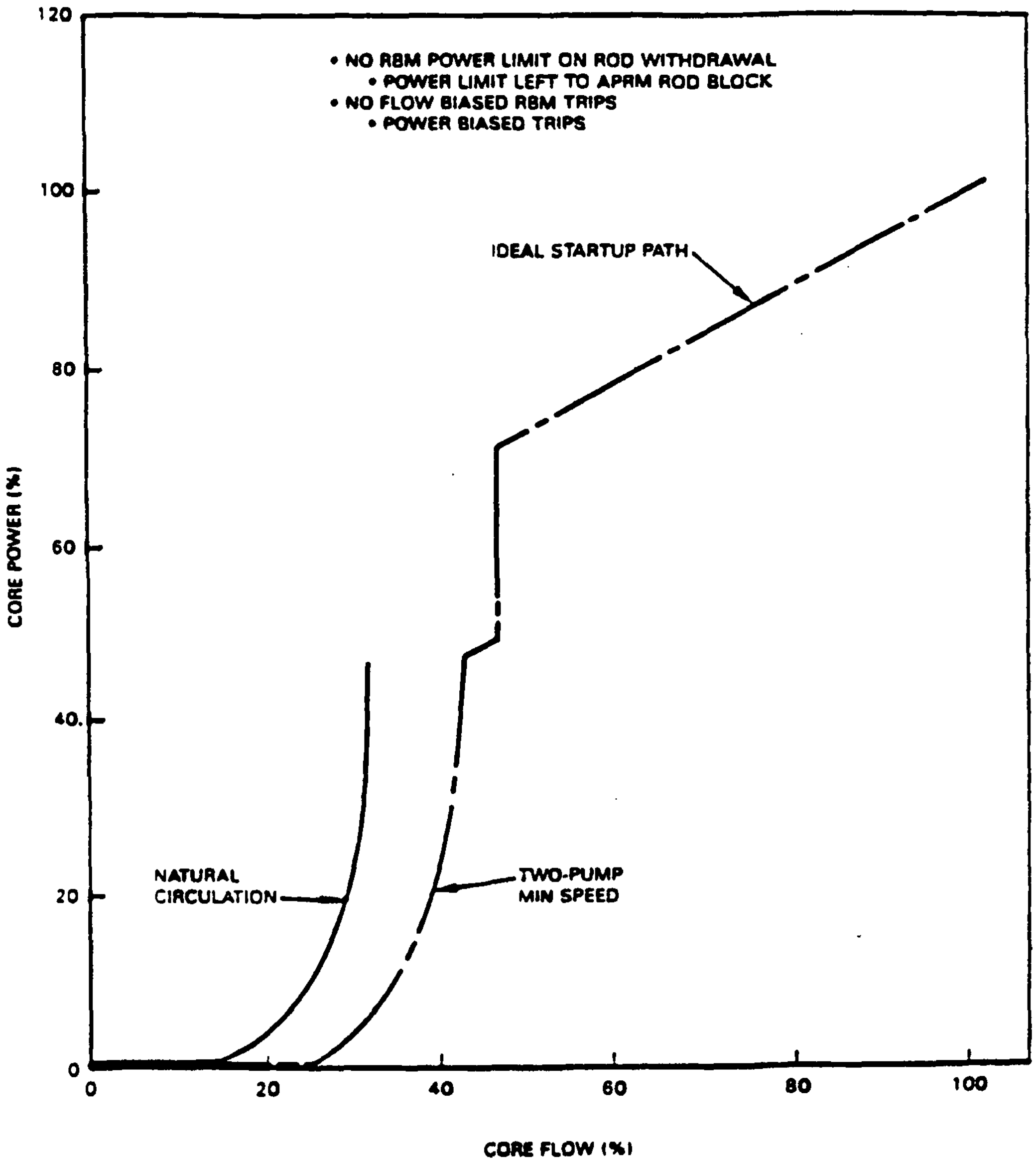


Figure 4-7 New RBM System Core Power Limit (Typical)

[[

]]

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

[[

]]

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

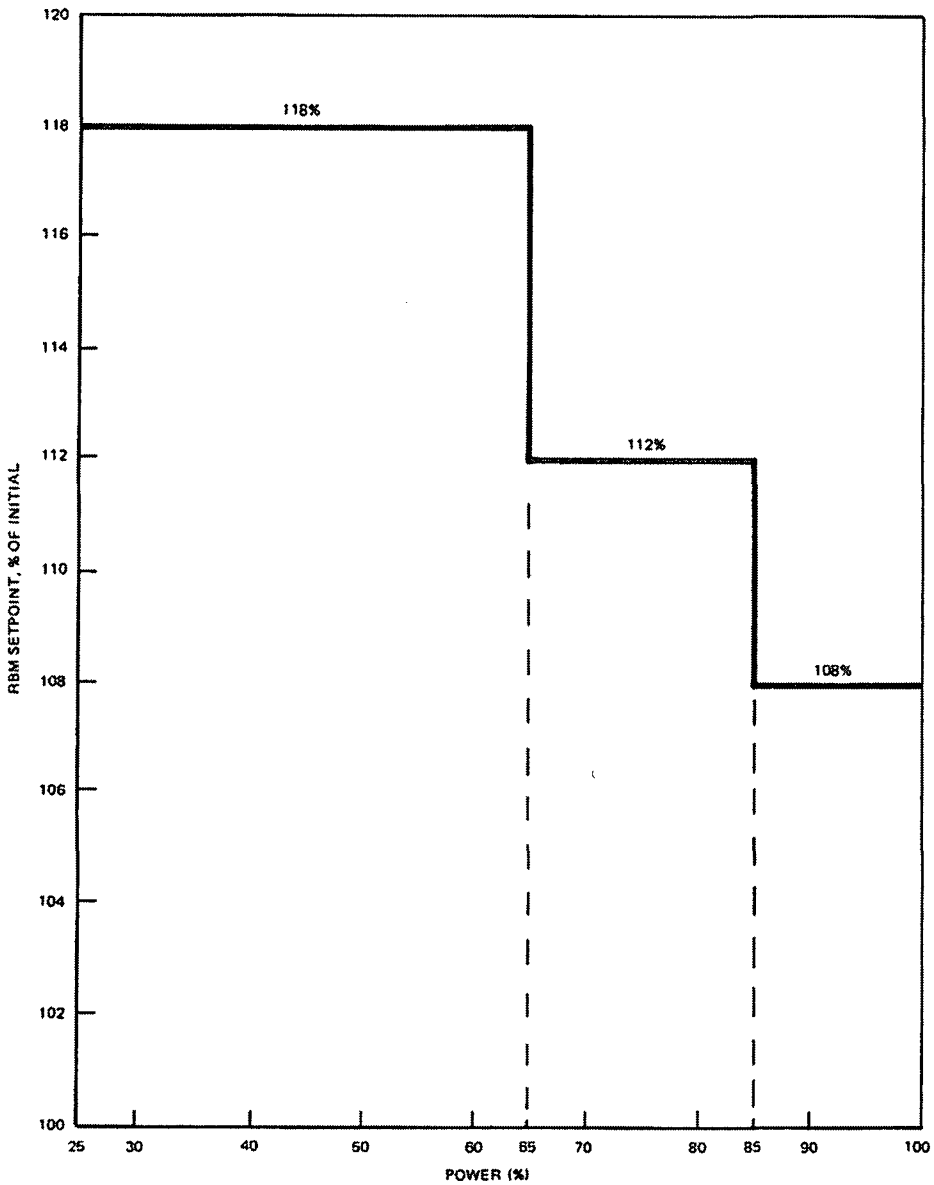
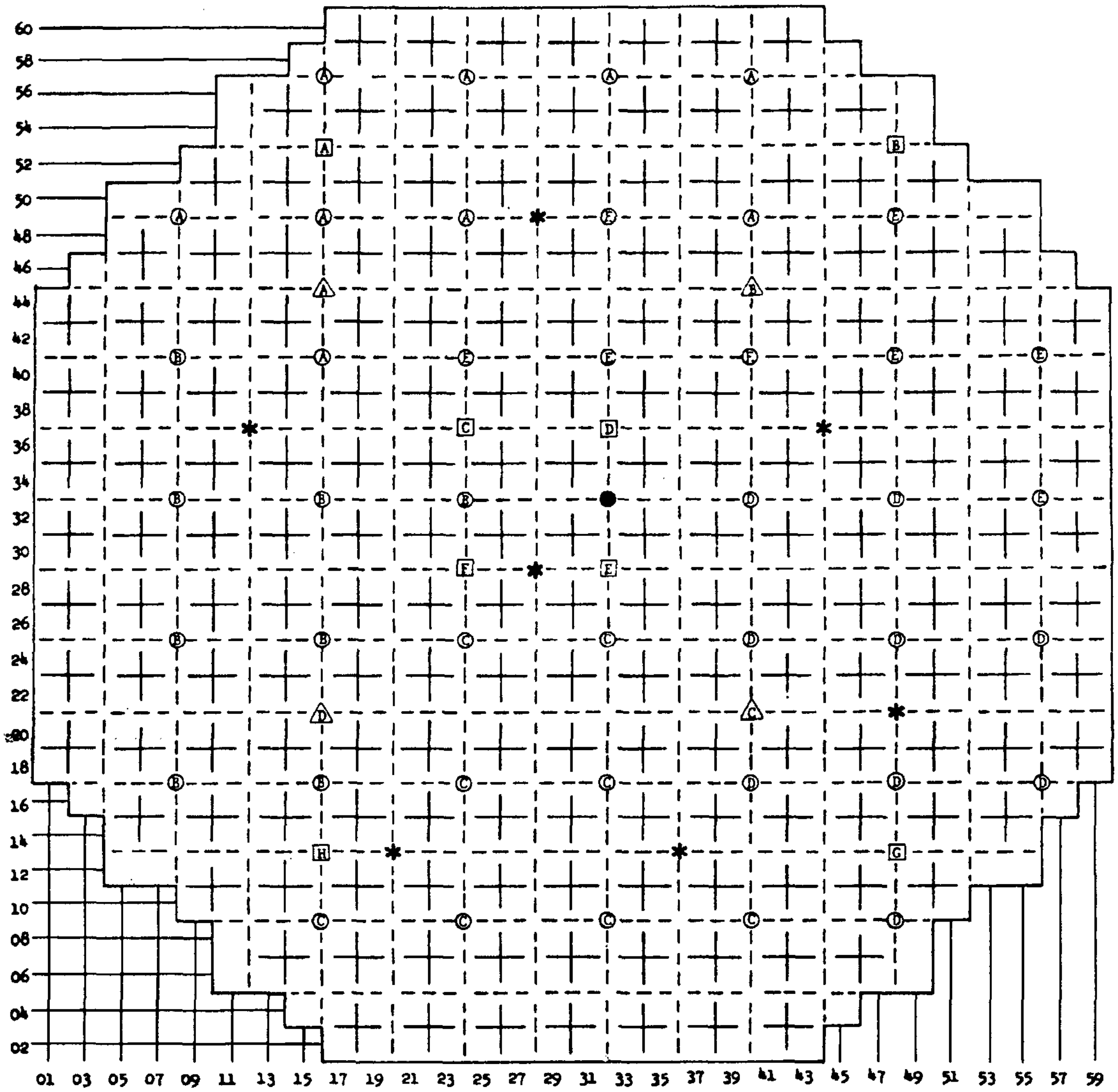


Figure 4-10 RBM Setpoint Versus Power (without Filter)



Ⓐ LPRM Location (Letter indicates TIP machine)

● LPRM Location (Common location for all TIP machines)

A IRM Locations (Letter indicates IRM channel)

A SRM Locations (Letter indicates SRM channel)

* Source Locations

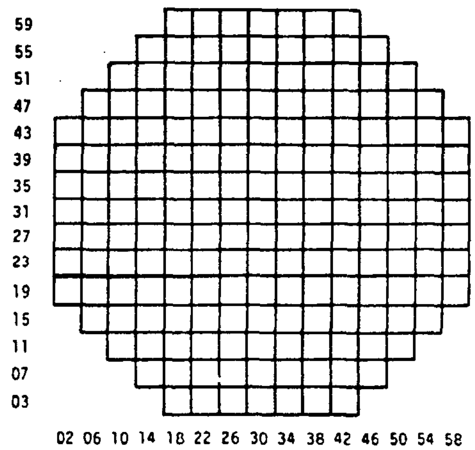


Figure 4-11 NMP2 Neutron Monitoring System

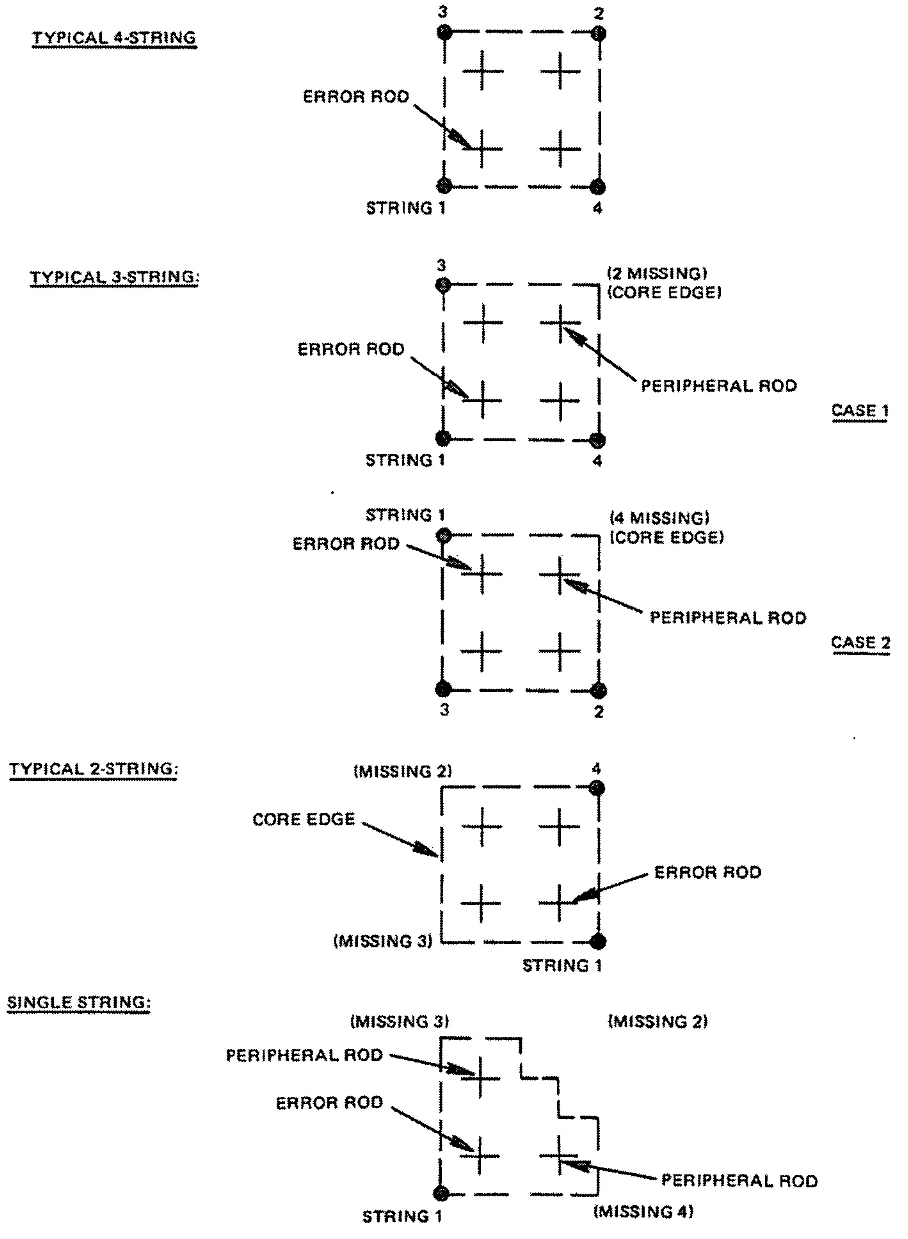


Figure 4-12 Rod Block Monitor Rod Group Geometries

[[

]]

Figure 4-13 Results of LPRM Failure Rate Sensitivity Studies

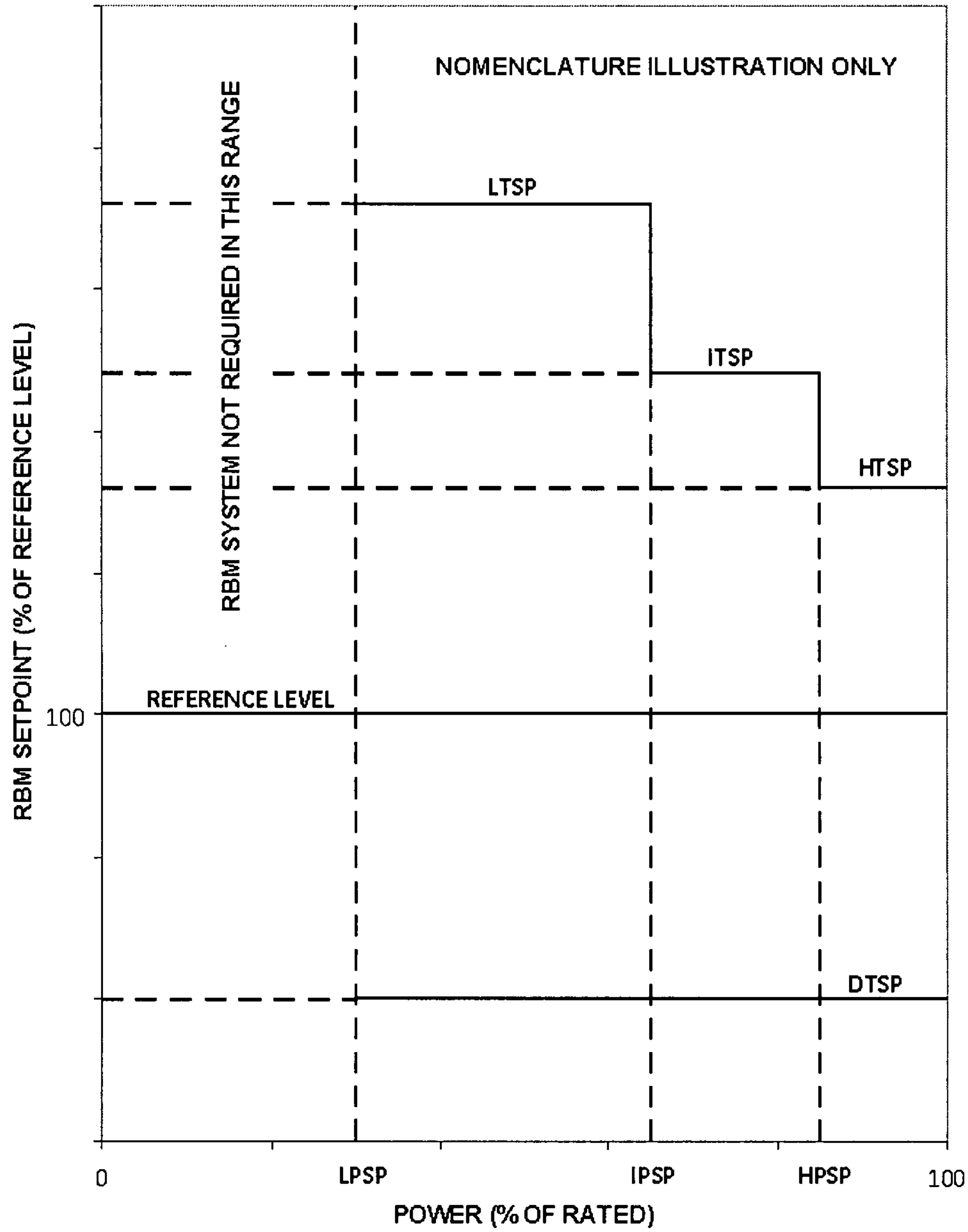


Figure 4-14 Power-Dependent RBM Trip Nomenclature

5.0 VESSEL OVERPRESSURE PROTECTION

The Main Steam Isolation Valve closure with a flux scram (MSIVF) event is used to determine the compliance to the ASME Pressure Vessel Code. This event was previously analyzed at the 102%P / 105%F state point for the NMP2 Cycle 11 reload licensing transient analysis. This is a cycle-specific calculation performed in accordance with Reference 5 at 102% of RTP and the maximum licensed core flow (maximum flow is limiting for this transient for NMP2). 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%F for this analysis) for NMP2 Cycle 11 is provided in Table 5-1.

Table 5-1 NMP2 Cycle 11 Sensitivity of Overpressure Analysis Results to Initial Flow

Initial Power / Flow (%Rated)	Peak Steam Dome Pressure (psig)	Peak Vessel Pressure (psig)
102 / 105	1270	1300
102 / 80	1270	1292

6.0 THERMAL-HYDRAULIC STABILITY

6.1 Introduction

The stability compliance of GE fuel designs with regulatory requirements of the NRC is documented in Section 9 of Reference 5. The NRC approval of the stability performance of GE fuel designs also includes operation in the MELLLA region of the power/flow 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. NMP2 has licensed Option III (Reference 6) as the long-term solution and has an approved Technical Specification for the Option III hardware. Reference 34 is the NRC approval of the NMP2 license amendment request for the Option III stability solution. The Option III hardware has been installed and connected to the Reactor Protection System. In the event that the Oscillation Power Range Monitor (OPRM) system is declared inoperable, NMP2 will operate under alternate methods.

The Option III detect and suppress stability solution has been implemented at NMP2. The demonstration calculations that are included in Sections 6.2 and 6.3 are based on the current Cycle 11 core design at the increased MELLLA power/flow map upper boundary. When the MELLLA upper boundary domain is implemented, cycle specific setpoints will be determined in accordance with Reference 5 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 Based Algorithm (GRBA), 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 is adequate. The Trip Enabled Region is shown in Figure 6-1.

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 11 core design at the increased MELLLA power/flow map upper boundary is determined per the NRC approved methodology (Reference 9). 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 CPR 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 BWROG Regional DIVOM Guideline (Reference 10). The analysis is performed with the Cycle 11 nominal core simulator wrapups at limiting conditions.

As shown in Table 6-1, with the rated OLMCPR of 1.39 and a SLMCPR of 1.07, an OPRM setpoint of 1.15 is the highest setpoint that may be used without stability setting the OLMCPR. The actual setpoint will be established in accordance with NMP2 Technical Specification 3.3.1.1.

Therefore, ARTS/MELLLA operation is justified for plant operation with stability Option III.

6.3 Backup Stability Protection

NMP2 implements the associated Backup Stability Protection (BSP) regions (Reference 8) as the stability licensing basis if the Option III OPRM system is declared inoperable. The stability interim corrective actions (ICAs) restrict plant operation in the high power, low core flow region of the BWR power/flow operating map. The ICAs provide guidance which reduces the likelihood of an instability event by limiting the period of operation in regions of the power and flow map most susceptible to thermal hydraulic instability. The ICAs also specify operator actions, which are capable of detecting conditions consistent with the onset of oscillations, and additional actions, which mitigate the consequences of oscillations consistent with degraded thermal hydraulic stability performance of the core.

The BSP regions consist of two regions (I-Scram and II-Controlled Entry), which are reduced from the three ICA regions (I-Scram, II-Exit and III-Controlled Entry) (References 7 and 8). The standard ICA region state points on the High Flow Control Line (HFCL) and on the Natural Circulation Line (NCL) define the base BSP region state points on the HFCL and on the NCL. The bounding plant-specific BSP region state points must enclose the corresponding base BSP region state points on the HFCL and the NCL. If a calculated BSP region state point is located inside the corresponding base BSP region state point, 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 is acceptable for use. 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 Generic Shape Function (GSF).

The BSP regions were expanded in the MELLLA region for power levels in excess of the original APRM Rod Block in accordance with the guidance in Reference 7. The demonstration of proposed BSP regions based on Cycle 11 is shown in Figure 6-2. The BSP regions as described in References 7 and 8 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

OPRM Setpoint	Δ_i*	DIVOM Slope = 0.481 OLMCPR(SS) MELLLA	DIVOM Slope = 0.481 OLMCPR(2RPT) MELLLA
1.05	0.171	1.166	1.087
1.06	0.204	1.186	1.106
1.07	0.236	1.207	1.125
1.08	0.269	1.229	1.146
1.09	0.301	1.251	1.167
1.10	0.334	1.275	1.188
1.11	0.365	1.298	1.210
1.12	0.396	1.322	1.232
1.13	0.427	1.347	1.255
1.14	0.458	1.372	1.279
1.15	0.489	1.399	1.304
		Off-rated OLMCPR @45% flow, estimated to be 1.613**	Rated Power OLMCPR, estimated to be 1.39

*Δ_i represents the Hot Channel Oscillation Magnitudes (Reference 11).

**Estimated value calculated by assuming that the Kf multiplier for ELLLA domain still applies.

Figure 6-1 MELLLA OPRM Trip Enabled Region

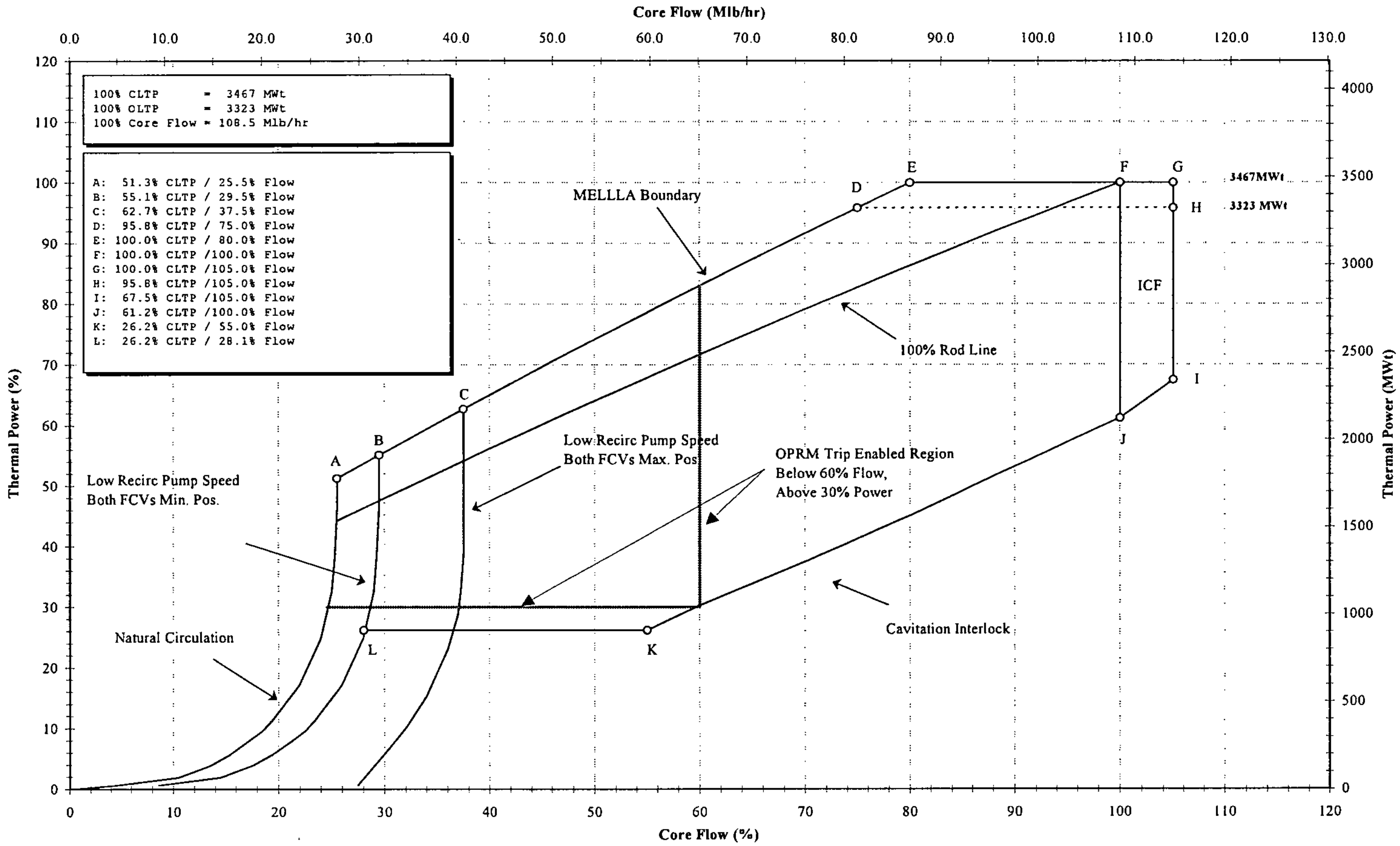
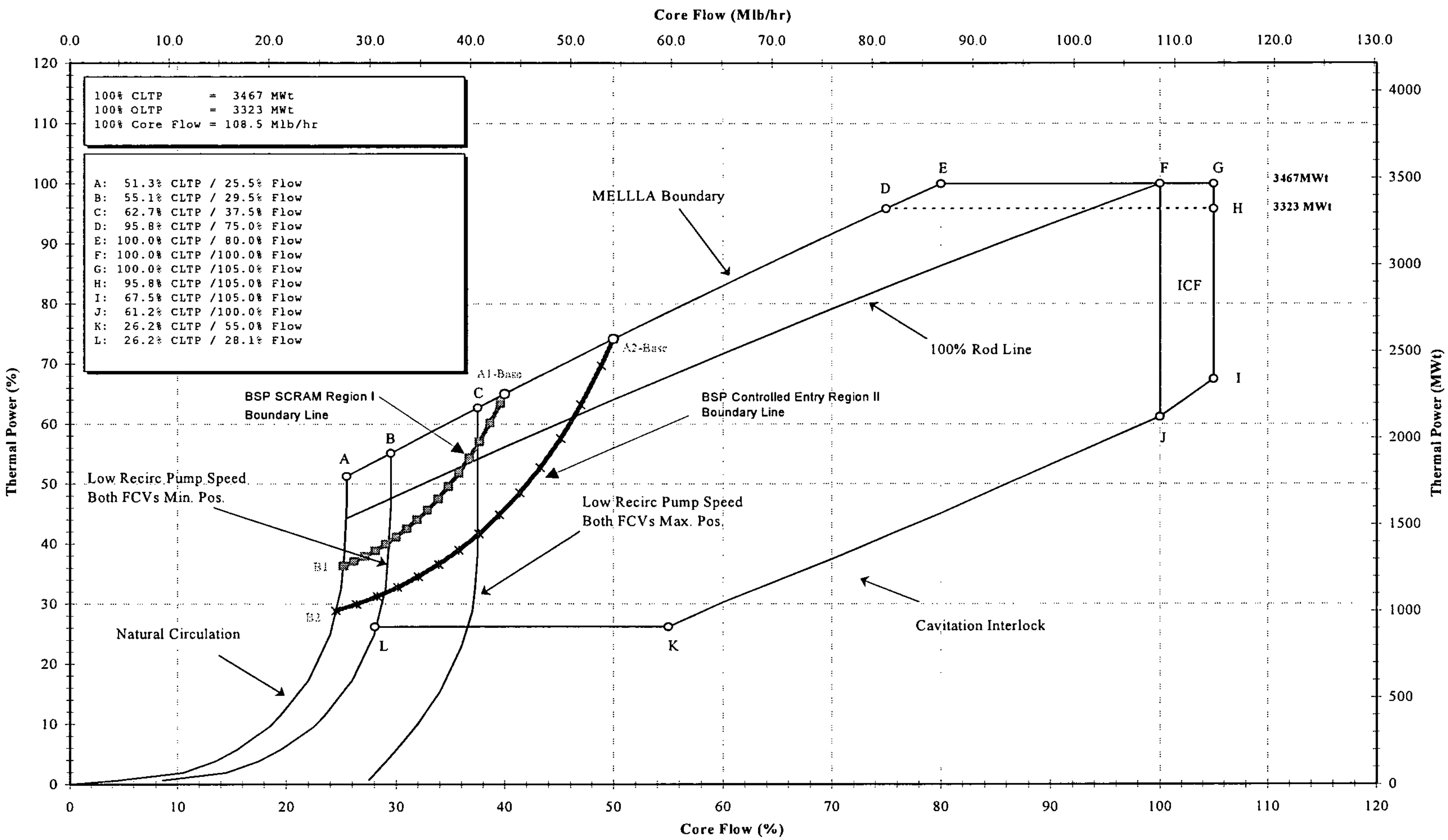


Figure 6-2 Demonstration of Proposed BSP Regions



7.0 LOSS-OF-COOLANT ACCIDENT ANALYSIS

The current licensing basis SAFER/GESTR-LOCA analysis for NMP2 (Reference 12 for GE14 fuel and Reference 13 for GE11 fuel) has been reviewed to determine the effect on the Emergency Core Cooling System (ECCS) performance resulting from NMP2 operation in the MELLLA domain. The Reference 12 and Reference 13 analyses considered NMP2 operation in the ELLLA domain. The loss of coolant accident analysis for NMP2 operation in the MELLLA domain are in conformance with the error reporting requirements of 10 CFR 50.46 through notification number 2006-01. Therefore, all known ECCS-LOCA analysis errors in accordance with 10CFR50.46 have been accounted for in the analysis in support of the application of ARTS-MELLLA for NMP2. The NMP2 current licensing basis peak cladding temperature (PCT) for GE14 fuel is shown in Table 7-2. This current licensing basis PCT is 1370°F and is set by the results of the LOCA analysis for the recirculation suction line break at CLTP/RCF with a mid-peaked axial power distribution.

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 power/flow 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 NMP2 licensing basis specifies a requirement in maximum LHGR as a function of drive flow, known as the APRM set down requirement. This lower LHGR requirement is applicable to core flows lower than 87% of rated core flow. 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. However, these limits are not credited in the LOCA analysis. Therefore, the LOCA analysis is not affected by the implementation of ARTS.

The nominal and Appendix K PCT response following a large recirculation line break for most plants show that the PCT effect due to MELLLA is small. In some cases, there may be a significant PCT increase if early boiling transition penetrates down to the highest-powered axial node in the fuel bundle. This can happen at core flows in the MELLLA region. [[

]] For small breaks, the fuel remains in nucleate boiling until uncovery and MELLLA is expected to have no adverse effect on the small break LOCA response.

Calculations assuming the MELLLA extended operation domain were performed to quantify the effect on PCT to the allowed operation envelope. The MELLLA assumptions for the limiting large recirculation line break case resulted in an [[

]]

MELLLA has a negligible effect on compliance with the other acceptance criteria of 10 CFR 50.46. Because cladding oxidation is primarily determined by PCT, MELLLA can affect the amount of cladding oxidation in those cases where there is a significant PCT increase. Jet pump BWRs have significant margin to the local cladding oxidation and core-wide metal-water reaction acceptance criteria, even for PCTs at the 2200°F limit. The compliance with the 2200°F limit ensures compliance with the local cladding oxidation and core-wide metal-water reaction acceptance criteria for GE14 fuel. Compliance with the coolable geometry and long-term cooling acceptance criteria were demonstrated generically for GE BWRs (Reference 14). 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 NMP2 MELLLA evaluation is based on plant specific calculations with GE14 fuel using SAFER/GESTR methodology (References 14 through 18). Calculations were performed for rated flow and power conditions to establish a baseline PCT, with model changes as have been identified since the last ECCS-LOCA analysis using the SAFER/GESTR methodology (Reference 12). 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. GE11 fuel remains resident in the core. [[
  ]]
- Restrictions imposed by the NRC on Upper Bound PCT (Reference 19) have been removed for NMP2 (Reference 39). The Upper Bound PCT has been shown bounded by the Licensing Basis PCT, consistent with the previous evaluation (Reference 12), and need not be recalculated for ARTS/MELLLA implementation.
- ECCS operation parameters are consistent with those used in the Reference 12 analysis, with one change. An increase to the feedwater temperature by 20 °F was applied.

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 NMP2 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 CLTP/RCF condition result in the highest PCT for both the small and large break LOCA and set the licensing basis PCT for NMP2. Calculations performed at the CLTP/ MELLLA Core flow condition result in lower PCT than the CLTP/RCF condition.

Table 7-1 ECCS-LOCA Analysis Bases for NMP2 ARTS/MELLLA

Parameter	Unit	Value
Original Licensed Thermal Power (OLTP)	MWt	3323
Current Licensed Thermal Power (CLTP)	MWt	3467
Vessel Steam Dome Pressure	psia	1055
Rated Core Flow	Mlb/hr	108.5
MELLLA Core Flow (80% rated flow at CLTP)	Mlb/hr	86.8

Table 7-2 ECCS-LOCA Peak Cladding Temperature for NMP2 ARTS/MELLLA

Case Description	Current Analysis PCT (°F) (Reference 12 - 403°F FWT)	Updated PCT (°F) (Rated Flow 423°F FWT)	PCT (°F) (MELLLA Flow 423°F FWT)
DBA Break:			
Appendix K Assumptions			
[[			]]
Small Break:			
Appendix K Assumptions			
[[			
			]]
Nominal Assumptions			
[[	]]		
[[		]]	
Licensing Basis PCT:	1370	1480	Note 2

Appendix K – 10.CFR50.46 Appendix K assumptions

ADS – Automatic Depressurization System Valves in service

Note 1 - Case Description that sets the licensing basis PCT

Note 2 – Licensing basis PCT is set by the Rated Flow 423°F FWT 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, condensation oscillation and chugging) for NMP2. The analysis presented here demonstrates that sufficient conservatism and margin in the containment hydrodynamic loads currently defined for NMP2 is available to compensate for any variance in these loads due to the extended operating domain, or that the currently defined loads are not affected. The SRV discharge load evaluation would normally consider any increases in the SRV opening setpoints. Because the ARTS/MELLLA operating domain does not require changes to the SRV setpoints, the pressure related SRV loads do not change.

The procedure used for this evaluation follows the methodology used to evaluate the containment LOCA hydrodynamic loads for the NMP2 4.3% Power Uprate in Reference 1.

8.1.1 Short-Term Pressure/Temperature Response

The short-term containment response covers the blowdown period during which the maximum drywell pressure, wetwell pressure, and maximum drywell to wetwell differential pressure occur. Consequently, analyses are performed for various cases that cover the full extent of NMP2 operation in the MELLLA domain. The objective of performing these analyses is to demonstrate that NMP2 operation in the MELLLA domain will not result in exceeding the containment design limits as stated in the NMP2 USAR. The results of these analyses are also used for evaluating the various containment hydrodynamic loads.

The short-term containment pressure and temperature response up to approximately 40 seconds for a DBA LOCA was analyzed for the following four cases:

1. 102.0% of RTP / 100.0% core flow
2. 102.0% of RTP / 105.0% core flow (ICF)
3. 102.0% of RTP / 80.0% core flow (MELLLA)
4. 56.2% of RTP / 29.5% core flow
   (Low Pump Speed and Minimum Recirculation Flow Control Valve Position)

The Minimum Flow Control Valve (MFCV) -MELLLA point is performed at a power level of 56.2% of 3467 MWt.

These cases were selected to conservatively cover the full extent of the MELLLA power/flow boundary including the ICF region. The ICF condition is included to provide a consistent set of MELLLA and ICF analyses that can be used to evaluate the effect of the different operating conditions.

8.1.2 LOCA Containment Hydrodynamic Loads

The NMP2 LOCA containment hydrodynamic loads assessment includes the following:

- Pool Swell (PS)
- Condensation Oscillation (CO)
- Chugging (CH)

The PS, CO, and CH 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 previously defined using the generic methodology from the Mark II Containment Program as described in Reference 20 and accepted by the NRC in References 21 and 22. The plant-specific dynamic loads are also defined in the Design Assessment Report (DAR) for NMP2 (Reference 23). The current evaluation of these loads to NMP2 is described in the Loads Reports for power uprate (Reference 1).

The containment hydrodynamic loads analyses presented in this section also include considerations of the currently licensed 20°F Feedwater Heater Out-of-Service (FWHOOS) and future applications for 120°F Final Feedwater Temperature Reduction (FFWTR) and FWHOOS.

8.2 Assumptions and Initial Conditions

The following initial containment conditions are used in the DBA-LOCA short-term containment pressure/temperature response analysis.

Parameter	Value
Drywell Pressure (psig)	0.75
Wetwell Pressure (psig)	0.75
Drywell Temperature (°F)	105
Suppression Pool Temperature (°F)	90
Drywell humidity (%)	40%
Wetwell humidity (%)	100%

These initial conditions are identical to those assumed in the current design basis DBA-LOCA short-term containment pressure/temperature response analysis of Reference 1, with the single exception that an initial drywell temperature of 135°F was assumed in the Reference 1 analysis, whereas an initial drywell temperature of 105°F is assumed for the MELLLA analysis. [[

]]

Initial containment conditions used for assessment of hydrodynamic loads are identical to the initial containment conditions used for the DBA-LOCA short-term containment pressure/temperature response analysis shown in the table above, with the single exception that an initial drywell temperature of 135°F is used for hydrodynamic loads assessment. The initial containment conditions used for assessment of hydrodynamic loads are therefore consistent with the current NMP2 design basis hydrodynamic loads evaluation of the NMP2 DAR (Reference 23).

The assumptions used in the evaluation of the short-term containment response for NMP2 operation in the ARTS/MELLLA domain are the same as those used in the current evaluation of Reference 1. Some of the important assumptions are also listed below.

1. Reactor power generation is assumed to cease concurrently with the time of the accident initiation. There is no delay period.
2. The break being analyzed is an instantaneous double-ended rupture of a recirculation suction line. This results in the maximum discharge rates to the drywell.
3. GE's LAMB computer code (Reference 27) is used to calculate the break flow rates and break enthalpies. These values are then used as inputs to the M3CPT computer code (References 28 and 29) to calculate the containment pressure and temperature response.
4. The vessel blowdown flow rates are based on the Moody Slip flow model. (Reference 30)
5. The 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. This assumption maximizes the wetwell airspace temperature and pressure.
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 0.0 seconds and entirely stop at 4.0 seconds. This assumption is consistent with the current analysis of the USAR.

8.3 Analyses Results

8.3.1 Short-Term Pressure/Temperature Response

The four cases listed in Table 8-1 were analyzed to determine the short-term peak containment pressure and temperature response. Table 8-2 illustrates the peak containment results compared to the containment design limits. The results show that the peak values are similar for most of the cases. This is a result of the break flow and enthalpy inputs, which provide a similar integrated energy input forcing function into the drywell. The limiting case (with respect to peak pressures and temperatures) occurs for the reactor condition with 102% of the RTP and 105% rated core flow.

It is observed from the results summary presented in Table 8-2 that, for peak drywell pressure and temperature and peak wetwell pressure, the two MELLLA cases are bounded by the Normal and Increased Core Flow (ICF) cases. It is also observed from the results presented in Table 8-2 that the peak downward slab differential pressure (18.6 psid) is bounded by the current USAR analysis. All peak values are within the design limits reported in the NMP2 USAR.

8.3.2 LOCA Containment Hydrodynamic Loads

Three types of hydrodynamic loads are addressed for the DBA-LOCA: a) pool swell loads, b) condensation oscillation loads, and c) chugging loads. The impact of ARTS/MELLLA on these loads is evaluated by comparing the pressure and temperature responses with those used in the load definitions for NMP2 Design Assessment Report (Reference 23).

8.3.2.1 Pool Swell (PS)

The pool swell loads include the vent clearing loads, the LOCA bubble wall pressure and submerged structure loads, wetwell airspace pressurization and the pool swell impact and drag loads. All of these loads are controlled by the initial drywell pressurization (first 2 seconds) following the initiation of the DBA-LOCA. The drywell pressure response used in the pool swell design load analysis is presented in Section 6A.4 of Reference 23, and the pressurization exhibited by this pressure response bounds the initial drywell pressurization predicted for ARTS/MELLLA. Therefore, it can be concluded that the ARTS/MELLLA operation has no adverse impact on the pool swell loads.

8.3.2.2 Condensation Oscillation (CO)

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 pool swell 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 NMP2 CO loads are based on Mark II 4TCO tests (Reference 24) and the generic Mark II CO load definition (Reference 25). 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 NMP2. The generic Mark II CO load was defined, enveloping all applicable test data obtained with pool temperature up to 160°F. A plant-specific NMP2 CO load definition was developed, enveloping all 4TCO test data for suppression pool temperatures equal to or less than 130°F. This definition was based on the calculated DBA-LOCA suppression pool temperature response that showed the end of CO load at 119°F pool temperature. This represented an 11°F margin to the 130°F pool temperature threshold for the CO load definition, which was approved by the NRC in Reference 26. The current analysis shows that for all cases the condition used to define CO load ends before the pool temperature reaches 120°F and the steam mass flux is below the steam mass flux values used in the 4TCO tests. Thus, it is concluded that ARTS/MELLLA has no impact on the current CO load definition for NMP2.

8.3.2.3 Chugging (CH)

The chugging load definition for NMP2 was based on the data from the chugging tests that covered thermal-hydraulic conditions expected for a Mark II containment geometry. Furthermore, chugging occurs when steam mass flux through the vent is not high enough to maintain a steady steam/water interface at the vent exit. Consequently, chugging occurs at the tail end of the DBA-LOCA or Intermediate Break Accident (IBA), or during a Small Break Accident (SBA) with the reactor at pressure. This means that chugging occurs during the LOCA phase when ARTS/MELLLA has a negligible impact on the containment pressure and temperature response. Therefore, it is concluded that the current chugging load definition for NMP2 is not impacted by ARTS/MELLLA.

8.4 Conclusions

It is concluded that ARTS/MELLLA has no adverse impact on the current NMP2 definition of the dynamic loads of (1) pool swell, (2) condensation oscillation and (3) chugging and that the existing definitions of LOCA dynamic loads of pool swell, condensation oscillation, and chugging for NMP2 remain applicable for ARTS/MELLLA.

8.5 Reactor Asymmetric Loads

In support of MELLLA implementation, the impact of expanding the reactor operating domain from the current ELLLA power/flow map boundary to the MELLLA power/flow map boundary on high-energy line break mass and energy releases to the annulus region (reactor pressure vessel (RPV) to shield wall) were evaluated. The evaluation was performed over the range of power / flow conditions associated with the MELLLA boundary.

8.5.1 Annulus Pressurization Analysis

The methodology for calculating blowdown mass and energy release profiles for AP loads analysis is the conservative methodology documented in Reference 32. For the MELLLA evaluation, relative changes in mass and energy release rates, associated with MELLLA related power-flow map changes are used the determine the impact of MELLLA on compartment differential pressures.

For the feedwater line break, MELLLA implementation will result in a compartment differential pressure increase of less than 2.25 percent.

For breaks other than the feedwater line, MELLLA implementation will result in an increase in compartment differential pressure of as much as 6.8 percent for full power conditions and as much as 3.0 percent in the vicinity of the minimum flow point on the MELLLA line.

8.5.2 Effect on Structural Response due to Reactor Asymmetric Loads

The increase in reactor asymmetric loads due to high-energy pipe breaks, other than the feedwater line break, between the RPV and the biological shield wall associated with MELLLA implementation is less than 6.8 percent for full power conditions and less than 3.0 percent in the vicinity of the minimum flow point on the MELLLA boundary. The change in reactor asymmetric loads due to a feedwater line break between the RPV and the biological shield wall within the RTP/MELLLA operating domain is less than 2.25 percent higher than the reactor asymmetric loads within the RTP/ELLLA operating domain. The increases are due to the slightly higher integrated energy releases that occur for the pipe breaks at MELLLA operating conditions. For the structural evaluation of the biological shield wall, RPV, CRD mechanism and piping systems that are connected to the RPV and penetrate the biological shield wall, the reactor asymmetric loads are combined with other loads, e.g. seismic loads, for comparison to allowable stress limits. An assessment of the effect of the increase in asymmetric loads was performed by comparing the existing NMP2 stress margins to the resultant margins if a conservative reduction in stress margin was assumed due to the increase in the asymmetric loads at RTP/MELLLA conditions. The result of this assessment concludes that sufficient margin exists for all components due to the increase in the asymmetric loads at RTP/MELLLA.

The structural evaluation of RPV internals is contained in Section 9.3.

Table 8-1 Cases Analyzed For Short-Term Containment Response

Case No.	Point	Power (MWt)	Core Flow (Mlbm/hr)	Feedwater Inlet Temperature (°F)	Dome Pressure (psia)
1	102%P/100%F (Rated)	3536.3	108.50	427.4	1055.0
2	102%P/105%F (ICF)	3536.3	113.92	427.4	1055.0
3	102%P/80%F (MELLLA)	3536.3	86.80	427.4	1055.0
4	56.2%P/29.5%F (Low Pump Speed MFCV-MELLLA)	1948.5	32.01	365.5	1000.0

Table 8-2 Summary of Sensitivity Study Results for Peak Drywell Pressure and Temperature and Initial Drywell Pressurization Rate

	Drywell Pressure (psig)	Wetwell Pressure (psig)	Differential Pressure (psid)	Drywell Temperature (°F)	Wetwell Temperature (°F)
Design Limit	45.0	45.0	25.0	340	270
Current	36.8	31.8	18.1	N/R	N/R
Current†	33.0	26.9	18.1	278	116
Rated‡	32.8	26.9	18.2	278	116
Rated	34.9	29.1	18.6	281	116
ICF	34.9	29.2	18.6	281	116
MELLLA	34.8	29.1	18.1	281	115
MELLLA-MPS	34.3	28.6	16.3	280	112

† Values shown for this row are limited to the first 40 seconds of the event. These are provided for comparison with the results of the following analysis.

‡ This case assumes an initial drywell temperature of 135°F similar to the Current cases. [[

]].

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 ELLLA (87% of RCF) and the ICF (105% of RCF) conditions due to the lower core flow condition in MELLLA (80% of RCF). Thus, no new RIPDs, fuel bundle lift and control rod guide tube (CRGT) conditions are generated by the MELLLA operating domain. The current RIPD basis remains applicable to the MELLLA condition.

9.2 Acoustic and Flow-Induced Loads

The acoustic and flow-induced loads are contributing factors to the NMP2 design basis load combination in the Faulted condition. The acoustic loads are imposed on the reactor internal structures as a result of the propagation of the decompression wave created by the assumption of an instantaneous Recirculation Suction Line Break (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 NMP2 geometry and fluid conditions applied to a reference BWR calculation. The loads were calculated by applying scaling factors that account for plant-specific geometry differences (e.g., size of the shroud, reactor vessel, and recirculation line) and thermal-hydraulic condition differences (e.g., downcomer subcooling) from the reference plant. The reference calculation was based on the GE methods utilized to support NRC Generic Letter 94-03 that was issued to address the shroud cracks detected at some BWRs.

The acoustic loads on the jet pumps and shroud applied for NMP2 represent NMP2-specific plant geometry configuration and operating conditions. The bounding natural frequencies for the jet pumps and shroud along with the bounding subcooling are applied. For acoustic loads on the shroud support, generic bounding BWR loads based on the GE approved methods were used. For NMP2, the most limiting subcooling condition is at the intersection of the minimum pump speed and the MELLLA or ELLLA maximum power boundary line. The initial thermal hydraulic conditions including the subcooling at this point are applied to the reference BWR calculation, along with the NMP2 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 NMP2 current licensing basis.
102%P / 100% F	Consistent with the NMP2 current licensing basis with feedwater temperature reduction.
102%P / 80%F	MELLLA corner at rated power with feedwater temperature reduction.
62.7%P / 37.5%F	Minimum pump speed point on the MELLLA boundary line, flow control valves maximum open position, with feedwater temperature reduction.
55.1%P / 29.5%F	Minimum pump speed point on the MELLLA boundary line, flow control valves minimum open position, with feedwater temperature reduction.

9.2.3 Results

The flow-induced loads for the shroud and jet pumps are shown in Table 9-1. NMP2-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% RCF) are slightly higher than the current uprated ELLLA condition (at the CLTP and 87% RCF) due to the increased subcooling in the downcomer associated with the MELLLA condition. From ELLLA to MELLLA, the downcomer subcooling increases thereby increasing the critical flow and the mass flux out of the break in a postulated RSLB. As a result, the flow-induced loads in MELLLA conditions increase slightly.

The acoustic loads in the MELLLA condition (at the CLTP and 80% RCF) equal the current ELLLA condition (at the CLTP and 87% RCF) because there are no vessel geometry changes associated with the MELLLA operating domain.

9.3 Reactor Pressure Vessel Internals Structural Integrity Evaluation

The structural integrity of the RPV internals was qualitatively evaluated for the loads associated with MELLLA operation for NMP2 considering the current design basis evaluations (Reference 39) at CLTP with FFWTR operation. The current design basis evaluations bound NMP2 operation with the currently licensed 20°F FWTR. The loads considered for MELLLA include Dead weights, Seismic Loads, RIPDs, Acoustic and Flow induced Loads due to Recirculation Suction Line Break Loss of Coolant Accident, SRV, Annulus Pressurization (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 are not certified to the

ASME Code; however, the requirements of the ASME Code are used as guidelines in their design basis analysis. The following RPV internal components were evaluated:

- Shroud
- Shroud support
- Core Plate
- Top Guide
- CRD Housing
- Control Rod Guide Tube
- Orificed Fuel Support
- Fuel channel
- Shroud Head and Separator Assembly
- Jet Pump
- 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. Because all applicable loads are unaffected, remain bounded, or change insignificantly with respect to CLTP with FFWTR, the above RPV Internal components remain qualified for ARTS/MELLLA conditions. 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 42). All other reactors of the same product line and size are classified as non-prototype and undergo a less rigorous confirmatory test.

NMP2 is currently licensed to operate at an ICF of up to 105% of RCF (108.5 Mlbs/hr) at 100% of CLTP. For MELLLA operation, the rated power output remains the same, but core flow is reduced to 80% 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. 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 NMP2 and data collected during plant start-up between November 1987 and March 1988. The critical reactor internals were instrumented with vibration sensors and the reactor was tested up to 100% core flow at 100% rod line. These data were used in the current NMP2 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 3467 MWt and 80% 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, 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, NMP2 and other plants if needed were reviewed to determine which components are likely to have significant vibration at the MELLLA conditions.

For jet pump sensing lines, since NMP2 is a FCV plant, the vane passing frequency (VPF) does not change due to MELLLA and hence there is no change from existing conditions.

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

Because the vibration levels are generally proportional to the square of the flow, the lower plenum components (Control Rod Guide Tube (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.

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

The FIV evaluation is conservative for the following reasons:

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

Therefore, the FIV will remain within acceptable limits.

9.5 Conclusion

The analyses documented in this section demonstrate that, from an FIV viewpoint, the reactor internals structural mechanical integrity is maintained to provide NMP2 safe operation in the MELLLA domain.

Table 9-1 Flow-Induced Loads on Shroud and Jet Pumps for NMP2

Component	Parameter	Loads (a)
Shroud	Baseline Force (kips)	93.090
	Baseline Moment at the Shroud Centerline (10^6 in-lbf)	8.076
Jet Pump	Baseline Force (kips)	5.884
	Baseline Moment at the Jet Pump Centerline (10^6 in-lbf)	0.340
Component	**Operating Condition**	**Load Multiplier (b)**
Jet Pump /Shroud	102%P / 100%F	[[
	102%P / 100%F FWTR	
	102 %P / 80%F (MELLLA) FWTR	
	62.7%P / 37.5%F FWTR	
	55.1%P / 29.5%F FWTR	]]

Note: (a) Loads at rated conditions (102% power/100% core flow).
(b) Loads multipliers [[ ]] in critical break flow assumption.

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

Component	Conditions	Force (kips)	Effective Force (kips)	Moment (106 in-lbf)	Effective Moment (106 in-lbf)
Shroud	55.1%P / 29.5%F FWTR	2080.0	1010.0	296.0	117.0
Jet Pump	55.1%P / 29.5%F FWTR	28.1	23.9	1.38	1.36

Table 9-3 Maximum Acoustic Loads on Shroud Support (MELLLA)

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

Table 9-4 RPV Internals Structural Evaluation Results

No	Component	Service Level	Critical Parameter, Units	CLTP with FFWTR Value	ARTS/ MELLLA Value	Allowable Value
1	Shroud	Upset	Buckling, ksi	3.58	Bounded by CLTP with FFWTR	6.25
2	Shroud support	Faulted	Vertical Load, (kips)	12,066	Bounded by CLTP with FFWTR	14,585
3	Core Plate	Upset	Buckling Load /Guide tube, kips	1.99	Bounded by CLTP with FFWTR	2.03
4	Top Guide	Faulted	(Pm+Pb), ksi	42.81	Bounded by CLTP with FFWTR	50.7
5	Control Rod Drive Housing	Faulted	(Pm), ksi	15.63	15.72	19.92
6	Control Rod Guide Tube	Upset	(Pm+Pb), ksi	15.06	Bounded by CLTP with FFWTR	24.00
7	Fuel Channel		Qualified, Note 1			
8	Orificed Fuel Support	Upset	Horizontal Load, (kips)	2.11	Bounded by CLTP with FFWTR	4.495
		Faulted	Vertical Load (kips)	8.04	Bounded by CLTP with FFWTR	90.24
9	Jet Pump	Faulted	(Pm+Pb), ksi	41,03	Bounded by CLTP with FFWTR	60,64
10	Core Spray Line & Sparger	Upset	Pm, ksi	16.70	Bounded by CLTP with FFWTR	20.92
11	Access Hole Cover	Faulted	(Pm+Pb), ksi	24.96	Bounded by CLTP with FFWTR	69.90
12	Shroud Head and Steam Separators Assembly	Faulted	Bearing Stress, ksi	18.36	Bounded by CLTP with FFWTR	37.60
13	Feedwater Sparger	Normal /Upset	Fatigue Usage	0.88	Bounded by CLTP with FFWTR	1.00
14	Steam Dryer	Faulted	Lifting Rod Buckling Load, Kips	75.18	Bounded by CLTP with FFWTR	88.99
15	In-core Housing & Guide Tube	Upset	Pm, ksi	16.58	Bounded by CLTP with FFWTR	16.66
16	Core ΔP & Liquid Control Line	Upset	Horizontal Load, (kips)	<46.32	Bounded by CLTP with FFWTR	49.20
17	LPCI Coupling	Emergency	Vertical Load (kips)	22.32	Bounded by CLTP with FFWTR	38.03

Note: 1. Fuel Channel remains qualified for MELLLA conditions. The RIPDs under all operating conditions are bounded by those for CLTP with FFWTR.

Pm – Primary Membrane Stress

Pb – Primary Bending Stress

10.0 ANTICIPATED TRANSIENT WITHOUT SCRAM

10.1 Approach/Methodology

The basis for the current ATWS requirements is 10CFR50.62. This regulation includes requirements for an ATWS Recirculation Pump Trip (RPT), an Alternate Rod Insertion (ARI) system, and an adequate Standby Liquid Control System (SLCS) injection rate. The purpose of the ATWS analysis is to demonstrate that these systems are adequate for operation in the MELLLA region. This is accomplished by performing a plant-specific analysis in accordance with the approved licensing methodology (Reference 33) 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 NMP2 were re-evaluated at the ELLLA point (100% of CLTP and 87% of RCF) and the MELLLA point (100% of CLTP and 80% of RCF) with ARI assumed to fail, thus requiring the operator to initiate SLCS injection for shutdown. These events were: (1) Closure of all MSIVs (MSIVC), (2) Pressure Regulator Failure (Open) to Maximum Steam Demand Flow (PRFO), and (3) Loss of Offsite Power (LOOP).

The MSIVC and PRFO events are the most limiting events for the ATWS acceptance criteria. The MSIVC and PRFO events result in reactor isolation and a large power increase without scram. Due to the reactor isolation, a high demand on the dual safety/relief valves (DS/RVs), i.e. SRVs, is required to mitigate the large pressure increase, with all of the energy being directed to containment and suppression pool. These factors combine to make the MSIVC and PRFO events the most limiting for fuel integrity, RPV integrity, and containment integrity.

The LOOP event is limiting in terms of the maximum SLCS pump discharge pressure during an ATWS event. The LOOP event results in a loss of instrument gas required to operate the DS/RVs in relief mode. Therefore, the DS/RV relief mode is modeled as available for only one open/close cycle. Subsequent DS/RV actuations are in safety mode, which maximizes upper plenum pressure after SLCS initiation. The LOOP event is non-limiting with respect to the ATWS acceptance criteria due to the recirculation and feedwater pump coastdown at time zero, which reduces the severity of the initial power increase.

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 since the main condenser remains available for most of the event. RPV and fuel integrities are not challenged since the vessel is shutdown (via boron injection) by the time the MSIVs isolate.

The following ATWS acceptance criteria were used to determine acceptability of the NMP2 operation in the MELLLA region (based on the results of the MSIVC and PRFO transients – the LOOP is non-limiting for these evaluations):

1. Fuel integrity:
   - Maximum clad temperature < 2200° F
   - Maximum local clad oxidation < 17%
2. RPV integrity:
   - Peak RPV pressure < 1500 psig (ASME service level C)
3. Containment integrity:
   - Peak suppression pool bulk temperature < 190°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 3467 MWt (100% of CLTP)	Consistency with NMP2 current licensing basis
Both beginning-of-cycle (BOC) and end-of-cycle (EOC) nuclear dynamic parameters were used in the calculations	Consistency with generic ATWS evaluation bases
Dynamic void and Doppler reactivity are based on NMP2 Cycle 11 data	ATWS analyses are performed conservatively compared to a nominal basis, which bounds cycle to cycle variation
Two SRV OOS, specified as the valves with the lowest setpoints	Consistency with the NMP2 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 LOOP event is assumed to be a loss of all auxiliary power transformers at event initiation	Consistency with generic ATWS evaluation bases

10.3 Analyses Results

Table 10-2 presents the results for the MSIVC and PRFO events. As shown, the peak vessel bottom pressure for this event is 1295 psig, which is below the ATWS vessel overpressure protection criterion of 1500 psig.

The highest calculated peak suppression pool temperature is 155°F, which is below the ATWS limit of 190°F. The highest calculated peak containment pressure is < 6.0 psig, which is below the ATWS limit of 45 psig. Thus, the containment criteria for ATWS are met.

Coolable core geometry is ensured by meeting the 2200°F PCT and the 17% local cladding oxidation acceptance criteria of 10 CFR 50.46. The highest calculated PCT is 1563°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 is approximately 1325.4 psig. This value is based on a peak reactor vessel upper plenum pressure of 1220.3 psig that occurs during the limiting ATWS event after SLCS initiation.

The relief valves used for the SLCS at NMP2 currently have a nominal trip setpoint of 1394 psig and a maximum setpoint drift of -3%, resulting in a lower analytical setpoint of 1352.2 psig. There is approximately 26.8 psi margin between the maximum SLCS discharge pressure of 1325.4 psig and the lower analytical setpoint of 1352.2 psig. A margin of 30 psi from the relief valve lower analytical setpoint is adequate to accommodate the SLCS pump pressure pulsation. In order to provide the minimum 30 psi margin to accommodate the SLC pump pressure pulsation, the SLCS relief valve setpoints will be raised to 1400 psig prior to the implementation of MELLLA. With the SLCS relief valve setpoints raised to 1400 psig, the resultant 32.6 psi margin from the lower analytical setpoint is adequate to prevent the SLCS relief valve from lifting during SLCS operation to meet the guidelines in NRC Information Notice 2001-13.

10.4 Conclusions

The results of the ATWS analysis performed for NMP2 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, NMP2 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
ELLLA Core Flow (Mlbm/hr / % rated)	94.4 / 87.0
MELLLA Core Flow (Mlbm/hr / % rated)	86.8 / 80.0
Core Thermal Power (MWt / %CLTP)	3467 / 100.0
Steam / Feed Flow (Mlbm/hr / %NBR)	15.01 / 100
Sodium Pentaborate Solution Concentration in the SLCS Storage Tank (% by weight)	13.6
Boron 10 Enrichment (atom %)	25.0
SLCS Injection Location	High Pressure Core Spray (HPCS)
Number of SLCS Pumps Operating	2
SLCS Injection Rate (gpm)	82.4
SLCS Liquid Transport Time (sec)	120
Initial Suppression Pool Liquid Volume (ft^3)	145200
Initial Suppression Pool Temperature (°F)	90
Number of Residual Heat Removal (RHR) Cooling Loops	2
RHR Heat Exchanger Effectiveness (Btu/sec-°F)	249.0
Service Water Temperature (°F)	84
Transient time at which the RHR suppression pool cooling is initiated (seconds)	1080
High Dome Pressure ATWS-RPT Setpoint (psig)	1095
SRV Capacity – per valve (lbm/hr) / Reference Pressure (psig) / Accumulation (%)	890371 / 1145 / 3
SRV Configuration	18 DS/RV (2 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	1283	1287	1286	1295
Peak Cladding Temperature (°F)	2200	1063	1527	1330	1563
Peak Local Cladding Oxidation (%)	17	< 17	< 17	< 17	< 17
Peak Suppression Pool Temperature (°F)	190	155	155	155	155
Peak Containment Pressure (psig)	45	< 6	< 6	< 6	< 6

11.0 STEAM DRYER AND SEPARATOR PERFORMANCE

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

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

12.0 HIGH ENERGY LINE BREAK

The following high energy line breaks (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).
- Instrument Liquid 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 NMP2 design basis. Analyses were performed at rated conditions, and MELLLA conditions at low Reactor Recirculation System (RRS) pump speed with FCV at minimum position for the break locations listed above, taking into account the changes in enthalpy and pressure at each operating condition.

The mass and energy releases at the MELLLA state points for the MSLB in the main steam tunnel, the RCIC line break, and the Instrument Liquid Line breaks were found to be unchanged from the HELB mass / energy release profiles assumed in the current NMP2 design basis.

The mass and energy release profiles assumed in the current NMP2 design basis HELB analyses for the FWLB and the RWCU line breaks are not bounding at the break location for the MELLLA conditions listed above.

Nine Mile Point Nuclear Station evaluated the effects of the higher mass and energy release profiles and concluded the resulting subcompartment pressures, temperatures and humidity levels are acceptable with respect to 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, Maximum Linear Heat Generation Rate (MLHGR), 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 power/flow 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 power/flow conditions.

14.0 REFERENCES

1. NEDC-31994P, Revision 1, Nine Mile Point Nuclear Station Unit 2, Power Uprate Licensing Evaluation for Power Uprate Nine Mile Point Nuclear Station Unit 2, May 1993, Including E&A No. 1 September 1994 and E&A No. 2 November 1994.
2. "Supplemental Reload Licensing Report for Nine Mile Point Nuclear Station Unit 2 Reload 10, Cycle 11," 0000-0023-4708-SRLR, Revision 0, February 2006.
3. NMP2 Updated Safety Analysis Report, Revision 17, October 2006.
4. Issuance of Amendment for Nine Mile Point Nuclear Station Unit 2, Amendment No. 80, March 31, 1998.
5. GESTAR II General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-15, September 2005.
6. NEDO-31960-A and NEDO-31960-A Supplement 1, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995.
7. BWROG-94078, "BWR Owners' Group Guidelines for Stability Interim Corrective Action," June 6, 1994.
8. GE to BWR Owners' Group Detect and Suppress II Committee, "Backup Stability Protection (BSP) for Inoperable Option III Solution," OG 02-0119-260, July 17, 2002.
9. NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.
10. "Plant-Specific Regional Mode DIVOM Procedure Guideline," GE-NE-0000-0028-9714-R1, June 2005.
11. GENE–A13–00381–05, "Licensing Basis Hot Bundle Oscillation Magnitude for Nine Mile Point 2," Rev. 1, April 1998.
12. GE-NE-0000-0024-6517-R0, "Nine Mile Point-2 SAFER/GESTR Loss-of-Coolant Accident Analysis for GE14 Fuel," March 2004.
13. GE–NE–J1103938–07–02P, Nine Mile Point Unit 2 ECCS–LOCA Evaluation Update for GE11, January 2002.
14. NEDO-20566A, "General Electric Model for LOCA Analysis in Accordance with 10 CFR 50 Appendix K," September 1986.
15. Letter, C.O. Thomas (NRC) to J.F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-23785, Revision 1, Volume III (P), `The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident,'" June 1, 1984.
16. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," Revision 1, October 1984.

17. NEDC-23785-2-P, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident; Volume I, GESTR-LOCA - A Model for the Prediction of Fuel Rod Thermal Performance," May 1984.
18. NEDC-32950P, "Compilation of Improvements to GENE's SAFER ECCS-LOCA Evaluation Model," January 2000.
19. NEDE-23785P-A, Vol. III, Supplement 1, Revision 1, "GESTR-LOCA and SAFER Models for Evaluation of Loss-of-Coolant Accident Volume III, Supplement 1, Additional Information for Upper Bound PCT Calculation," March 2002.
20. NEDO-21061, General Electric Company, "Mark II Containment Dynamic Forcing Functions Information Report," Revision 4, November 1981.
21. NUREG-0487, U.S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," October 1978, Supplement 1, September 1980, and Supplement 2, February 1981.
22. NUREG-0808, U.S. Nuclear Regulatory Commission, "Mark II Containment Program Load Evaluation and Acceptance Criteria," August 1981.
23. Nine Mile Point Nuclear Station Unit 2, Design Assessment Report, NMP2 USAR Appendix 6A, Revision 8, 1995.
24. NEDE-24811-P, "4T Condensation Oscillation (4TCO) Test Program, Final Test Report," May 1980.
25. NEDE-24288-P, "Mark II Containment Program, Generic Condensation Oscillation Definition Report," November 1980.
26. US NRC Docket No. 50-410, "Safety Evaluation Concerning Downcomers in Nine Mile Point, Unit 2 (TAC 63425)," June 29, 1988.
27. NEDE-20566-P-A, "General Electric Model for LOCA Analysis in Accordance with 10CFR50 Appendix K," September 1986.
28. NEDO-10320, "The GE Pressure Suppression Containment Analytical Model," April 1971.
29. NEDO-20533, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," June 1974.
30. APED-4827, "Maximum Two-Phase Vessel Blowdown from Pipes," F. J. Moody, April 20, 1965.
31. GE-NE-187-08-1090, Rev. 1, "Balance of Plant Summary Report Prepared for Nine Mile Point Nuclear Station-Unit 2 Power Uprate (3467 MWt)," July 1992.
32. GE Nuclear Energy, "Technical Description: Annulus Pressurization Load Adequacy Evaluation," NEDO-24548, January 1979.
33. NEDC-24154P-A, "Qualification of the One Dimensional Core Transient Model (ODYN) for Boiling Water Reactors (Supplement 1 – Volume 4)," February 2000.

34. Letter NRC to J. Mueller, Subject: Nine Mile Point Nuclear Station, Unit 2 – Issuance of Amendment Re: Oscillation Power Range Neutron Monitor System (TAC No. MA7119). March 2, 2000.

35. Deleted

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

37. J. M. Healzer, J. E. Hench, E. Janssen, and S. Levy, "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," APED-5186, Class II, 1966.

38. NEDC-32410P-A, Class III, October 1995, Licensing Topical Report: Nuclear Measurement Analysis and Control Power Range Neutron Monitor Retrofit Plus Option III Stability Trip Function, Volumes 1 and 2.

39. GE-NE-0000-0016-5640-00, "GE14 Fuel Design Cycle-Independent Analyses For Nine Mile Point Unit 2," Revision 0, Class III, March 2004.

40. Letter from GE to US Nuclear Regulatory Commission, "Part 21 Transfer of Information: "Turbine Control System Impact on Transient Analyses", MFN-04-116, November 12, 2004.

41. U.S. Nuclear Regulatory Commission Regulatory Guide 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants", Revision 2, May 31, 2005.

42. U.S. Nuclear Regulatory Commission Regulatory Guide 1.20 "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" Revision 2, May 1976.

43. NRC Information Notice 2001-13, "Inadequate Standby Liquid Control System Relief Valve Margin".

ATTACHMENT A

0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006

GE Number: 0000-0053-1006 NMP2 A-M-T506-RBM-Calc-2006
Revision Number: 0
DRF Folder: 0000-0053-1006 Rev. 1
DRF Number: 0000-0053-1000
January 2007

Instrument Limits Calculation

CONSTELLATION GENERATION GROUP

Nine Mile Point Nuclear Station Unit 2

Rod Block Monitor (NUMAC ARTS-MELLLA)

Preparer: Documented per Electronic Independent Verification (EIV)
Martin Brittingham /Date

Verifier: Documented per Electronic Independent Verification (EIV)
Robert Copp /Date

Approved: Documented per Electronic Independent Verification (EIV)
Bret Nelson /Date

Contents:

This document is a supplement analysis data sheet to Reference 1. Included in this document in sequential order are:

- the setpoint functions for the system,
- the setpoint function analyses inputs and the source reference of the inputs,
- the devices in the setpoint function instrument loop,
- the component analysis inputs and input sources,
- the calculated results,
- input comments and result recommendations,
- references.

System: Rod Block Monitor (RBM)

The following setpoint functions are included in this document:

- 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)

1. [[

			]]

3. Summary Results:

Calculated Values

Setpoint Function	Analytic Limit (from Section 1) (% ARTS-MELLLA RTP)		Allowable Value (% ARTS-MELLLA RTP) τ		Nominal Trip Setpoint (% ARTS-MELLLA RTP) τ		Meets LER Avoid-ance Criteria	Meets Spurious Trip Avoid-ance Criteria
	Unfiltered	Filtered	Unfiltered	Filtered	Unfiltered	Filtered		
Low Power Setpoint (LPSP)	30%	30%	[[					
Intermediate Power Setpoint (IPSP)	65%	65%						
High Power Setpoint (HPSP)	85%	85%						]]

Setpoint Function	Analytic Limit (from Section 1) (% ARTS-MELLLA RBM Average Flux)		Allowable Value (% ARTS-MELLLA RBM Average Flux)τ		Nominal Trip Setpoint (% ARTS-MELLLA RBM Average Flux)τ		Meets LER Avoid-ance Criteria	Meets Spurious Trip Avoid-ance Criteria
	Unfiltered	Filtered	Unfiltered	Filtered	Unfiltered	Filtered		
Low Power Trip Setpoint (LTSP)	[[							
Intermediate Power Trip Setpoint (ITSP)								
High Power Trip Setpoint (HTSP)								]]

τSee Comment 12.

[[

]]

4. Comments and Recommendations:

[[

]]

ATTACHMENT (6)

AFFIDAVIT BY GENERAL ELECTRIC

General Electric Company

AFFIDAVIT

I, Robert E. Brown, state as follows:

(1) I am General Manager, Regulatory Affairs, General Electric Company ("GE"), and have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in the GE proprietary report, NEDC-33286P, Nine Mile Point Nuclear Station Unit 2 APRM/RBM/Technical Specifications / Maximum Extended Load Line Limit Analysis (ARTS/MELLLA), Class III (GE Proprietary Information), Revision 0, dated March 2007. The GE proprietary information is identified by [[dotted underline inside double square brackets{3}]]. Figures and other large objects are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner, GE relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for "trade secrets" (Exemption 4). The material for which exemption from disclosure is here sought also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

(4) Some examples of categories of information which fit into the definition of proprietary information are:

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

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

c. Information which reveals aspects of past, present, or future General Electric customer-funded development plans and programs, resulting in potential products to General Electric;

d. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. above.

(5) To address 10 CFR 2.390 (b) (4), the information sought to be withheld is being submitted to NRC in confidence. The information is of a sort customarily held in confidence by GE, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GE, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within GE is limited on a "need to know" basis.

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results of analytical models, methods and processes, including computer codes, which GE has developed, obtained NRC approval of, and applied to perform evaluations of transient and accident events in the GE Boiling Water Reactor ("BWR"). The development and approval of these system, component, and thermal hydraulic models and computer codes was achieved at a significant cost to GE, on the order of several million dollars.

The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GE asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GE's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GE's comprehensive BWR safety and technology

base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GE.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

GE's competitive advantage will be lost if its competitors are able to use the results of the GE experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

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

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

Executed on this 22nd day of March 2007.

R. E. Brown

Robert E. Brown

General Manager, Regulatory Affairs