ENCLOSURE 11

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NEDO-33173, Revision 2

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GE Hitachi Nuclear Energy

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Non-Proprietary Information

Licensing Topical Report

Applicability of GE Methods to Expanded Operating Domains

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EXECUTIVE SUMMARY

In the NRC review of GE's generic Maximum Extended Load Line Limit Analysis Plus (MELLLA+) submittal [Reference 1] and the Vermont Yankee Nuclear Power Station (VYNPS) Constant Pressure Power Uprate submittal [Reference 2], the NRC requested additional information (RAI) related to the uncertainties and biases utilized in GE's bundle lattice and core simulation methodologies and the potential effect on safety parameters influenced by such uncertainties and biases. The VYNPS responses to the NRC proposed an additional margin to the safety limit minimum critical power ratio (SLMCPR) and provided bases for the conclusion that other safety parameters did not require additional margin. [References 3 through 7] The MELLLA+ submittal has subsequently been approved [Reference 36].

Revision 0 of this LTR addressing the application of GEH's analytical methods was approved by Reference 37. The NRC Safety Evaluation (SE) approving Revision 0 of this LTR proposed additional margin to the SLMCPR. Revision 2 of this LTR demonstrates that the original uncertainties in References 12 and 13 are adequate for expanded operating domains and that the additional SLMCPR margin proposed in the Revision 0 SE is not necessary. In particular, information provided in Supplement 2, Parts 1 - 3 [References 38, 39, and 40], demonstrates that the original design basis nuclear uncertainties continue to be appropriate. The range of applicability includes any expanded operating range up to 120% of Original Licensed Thermal Power and including the MELLLA+ operating domain expansion. In addition to Supplement 2, other supplements will be provided to demonstrate the adequacy of GE's methods.

Through Supplement 2, the treatment of the uncertainties in the safety limit development is discussed and supported by additional gamma scan information. The effect on six safety parameters is addressed: critical power (safety and operating limit), shutdown margin, fuel rod thermal-mechanical performance, LOCA-related nodal power limits, stability, and licensed pellet exposure.

REVISIONS

Revision	Description of Change
0	Original document submitted February 2006
1	-A Version
2	This revision incorporates the information provided in NEDC-33173P, Supplement 2, Parts 1, 2, and 3 to justify the original design basis nuclear uncertainties. This revision eliminates the additional SLMCPR margin defined in Revision 0.

ACRONYMS AND ABBREVIATIONS

Term	Definition
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without Scram
BOC	Beginning Of Cycle
BT	Boiling Transition
BWR	Boiling Water Reactor
BWROG	BWR Owners Group
CDA	Confirmation Density Algorithm
CPPU	Constant Pressure Power Uprate
CPR	Critical Power Ratio
DIVOM	Delta over Initial MCPR Versus Oscillation Magnitude
DSS-CD	Detect and Suppression Solution – Confirmation Density
ΔCPR	Delta Critical Power Ratio
ECCS	Emergency Core Coolant System
EOC	End Of Cycle
EOP	Emergency Operating Procedure
EPU	Extended Power Uprate
FMCPR	Final Minimum Critical Power Ratio
FWCF	Feedwater Controller Failure Event
FWHOOS	Feedwater Heating Out-of-Service
FSAR	Final Safety Analysis Report
GE	General Electric Company
GESTAR	General Electric Standard Application for Reload Fuel
GEXL	GE Boiling Transition Correlation
GSTRM	GESTR Mechanical
HBB	Hard Bottom Burn
HCOM	Hot Channel Oscillation Magnitude
ICPR	Initial Critical Power Ratio
IV	Instantaneous Void
LHGR	Linear Heat Generation Rate
LOCA	Loss Of Coolant Accident
LTR	Licensing Topical Report
LPRM	Local Power Range Monitor
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA+, M+	Maximum Extended Load Line Limit Analysis Plus

Term	Definition
Methods LTR	Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report
MNCP	A General Monte Carlo N-Particle Transport Code
NRC	Nuclear Regulatory Commission
ODYN	1-D Transient Model
ODYSY	GE Best-Estimate Frequency Domain Stability Code
OLMCPR	Operating Limit MCPR
OLTP	Original Licensed Thermal Power
OPRM	Oscillation Power Range Monitor
Option II	Stability Detect and Suppress LTS for BWR/2
Option III	Stability OPRM-Based Detect and Suppress LTS
PANACEA	Current GE BWR Core Simulator
PCT	Peak Cladding Temperature
PHE	Peak-Hot Excess
PLR	Part Length Rod
PU	Power Uprate
RAI	Request for Additional Information
RPS	Reactor Protection System
SAFDLs	Specified Acceptable Fuel Design Limits
SDM	Shutdown Margin
SE, SER	Safety Evaluation Report
SLMCPR	Safety Limit MCPR
SLO	Single Loop Operation
TGBLA	Current GE BWR lattice physics code
TIP	Traversing In-Core Probes
TRACG	Transient Reactor Analysis Code (GE proprietary version)
TS	Technical Specification
UB	Under Burn
UTL	Upper Tolerance Limit
VH	Void History
1-D	One Dimensional
3-D	Three Dimensional

1.0 INTRODUCTION

1.1 BACKGROUND

Based on previous NRC-approved licensing topical reports and associated NRC Safety Evaluations (SE) for GE's methods, GE has evaluated the accuracy of its methodologies as it has introduced new fuel designs and operating strategies. In the review of the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) submittal [Reference 1] and the Vermont Yankee Nuclear Power Station (VYNPS) Constant Pressure Power Uprate submittal [Reference 2], the NRC requested additional information related to the standard uncertainties and biases utilized in GE's bundle lattice and core simulation methodologies and the potential effect on safety parameters influenced by such uncertainties and biases. The VYNPS RAI responses accepted by the NRC proposed an additional margin to the safety limit minimum critical power ratio (SLMCPR) of 0.02 and provided the bases for the conclusion that other safety parameters did not require additional margin. [References 3 through 7] The MELLLA+ submittal has subsequently been approved [Reference 36].

1.2 PURPOSE

The purpose of the Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report (Methods LTR) is to provide a licensing basis that allows the NRC to issue SEs for expanded operating domains including Constant Pressure or Extended Power Uprate applications and the MELLLA+ LTR. Revision 2 of NEDC-33173P seeks NRC approval for the use of GE's methods for expanded operating domains, bounded by EPU or CPPU power uprates and MELLLA+, without additional SLMCPR margin based upon the information provided in NEDC-33173P, Supplement 2, Parts 1, 2, and 3.

Upon approval of the Methods LTR, each licensee's application for an expanded operating range (CPPU or EPU) may refer to the Methods LTR as a basis for the license change request regarding the applicability of GE's methods to the requested changes. The Methods LTR is a required part of the implementation of the MELLLA+ LTR [Reference 1]. Approval of the Methods LTR would eliminate repetitive RAIs, improve the NRC review schedule, and minimize the resources expended on these reviews by NRC, GE, and the licensees.

1.3 ANALYSIS PROCESS

The approach applied to CPPU, EPU, and MELLLA+ evaluations is discussed in each of the applicable LTRs [References 36, 8, 9, and 10]. An equilibrium cycle core design is the generic approach applied in each of these methods for reactor core and fuel performance related evaluations supporting license change requests. Following the licensing of the proposed changes, the core design for the operating cycle, in which implementation will take place, is evaluated and documented per GESTAR II requirements [Reference 11]. The GESTAR based evaluations effectively set the operating limits for the core. A summary of the applicable limits and the associated methods are given in Table 1-1.

Most licensed core designs typically involve mixed cores (cores containing more than one fuel design or geometry). A licensee may have utilized more than one fuel vendor, in which case there will be legacy fuel bundle designs resident in the current cycle that were not originally designed with GE methods. In these cases, GE complies with the requirements of GESTAR by working with the licensee and vendor to put a proprietary agreement in place. Under this (restrictive and limited) proprietary agreement, sufficient data (e.g., cladding thickness and material type, pellet diameter and density, etc.) is obtained to model the other vendor's fuel design using GE's standard, approved methods. The fuel vendor's original limits are used directly or, as in the case for critical power, an equivalent GE correlation is developed from supplied data. In either case, considerations for uncertainties are taken, and if necessary, additional margin for the legacy fuel uncertainty is incorporated into the applicable limits. This approach is consistent with GE's current approved application methodology.

1.4 OVERVIEW

The subsequent sections of the Methods LTR provide a review of GE methodologies, uncertainties, and biases for acceptability to license applications for expanded operating domains (e.g., CPPU, EPU, and MELLLA+). The uncertainty parameters of interest are identified and their treatment discussed in the context of applications to CPPU, EPU, and MELLLA+ operations. The key safety parameters potentially influenced by these uncertainties are established and the effect of the increase is evaluated. The adequacy of the existing margin for each of these safety parameters is provided.

Section 2 focuses on the evaluation of the effect of uncertainties in the determination of safety parameters for CPPU and EPU applications. Section 3 extends the Section 2 basis to the MELLLA+ operating domain.

Section 4 presents the licensing application framework for the Methods LTR including the applicability range in terms product line, power uprate, and operating domain parameters. The plant specific application process is also included in Section 4. Section 5 summarizes the evaluation of each safety parameter.

Table 1-1 Fuel Design Limits & Associated Methods

Limit	Primary Technology	Description	Evaluation Frequency & Notes
SLMCPR	SLMCPR, PANACEA	The SLMCPR is a MCPR value at which 99.9% of the fuel rods in the core are expected to avoid BT. This value considers the core power distribution and uncertainties.	The limit is evaluated on a plant/cycle specific basis (i.e., each core design).
OLMCPR	ODYN, TRACG, PANACEA	The OLMCPR is additional margin above the SLMCPR to account for the MCPR change due to AOOs. Adherence to the limit assures that in the event of an AOO, 99.9% of the fuel rods are expected to avoid BT.	The limit is evaluated on a plant/cycle specific basis. The FSAR transients that are limiting or potentially limiting with respect to pressure and fuel thermal limits are analyzed for each reload. Transients are confirmed to be within the LHGR basis.
SDM	PANACEA	SDM is maintained regardless of the core design (the value of the limit does not vary with core characteristics like SLMCPR or OLMCPR). The shutdown margin requirement assures that the reactor can be brought and held subcritical with the control system alone. Most BWRs have a Technical Specification (TS) value of 0.38%. The "working definition" of SDM is the quantity of reactivity needed to reach criticality in a xenon free core with the strongest worth control rod fully withdrawn and all other control rods inserted.	Each core is designed to conform to this limit. SDM margin is demonstrated on a plant/cycle specific basis.
LHGR	GSTRM (GESTR-Mechanical)	LHGR Operating Limits represent an envelope of acceptable linear heat generation rates, as a function of exposure, designed to maintain fuel integrity during normal operation, including Anticipated Operational Occurrences. The LHGR limits reflect the application of SAFDLs on the following fuel performance parameters: • Fuel temperature • Cladding stress • Cladding strain • Cladding fatigue usage • Fuel rod internal pressure • Cladding creep	LHGR Operating Limits are developed generically for each fuel product line (e.g., GE14). They are determined from thermal-mechanical considerations and independent of any particular core design.
MAPLHGR	SAFER	MAPLHGR is a an average planar linear heat generation rate limit that is a product of the plant ECCS-LOCA evaluation performed to demonstrate compliance with 10CFR50.46 acceptance criteria.	ECCS-LOCA evaluations are performed as plant specific, cycle independent analyses. These analyses are typically performed for each initial introduction of new fuel product lines. The analysis output is a Licensing Basis PCT and a set of parameters that are confirmed every cycle to ensure applicability of the analysis.
Stability	ODYSY TRACG	There are several accepted stability solutions, each designed to protect the SLMCPR. The solutions include prevention and detect and suppress strategies, as well as combinations of both elements.	The stability methodologies are applied and/or confirmed for every reload (every cycle).
Exposure	GSTRM (GESTR-Mechanical)	The licensed exposure limit is a result of the LHGR evaluation methodology discussed above.	The exposure limit is developed generically for each fuel product line from thermal-mechanical considerations. It is independent of the core design.

2.0 SAFETY PARAMETERS INFLUENCED BY UNCERTAINTIES

2.1 INTRODUCTION

GE has reviewed its methodologies to determine the uncertainties and biases that were confirmed by earlier gamma scan test data or measurements of irradiated fuel isotopics. The purpose of this review was to confirm that the existing uncertainties included in GE's NRC-approved treatment of uncertainties and biases address the NRC staff questions regarding the absence of recent confirmatory test data. Additional data supporting the uncertainties and biases for modern core and fuel designs following the Revision 0 review have now been submitted in NEDC-33173P, Supplement 2, Parts 1, 2, and 3. Supplement 2, Parts 1 and 3 pertain to bundle gamma scans performed at the Cofrentes plant in 2002 and in 2005, and Part 2 pertains to pin-by-pin gamma scans performed at the FitzPatrick plant in 2006.

The associated fuel parameters related to such test data and measurements that are not otherwise measurable directly or indirectly by existing operating plant instrumentation, e.g., local power range monitors (LPRMs) and traversing in-core probes (TIPs), are:

1. Local fuel pin power and exposure (depletion) vs. axial position,
2. Relative local fuel pin power and exposure (local in-bundle peaking),
3. Void reactivity coefficient, and
4. [[ ]]

The fuel parameter uncertainties of interest are thus related to relative local and pin power peaking, void reactivity coefficient, and [[ ]]. Other nodal fuel and bundle parameters, e.g., lattice reactivity, bundle power, and bundle axial power shape, are satisfactorily and adequately confirmed by comparisons to operating plant data or tests, e.g., TIP data and shutdown margin demonstrations.

The safety parameters potentially influenced by local and relative local pin power uncertainties and the [[ ]] uncertainty are:

1. Critical power (controlled by the SLMCPR and OLMCPR),
2. Shutdown margin (controlled with a technical specification limit of 0.38% $\Delta k/k$),
3. Fuel rod thermal-mechanical performance (controlled by limits on linear heat generation rate, LHGR),
4. LOCA-related nodal power limits (controlled via the maximum average planar linear heat generation rate, MAPLHGR),

5. Stability (protected by the SLMCPR, OLMCPR, and stability solutions), and
6. Licensed pellet exposure (e.g., 70 GWd/MT for GE14 fuel)

Each of the uncertainties in question is currently included and addressed in the treatment of uncertainties and biases in GE's NRC-approved methodologies to determine these safety parameters. GE believes it is appropriate to continue to utilize the NRC-approved GE treatment of uncertainties and biases. If consideration of larger uncertainties is deemed appropriate, such uncertainties can be utilized in the existing treatments of propagation and combination of uncertainties. Direct application of biases into best estimate codes in an attempt to address potential uncertainty concerns is not appropriate because such introduction of unqualified biases can lead to potential non-conservatisms in resulting applications. Therefore, the fidelity of GE's codes and methods is best maintained by not artificially adding biases. Conservative limits on safety parameters, developed with consideration for such uncertainties, provide adequate and reasonable assurance of safety.

A discussion of the adequacy of the margin existing in each of these safety parameters is provided below.

2.2 CRITICAL POWER

Fuel bundle critical power is controlled through two analytical limits, the Safety Limit Minimum Critical Power Ratio (SLMCPR) and the Operating Limit Minimum Critical Power Ratio (OLMCPR). The GE treatment of these limits considers uncertainties and biases contained in the methods used to evaluate MCPR.

2.2.1 Safety Limit Critical Power Ratio (SLMCPR)

The SLMCPR is determined as a MCPR value at which 99.9% of the fuel rods in the core are expected to avoid Boiling Transition (BT). The development of the SLMCPR considers uncertainties associated with the determination of total core thermal power from plant instrumentation, as well as the predicted power and flow distribution within the core. The methods and uncertainties used to evaluate the SLMCPR have been approved by the NRC and are documented in NEDC-32601P-A and NEDC-32694P-A [References 12 and 13]. NEDC-32601P-A contains the SLMCPR methodology and uncertainties related to the thermal-hydraulic, pin power peaking and plant instrumentation. NEDC-32694P-A contains uncertainties related to the plant process computer's evaluation of the bundle power distribution.

2.2.1.1 Fuel Parameters That Affect SLMCPR

Table 2-1 and Table 2-2 contain a summary of the uncertainties relevant to the evaluation the SLMCPR.

Table 2-1 Summary of SLMCPR Uncertainties

Uncertainty Parameter	Uncertainty σ (%)	Evaluation Basis
Feedwater Flow System Overall Flow Uncertainty	[[	Section 2.2 of NEDE-32601P-A
Feedwater Temperature Measurement		Section 2.3 of NEDE-32601P-A
Reactor Pressure Measurement		Section 2.4 of NEDE-32601P-A
Core Inlet Temperature		Section 2.5 of NEDE-32601P-A
Total Core Flow Measurement		Section 2.6 of NEDE-32601P-A
TIP Reading and Bundle Power		Table 2-2 Below
TIP Reading Random Uncertainty		Section 2.1 of NEDE-32601P-A
Channel Flow Area Variation		Section 2.7 of NEDE-32601P-A
Friction Factor Multiplier Uncertainty		Section 2.8 of NEDE-32601P-A
Channel Friction Factor Multiplier		Section 2.9 of NEDE-32601P-A
R–factor Uncertainty	]]	Section 3 & Appendix C of NEDE-32601P-A
Critical Power Uncertainty	Different for Each Fuel Type	Evaluated for each fuel product Line Using full-scale critical power test data

The measurement uncertainty items in Table 2-1 (e.g., feedwater temperature) are related to the determination of core thermal power through a heat balance. The total core flow, friction factor, and flow area uncertainties relate to the determination channel flows. The TIP and R-factor uncertainties are relevant to the prediction of bundle and local power. The critical power uncertainty is associated with the GEXL correlation's accuracy for MCPR prediction.

The R-factor is an input to the GEXL critical power correlation that captures the local peaking (pin powers and lattice location) influence on the predicted onset of BT. The R-factor uncertainty is related to the uncertainty associated with nuclear methods in determining the fuel pin power peaking. In addition, the (total) R-factor uncertainty includes terms for manufacturing and channel bow uncertainties.

Uncertainties in bundle power are derived from the parameters shown in Table 2-2, which lists the parameters at the time of the approval of NEDE-32694P-A and their evaluation basis. The parameters are generally based on TIP comparisons from operating plants, [[ ]] from gamma scan measurements.

Table 2-2 Summary of Bundle Power Uncertainties

Uncertainty Parameter	Uncertainty σ (%)	Evaluation Basis
[[		
		]]

The local pin power peaking (axial and in-bundle) and [[ ]] uncertainties are factors that affect SLMCPR. The SLMCPR is not affected by void reactivity coefficient uncertainties.

2.2.1.2 Treatment of Fuel Parameter Uncertainties

GE's NRC-approved process for determining the SLMCPR incorporates the applicable uncertainties in the lattice and core physics parameters, and the method of determining SLMCPR assures that fuel is adequately protected from BT when such uncertainties are incorporated. Uncertainties in local pin power peaking, [[ ]] are explicitly included in the SLMCPR determination and considered separately, then cumulatively below.

Pin Power Peaking

A key method related uncertainty is the local (pin) peaking factor uncertainty. This value is primarily associated with the lattice code TGBLA [Reference 15]. The 1σ uncertainty was evaluated to be [[ ]] in NEDE 32601P-A, based on comparisons with MCNP Monte Carlo evaluations. The overall pin peaking uncertainty, including operational, flux gradient, and manufacturing effects was confirmed by comparison to pin gamma scan measurements performed in an 8x8 lead use assembly. Additional detail regarding the accuracy of the TGBLA code for the evaluation of pin power peaking can be found in the accepted VYNPS RAI responses summarized in Table 2-3.

Table 2-3 Summary of Pin Power Uncertainty Subjects

Related Technology	Subject	RAI
TGBLA, MCNP	Explanation provided to justify acceptability of basing assessment of pin power accuracy on BOL conditions	SRXB-A-37
TGBLA, MCNP	Explanation provided for use of different uncertainties for GE14 and later designs. Refer to response to SRXB-6	SRXB-A-38
TGBLA, MCNP	Explanation provided regarding Cross Sections for High void operation. Refer to generic EPU and MELLLA+ studies.	SRXB-A-46
PANACEA, ISCOR	Justify acceptability of basing assessment of pin power accuracy on code-to-code comparisons. Alternate approach and SLMCPR procedures proposed in response to SRXB-6	SRXB-A-34

The data presented in NEDE-32601P and in the RAI responses above were for the most part based on GE designs. TGBLA-MCNP [Reference 16] comparisons carried out on other vendor's fuel designs show results consistent with those obtained with the GE designs. Table 2-4 is a summary of standard deviation between TGBLA and MCNP pin powers for GE11, GE14, and several Non-GE fuel designs. These results show the overall TGBLA pin power accuracy to be similar for the Non-GE designs and the GE 9x9 and 10x10 designs.

Table 2-4 Summary of TGBLA-MCNP Pin Power Comparisons

Product	Standard Deviation Range 0% Voids	Standard Deviation Range 40% Voids	Standard Deviation Range 70% Voids
[[			
			]]

Additional data has been provided in NEDC-33173P, Supplement 2, Part 2. As discussed in the Section 7.2 and shown in Table 7.2-1 of Supplement 2 Part 2, the largest measured pin power uncertainty of [[ ]], is considerably smaller than the original value of [[ ]] derived from the first four rows of Table 2-11 of this document. This data confirms the adequacy of the assumed safety limit uncertainties by examination of pin-wise gamma scans on modern 10x10 fuel designs.

Four Bundle Power

GE has continued to provide the NRC with BWR fleet information on the consistency of integral TIP comparisons on periodic basis, e.g., in fuel technology updates. These comparisons provide the basis for the [[ ]] in Table 2-2. In 2005, GE provided a large amount of data for uprated plants loaded primarily with 10x10 fuel in methods related RAI responses on the MELLLA+ docket [Reference 17]. The results of plant tracking studies performed with the current methods are summarized in Table 2-5, which yield an overall [[ ]]. Examination of these data confirms the applicability and conservatism of the original [[ ]] uncertainty documented in GE's approved topical reports [Reference 12, NEDC-32601P-A and Reference 13, NEDC-32601P-A] describing the SLMCPR methodology, for uprated power densities as high as 62 KW/liter.

Table 2-5 Summary of High Power Density Plant Tracking Results

Plant	GE BWR Type	Number of Bundles	Original Licensed Thermal Power (OLTP) MWt	Rated Flow (Flow at OLTP) Mlbm/hr	Licensed Power Uprate (PU) % OLTP	Licensed Core Flow Range at PU % Rated Flow	Power Density at Licensed PU kW/l	Cycle	Number of TIP sets	Radial RMS	Nodal RMS
[[											
											]]

1 -- Plant E is a thermal TIP Plant. All the others have Gamma TIPs

Additional detail for the core tracking and four bundle power subjects can be found in the accepted VYNPS RAI responses summarized below in Table 2-6.

Table 2-6 Summary of Four Bundle Power Subjects

Related Technology	Subject	RAI
PANACEA, ISCOR	Information provided for maximum bundle power and power density before and after EPU	SRXB-A-64
PANACEA, ISCOR	Explanation provided for increase in nodal uncertainties with elevation	SRXB-A-25
PANACEA, ISCOR	Information and discussion supplied regarding criteria for axial and nodal uncertainties	SRXB-A-27
PANACEA, ISCOR	Information and discussion of SLMCPR evaluation and monitoring accounting for axial and nodal uncertainties	SRXB-A-28
PANACEA, ISCOR	Application of nodal uncertainties and increases with exposure. Refer to SRXB-6 and SRXB-31.	SRXB-A-32
PANACEA, ISCOR	Core Follow Data Supplied	SRXB-A-35
PANACEA, ISCOR	Explanation of effect on pin power due to neighboring bundles provided with explicit results for 10x10 lattices	SRXB-A-39
PANACEA, ISCOR	Discussion of bypass voiding on instrumentation provided	SRXB-A-44
PANACEA, ISCOR	Refer to SRXB-A-19 for Representative Core definition	SRXB-A-9
PANACEA, ISCOR	Reasons for differences between PCTIP and axial power distributions provided	SRXB-A-36
PANACEA, ISCOR, ODYN	Explanation of inclusion of axial and nodal uncertainties in transient and accident evaluations provided	SRXB-A-29

Bundle Power

[[ ]] is a component of the total bundle power uncertainty. The total bundle power uncertainty for application within GE's approved SLMCPR determination process consists of the component uncertainties in Table 2-2, which is from Table 4.2, page 4-2 in NEDC-32694P-A. The basis of the SLMCPR uncertainties is embodied in the 3D Simulator PANACEA and the SLMCPR methods. [[

Plant and Cycle	[[ ]] RMS Difference (%)	Number of 4 Bundle Sets	New Fuel Geometry	Core Power Level (MWt)	Avg. Power Density (kW/l)	New Fuel Batch Fraction
Hatch 1 EOC1	[[	24	7x7	2436 (100%)	51.2	Initial core
Hatch 1 EOC3		26	8x8 (C2) 8x8R (C3)	2436 (100%)	51.2	92 (C2) 168 (C3)
Weighted Average						
Cofrentes EOC13		8	9x9 10x10 SVEA	2891 (100%)	52.4	64 (GE12) 128 (SVEA)
Weighted Average						
Cofrentes EOC15		8	10x10 GE14 10x10 OPTIMA2	3238 (100%)	58.6	72 (GE14) 136 (OPTIMA2)
Weighted Average	]]					

This demonstrates that the steady-state nuclear methods adequately predict the power distribution for these situations such that the existing [[ ]] uncertainty of [[ ]] used in the SLMCPR process does not require an adjustment.

Additional detail regarding the bundle power subject can be found in the accepted VYNPS RAI responses shown in Table 2-7 below.

Table 2-7 Bundle Power Subject

Related Technology	Subject	RAI
PANACEA, ISCOR	Explanation supplied for the uncertainties applied to LHGR. Refer to SRXB-A-68	SRXB-A-24
PANACEA, ISCOR	Explain provided for increase in nodal uncertainties with elevation	SRXB-A-25
PANACEA, ISCOR	Information and discussion supplied regarding criteria for axial and nodal uncertainties	SRXB-A-27
PANACEA, ISCOR	Information and discussion of SLMCPR evaluation and monitoring supplied for axial and nodal uncertainties in safety limit analyses	SRXB-A-28

The effects of [[ ]] in Table 2-2 on the bundle power uncertainty for SLMCPR determination [[ ]]

Critical Power Correlation

In addition to power distribution uncertainties, thermal-hydraulic parameters are also included in the SLMCPR evaluation. The GEXL correlation uncertainty is used to establish the probability of boiling transition. The application range of the GEXL correlation is illustrated in Figure 2-1.

The critical power correlation is developed from full-scale critical power test data for each fuel product line. The critical power data are obtained for bundle mass fluxes ranging from [[ ]], inlet subcooling [[ ]] and pressures from [[ ]]. These data cover flow ranges from less than natural circulation to well beyond rated flow and include the flow ranges for EPU and MELLLA+ applications. These data cover bundle power levels up to the actual critical power for each set of conditions, which is in the range of [[ ]] for 10x10 fuel. These fluid parameter ranges also cover the expected ranges for LOCA and transient events. The development of GEXL correlation coefficients and constants for a fuel assembly follows the NRC approved process described in GESTAR II [Reference 11]. Figure 2-1 shows the GE14 application range together with the expected range for typical operational transients. The box representing the correlation application range encloses the expected ranges for transients. For LOCA application, the GEXL correlation is used for the calculation of the early boiling transition during the flow coast down immediately following the break. This typically occurs when the flow has dropped to 30-50% of the initial value. This is well within the application range for the GEXL correlation. The range of bundle powers and hydraulic conditions for the GEXL correlation covers those expected in MELLLA+ and EPU operation.

Figure 2-1 GEXL14 Application Range

[[

]]

Void Fraction

Steam void fraction uncertainty does not appear explicitly in Table 2-1, but is incorporated into the SLMCPR evaluation through the other flow related uncertainties. The void correlation is based on void fraction data up to approximately [[ ]], which covers the void fraction range expected for normal steady state operation and the abnormal operational occurrences that set the operating limit minimum critical power ratio (OLMCPR). Attachment A, "BWR Fuel Void Fraction," of Appendix A to NEDC-32601P-A [Reference 12], contains an extensive discussion of the void correlation, fuel design evolution, and sensitivities (e.g., nuclear performance).

As discussed in Attachment A to NEDC-32601P-A, the part length rod (PLR) is the major new feature in current fuel products. The impact of PLRs has been experimentally investigated for a 4X4 bundle for a pressure of 145 psia and more recently for an 8X8 bundle at rated BWR pressure of 1044 psia. A small increase, approximately [[ ]], was observed in void fraction downstream of the PLRs compared to the case with no PLR for the low-pressure 4X4 data. More recent representative 8X8 data taken at normal operating pressure shows a small increase, on the order of [[

]].

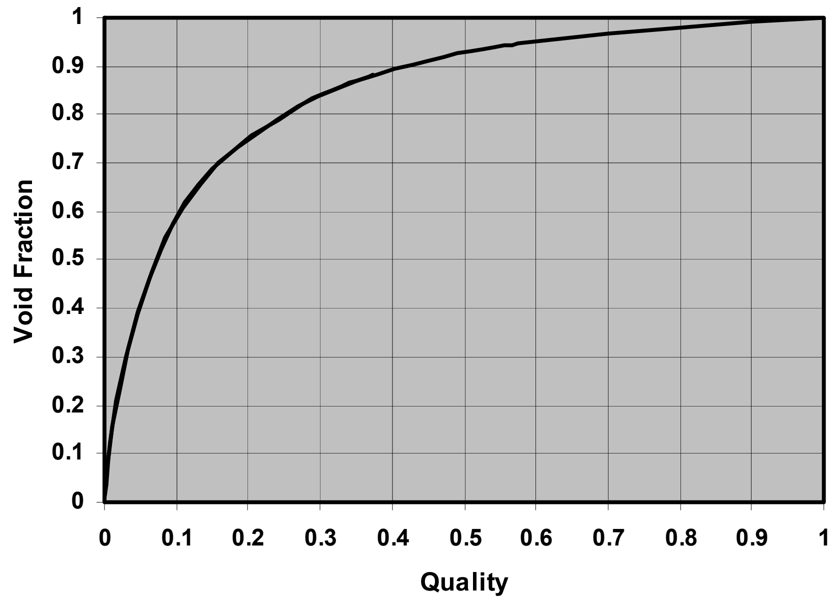
A void fraction of [[ ]] is relatively high and typical of the conditions where boiling transition will occur in a BWR fuel bundle. Also, since the OLMCPR is determined such that boiling transition will not occur, it is highly unlikely that a void fraction of [[ ]] will be exceeded (e.g., perhaps momentarily during a transient) by any significant amount. Some aspects of void fraction and bundle power warrant a brief discussion. For illustrative purposes, consider a one-dimensional, steady state energy balance for a BWR fuel channel. It can be shown that the flow quality is

$$X(z) = \frac{h_{in} - h_f}{h_{fg}} + \frac{1}{\dot{m} h_{fg}} \int_0^z \dot{q}'(\xi) d\xi ,$$

where the definition of flow quality is given by $X = \dfrac{\dot{m}_g}{\dot{m}_f + \dot{m}_g}$

The flow quality is a function of pressure (fluid properties), inlet flow rate and subcooling, and the heat addition rate. For the case of “z” equal to the exit elevation, the integral term essentially represents the channel power. The steady state exit quality is directly proportional to the integrated channel power.

Figure 2-2 Typical Void-Quality Relation at High Power/Flow Ratio



It should be recognized that a BWR fuel bundle is designed and operated such that boiling transition will not occur during steady-state or abnormal operational occurrences, and, therefore, high void fractions, i.e., higher than [[ ]], will not occur. Figure 2-2 illustrates this point, noting that less than half of the quality range (X < 0.5) covers up to 90% void fraction. A significant power increase (or a factor of 2 change in quality) is required to drive the void fraction from 90 to 100%. It would require a bundle power of approximately [[ ]] for a bundle at rated flow to reach a void fraction of [[ ]], while in reality a high power fuel bundle operates at approximately [[ ]].

The void quality correlation is based on sound physical principles, particularly for high void fractions, and extrapolates the measured data to a void fraction of 1.0. Using the Zuber-Findlay expression [Reference 14] for two-phase flow, the void fraction α can be expressed as

$$\alpha = \frac{j_g}{C_0 j + \overline{V}_{gj}}$$

Where:

C_0 = distribution parameter

$\overline{V}_{gj}$ = drift velocity

j_g = volumetric flux of steam vapor

j = volumetric flux of the mixture

The drift velocity is the difference in velocity between the vapor and liquid phase. Generally the vapor phase velocity is greater because of buoyant forces. At high quality, the annular flow regime predominates. In the annular flow regime the liquid phase surrounds the fuel rods and channel. As the void fraction increases, the drift velocity decreases, as the buoyant forces become less important. In the GE void correlation, the drift velocity is characterized as

$$\overline{V_{gj}} \propto (1-\alpha)$$

This characterization is applied over the entire annular flow region, or for void fractions greater than about 0.4. For high void fractions and small values of $\overline{V_{gj}}$, the void fraction is dominated by the ratio of vapor mass flux to total mass flux, determined by a simple mass and energy balance for each node. The outstanding agreement over the entire range shown in the response to SRXB-A-69 validates this simple model for the drift flux. An extrapolation based on this model to void fractions all the way to pure steam flow is justified. In summary, the GE void correlation is based on test data and covers a broad range of conditions. The correlation supports the full range of conditions expected during BWR operation, including CPPU, EPU and MELLLA+ conditions. The correlation uncertainty is appropriately accounted for in the SLMCPR. It is not necessary to incorporate additional margin for void fraction uncertainty.

Additional detail regarding the thermal-hydraulic subjects can be found in the accepted VYNPS RAI responses shown in Table 2-8 below.

Table 2-8 Thermal-Hydraulic Subjects

Related Technology	Subject	RAI
Void and pressure drop correlations	Pressure Drop data base information provided, reference made to generic MELLLA+ report	SRXB-A-52
Void and pressure drop correlations	Void fraction measurement data made through Safety Limit Document reference	SRXB-A-53
Void and pressure drop correlations	Are void fraction uncertainties included in water density? Explanation provided	SRXB-A-54
Void and pressure drop correlations	Explanation and information provided regarding Void fraction uncertainties	SRXB-A-69
Void and pressure drop correlations	Explanation provided regarding acceptable to exceed correlations range. Refer to SRXB-A-55	SRXB-A-70

2.2.1.3 Adequacy of Existing Treatment

The standard GEH methodologies utilized to establish the SLMCPR conservatively address uncertainty issues and provide reasonable assurance of safety for CPPU and EPU applications including MELLLA+.

2.2.2 Operating Limit Critical Power Ratio (OLMCPR)

The analysis of anticipated operational occurrences (AOOs) examines the change in critical power ratio relative to the starting initial conditions and determines the most limiting event.

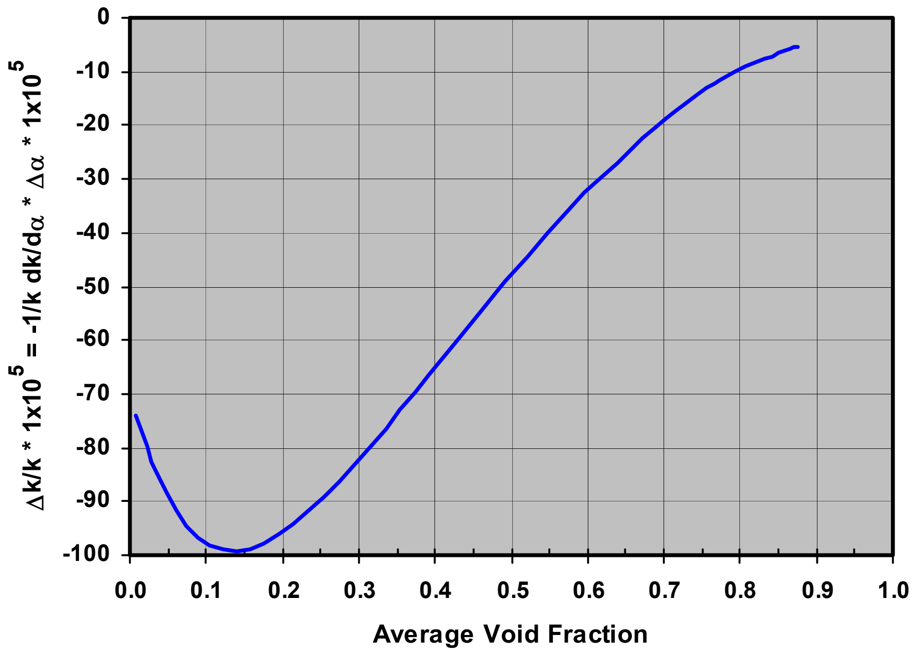
2.2.2.1 Fuel Parameters That Affect OLMCPR

The fuel parameters identified previously, i.e., the local pin power peaking, void reactivity coefficient, and three dimensional power distribution are factors in the evaluation of limiting AOOs. The typical AOO response (e.g., pressurization event) is mainly affected by the reactivity void coefficient and the axial power distribution at the beginning of the event. Power distributions peaked to the top of the core will reduce the scram reactivity early in the transient and most of the time will increase the transient MCPR change. The transient response also depends on the void and Doppler coefficients of reactivity. An increase in fuel temperature increases the resonance absorption in the fuel isotopes and reduces the reactivity during a pressurization transient. The overall Doppler effect is, however, quite small in BWRs and uncertainties in Doppler reactivity have a negligible effect on transient behavior. The transient behavior is more sensitive to the void reactivity coefficient. A larger void coefficient can increase the initial flux increase during a pressurization transient such as a turbine trip, but will also act to aid in shutdown once the increase in power results in revoiding the core.

Figure 2-2 shows a typical plot of the void-quality relationship for a flow typical of a high power/flow ratio fuel bundle for the entire range from zero to one. Recognizing the relationship between quality and energy input (channel power), the figure has two interesting points relevant to discussions of the void coefficient and void feedback. First, Figure 2-2 shows that the lower end of the quality range has a relatively steep slope. Small power changes in this lower quality range correspond to a relatively large void fraction change. This behavior has implications relative to the impact of the void coefficient. In general, the void coefficient becomes more negative with increasing (average) void fraction. However, the net power effect considering the void-quality behavior is that in general, core power response is more strongly influenced by regions of the core with low void fraction. In other words, the quantity $\Delta\alpha \simeq \left(\partial\alpha/\partial X\right)\Delta X$ tends to be larger at low void fraction, so that the effective feedback $\Delta k/k \simeq 1/k\left(\partial k/\partial\alpha\right)\Delta\alpha$ tends to be larger. Second, the higher quality (or power) range is relatively flat with respect to void fraction. Changes in power at high quality result in relatively small void fraction changes. In terms of core power response, effective void feedback tends to be milder at higher void fractions.

Void coefficient uncertainties and biases have a lower effective worth (in terms of reactivity feedback) at high void conditions than at lower void conditions. This relative difference is depicted in Figure 2-3, which was derived from the void and quality values shown in Figure 2-2 combined with a simple expression for the derivative $\partial\alpha/\partial X = f(X)$ based on a homogeneous flow model. Figure 2-3 shows the reactivity effect of a small quality perturbation (ΔX = 0.001) using a representative void coefficient over a range of void fraction values.

Figure 2-3 Reactivity Change for a Small Quality Perturbation ($\Delta X = 0.001$) as a Function of Void Fraction



Accommodation for uncertainties in local pin power peaking and [[ ]] (and bundle power), i.e., consideration of bundle and nodal powers higher (or lower) than expectations, is directly incorporated in the licensing methodology. Thus, there is no effect on ΔCPR due to the NRC staff questions regarding the local pin power peaking and [[ ]] uncertainties.

2.2.2.2 Treatment of Fuel Parameter Uncertainties

As stated above, the core axial power shape can influence the transient response. Uncertainties in the axial power shape are not directly included in the transient response uncertainty. Rather the input conditions for the transient are developed in a way that ensures that the axial shape is conservative. [[

]] This assures that the analysis is both realistic but conservative.

Both the ODYN and TRACG transient methodologies [References 18, 19, and 20] have established application ranges for void coefficient uncertainty. The approval of and GE confidence in the basis for these methodologies are based upon comparison of calculations for a wide variety of plant transients in which the nominal void coefficient is used. The acceptable performance of these codes relative to the data justifies that no large errors in void coefficient exist. The response to VYNPS questions related to void coefficients are SRXB-A-51 and SRXB-A-68.

The TGBLA06 methodology is applied in core design, transient analysis, stability analysis, and monitoring. TGBLA06 and MCNP have been utilized to generate void coefficient data and for 5 representative 10x10 lattices for the full range of instantaneous void (called IV) conditions. Complete results are contained in the response to VYNPS RAI SRXB-A-68. The calculations are based on a 40% void history (called VH) depletion followed by branch calculations at 0, 40, and 70% IV. The results are extrapolated above 70% IV. The average bias over the full exposure range is approximately [[ ]] at 70% IV. The average bias at 40% IV is approximately [[ ]]. Over this IV range, the magnitude of the bias is considered [[ ]]. The average uncertainty at 70% IV is [[ ]]. This uncertainty is representative of the 40% void fraction range (also [[ ]]). The value assumed in the Revised Supplementary Information Regarding Amendment 11 to GESTAR [Reference 21] is [[ ]]

Additional analyses have been performed in which MCNP calculations have been performed from 40% void history, 70% void history, and 90% void history. MCNP branch cases have been performed to instantaneous voids of 70%, 80% and 90%. These analyses were performed for lattice exposures of [[

]]

In summary, for applications that utilize TGBLA06 based modeling (PANAC11, ODYN, TRACG, and ODYSY) the evaluation discussed above for [[ ]] void fraction (Table SRXB-68-1 of VYNPS RAI SRXB-A-68) is applicable to the consideration of both the TGBLA06 cross section extrapolation process and the TGBLA06 void history assumption. An assumption of [[ ]] bias and a 2σ uncertainty of [[ ]] is justified.

The key transients analyzed in the response to VYNPS RAI SRXB-A-68 were pressurization events in which the void fraction decreases due to increasing core pressure and then later increases due to higher heat flux. These conclusions can also be applied to cold water events. The transient response to cold-water events initiated by lower feedwater temperature is generally less severe than the pressurization events initiated from full power. For example,

- The feedwater controller event (FWCF) triggers a rise in reactor power, which in turn initiates a turbine trip. Hence sensitivities developed for other pressurization events apply to the FWCF transient.
- The loss of feedwater heating (LOFW) event initiates a slow rise in power to a level just below the APRM scram set point. This event is analyzed by the PANACEA steady-state simulator. The initial and final core void fractions for this event are nearly the same, because the effect of the reduced inlet temperature is offset by the increased reactor power. The sensitivity of this event to variations in void coefficient is negligibly small as discussed in Section 8.4.1.5 of NEDE-32906P-A. [Reference 20]
- Transients initiated from operation with feedwater heating out of service (FWHOOS) are less severe, because they start from a lower power and result in a lower pressurization rate. Sensitivities developed for other transients initiated from full power can be applied to one initiated from FWHOOS conditions.

The ODYN model uncertainty is based on comparisons to the benchmark Peach Bottom turbine trip tests. [[

]]

Because inputs to the OLMCPR analysis are conservative, and the pressurization transients that typically establish the limiting ΔCPRs are conservatively analyzed by TRACG or ODYN, the conservatisms in the process of determining OLMCPRs address NRC questions related to gamma scans and fuel isotopics as they relate to OLMCPR.

Additional detail regarding the OLMCPR subjects can be found in the accepted VYNPS RAI responses shown in Table 2-9 below.

Table 2-9 OLMCPR Subjects

Related Technology	Subject	RAI
ODYN	NRC staff approved evaluation model identified for ATWS and discussion provided on EOP's	SRXB-A-22
ODYN	Explanation of uncertainties in power during transients	SRXB-A-58
ODYN	Over pressure protection analysis code was identified	SRXB-A-7
TGBLA, MCNP	Explanation of Cross Sections for High void operation provided. Refer to generic EPU and MELLLA+ studies	SRXB-A-46
TGBLA, MCNP	Plots of isotopic concentrations provided	SRXB-A-47
TGBLA, MCNP	Information on the isotopic influence on void coefficient	SRXB-A-48
TGBLA, MCNP	Discussion provided on Void reactivity coefficients for transients and accidents, including ATWS and SBO.	SRXB-A-51
TGBLA, MCNP	Explanation provided on the effect of EPU on spent fuel storage Refer to SRXB-A-11	SRXB-A-61
TGBLA, MCNP	Describe transients used to determine MCPR	SRXB-A-63
TGBLA, MCNP	CASMO/TGBLA code comparisons	SRXB-A-66
TGBLA, MCNP	Void reactivity coefficients -- provided more information than response to SRXB-A-51	SRXB-A-68
TGBLA, MCNP	Clarification and detail on response to SRXB-A-57	SRXB-A-71

2.2.2.3 Adequacy of Existing Treatment and Alternate Approach

The standard GE methodologies utilized to establish the OLMCPR conservatively address uncertainty issues and provide reasonable assurance of safety for CPPU and EPU applications including MELLLA+.

2.3 SHUTDOWN MARGIN (SDM)

The Technical Specification for Shutdown Margin requires that the core be designed so that it can be shut down at any time in life while in the most reactive condition (usually cold, 20°C) with the most reactive control blade removed. This condition is verified by experiment at cycle startup and is often repeated later in the operating cycle.

2.3.1 Fuel Parameters That Affect SDM

The analysis of SDM considers whether core reactivity can be safely controlled. The fuel parameters identified previously, i.e., the local pin power peaking and [[ ]], are secondary factors in the evaluation of SDM since uncertainties in those parameters may ultimately influence prediction of fuel depletion and, thus, fuel reactivity. Void reactivity coefficient is not a contributor since essentially zero voiding is present at hot or cold shutdown conditions. The GE bundle lattice and core simulation methodologies are best estimate predictions so that validation of operating benchmark data, core follow, and core licensing can proceed using consistent methodology. Comparisons to actual plant cold critical states are an important part of this validation because errors in bundle or nodal power (or exposure) would tend to degrade the ability of the core simulator to establish a stable bias (in eigenvalue), which is a measure of the ability of the model to reliably predict core hot and cold critical conditions. Conversely, the establishment of a stable eigenvalue bias for hot and cold critical conditions is indicative of adequate fidelity of the model to predict bundle and nodal power and exposure.

2.3.2 Treatment of Fuel Parameter Uncertainties

A shutdown margin demonstration experiment is performed at the beginning of each operating cycle. This demonstration is performed in the cold, or most reactive criticality condition. The demonstration configuration attempts to simulate the most reactive rod out condition. In order to obtain a critical condition, other rods are also withdrawn. The 3D simulator [Reference 15] is used to calculate the demonstration condition. Let k_{demo} be the calculated critical eigenvalue for the demonstration condition. The cold shutdown technical specification requires that

$$k_{sro} \leq k_{demo}(1. - 0.0038)$$

where k_{sro} is the calculated criticality for the strongest rod withdrawn condition and 0.0038 is the required shutdown margin. This required shutdown margin is meant to account for possible differences in critical eigenvalue between the demonstration condition and the technical specification condition. The value was originally determined to account three uncertainties on the critical configuration: the impact of manufacturing tolerances, variations in predictive capability within the same core and variations in exposure on the critical configuration. The 0.0038 magnitude represents the 2-sigma value of the RMS combinations of the aforementioned uncertainties. The current validity of the 0.0038 requirement can be determined by comparing critical eigenvalue demonstrations, all of which are carried out on the same core. Figure 2-4 below is a reproduction of one shown in the response to [Reference 17] and is a summary of the cold critical analyses carried out on the five reference plants.

Figure 2-4 Reference Plants Cold Critical Eigenvalues

[[

]]

Of the 39 critical experiments shown in Figure 2-4, there were five cores, summarized in Table 2-10, for which multiple cold critical experiments were performed on the same core. The standard deviation of the critical eigenvalues for the cores in Table 2-10 relative to the average obtained for the same core is [[ ]]. This standard deviation can be compared to the Technical Specification allowance of 0.38% Δk/k., indicating that for application to high power density cores, the data supports the continued use of the current Technical Specification limit.

Table 2-10 Summary of Same Core Critical Experiments

Plant	Cycle	Cycle Exposure (GWD/ST)	Number of Critical Experiments	Standard Deviation of k_{demo}
[[				
				]]

While the Technical Specification for SDM is 0.38% Δk/k reactivity (for an in-sequence check only), normal GE design procedure is to provide design cold shutdown margins of 1% or more depending on customer request and GE procedure. The standard design SDM is 1.0% Δk/k to provide additional flexibility in cycle length and operations, although each plant is free to require more design margin if deemed appropriate. The uncertainty in cold critical predictive capability is considered and included in this choice of SDM requirement. The ability to meet the projected margin has also been evaluated for the data presented in Figure 2-5. Before cycle startup, a cold critical eigenvalue is projected for the cycle. This critical eigenvalue is based on previous cycle experience and is the result of a well-defined design procedure. The difference between the projected and measured eigenvalue is plotted in Figure 2-5 as a function of cycle exposure. The standard deviation of the differences is [[ ]]. The behavior shown in Figure 2-5 shows that the nuclear methods together with procedures for projecting critical eigenvalues for the next cycle accurately predict design margins.

Figure 2-5 Difference Between Measured and Predicted Cold Critical Eigenvalues

[[

]]

A failure to meet the Technical Specification SDM requirement is severe in that a redesign of the core loading and/or fuel design would be required to restart the plant. A design margin of 1% SDM has been used by GE for many years to ensure that ≥ 0.38% $\Delta k/k$ is always satisfied. The additional margin between the Technical Specification SDM and 1% allows for the following factors to impact the prediction capability of the simulator:

1. Operation of the plant different than that projected
2. Fuel manufacturing tolerances
3. Control rod worth reduction due to depletion of control rod absorber material
4. Methodology approximations
5. Inexact tracking of actual plant parameters
6. Other unidentified factors

Of these factors, the most significant is allowance for operation different from that projected. Each core design must maintain sufficient operational flexibility to protect the core and fuel while meeting economic objectives. Factors affecting the GE application methodology are quantified through the uncertainty in cold critical eigenvalue and deviation from expectations.

The accepted response to VYNPS RAI SRXB-A-67 contains additional detail and information on shutdown margin qualification.

2.3.3 Adequacy of Existing Treatment and Alternate Approach

The current design process and Technical Specification SDM, in combination with the existing plant verification of SDM and trending of hot eigenvalues, provide reasonable assurance of adequate SDM. The GE procedure of designing for 1% SDM provides substantial additional assurance of adequate SDM.

2.4 FUEL ROD THERMAL-MECHANICAL PERFORMANCE

For each GE/GNF fuel design, thermal-mechanical based linear heat generation rate limits (LHGR Operating Limits) are specified for each fuel rod type (for both UO_2 and gadolinia-bearing rods) such that, if each rod type is operated within its LHGR limit, all thermal-mechanical design and licensing criteria, including those which address response to anticipated operational occurrences (AOOs), are explicitly satisfied and fuel rod integrity is maintained.

2.4.1 Fuel Parameters That Affect Thermal-Mechanical Limits

The fuel parameters identified previously, i.e., the local pin power peaking, void reactivity coefficient, [[ ]], are factors, to differing extents, in the development of LHGR Operating Limits. These fuel parameters ultimately determine the local power, which is explicitly addressed by the LHGR Operating Limit.

2.4.2 Treatment of Fuel Parameter Uncertainties

A number of fuel rod thermal-mechanical analyses are performed to evaluate fuel performance relative to Specified Acceptable Fuel Design Limits (SAFDLs). The SAFDLs include considerations such as the fuel rod internal pressure developed during normal steady-state operation, and the maximum fuel temperature and cladding strain experienced during Anticipated Operational Occurrences (AOOs). An output from these analyses is the specification of an LHGR Operating Limit, in conjunction with a [[ ]] exposure limit. LHGR Operating Limits are determined and specified in the form of allowable [[ ]] LHGR as a function of [[ ]] exposure. These fuel rod thermal-mechanical performance based operating limits are specified for each fuel rod type (UO_2 or $(U,Gd)O_2$ for various gadolinia concentrations) so that if each fuel rod type is operated within its respective exposure-dependent LHGR limit, all thermal-mechanical design and licensing criteria (SAFDLs), including those which address response to AOOs, are explicitly satisfied.

The exposure-dependent LHGR Operating Limits are determined through the performance of a number of fuel rod thermal-mechanical analyses. An important assumption with these analyses is [[

]]. This assumption represents a significant conservatism; [[

]]

With this conservative [[ ]] assumption, the thermal-mechanical analyses are performed either on a worst tolerance basis or statistically. For those analyses performed statistically, such as the fuel rod internal pressure analysis, the uncertainty in each fuel rod fabrication parameter is determined and specifically addressed. The fuel rod thermal-mechanical model prediction uncertainty is also determined and addressed. [[

]]

For the GE14 fuel rod thermal-mechanical design and licensing analyses, the values of the preceding component uncertainties are: [[

]]

The LHGR Operating Limit is derived for an individual fuel design using the following basic procedure.

- [[

-

]]

[[

]]

Table 2-11 Summary of Uncertainty Components for LHGR Evaluations

Component	NEDE-32601 (1)	Revision 0 (1)	Revision 0 (2)	Revision 2
[[				
				]]

Notes:

(1) Values from NEDC-33173P Revision 0 Safety Evaluation Table 3-11 [Reference 37]

(2) Separate from the Methods LTR Supplement 2 uncertainty qualification, it was noticed that the update uncertainty should be [[ ]] as stipulated in RAI II.5 of NEDC-32694P-A [Reference 13].

[[

]].

Additional detail regarding the LHGR Operating Limit subjects can be found in the accepted VYNPS RAI responses shown in Table 2-12 below. The relationship between the methods uncertainties and LHGR criteria is summarized in the response to SRXB-A-65.

Table 2-12 Fuel Performance Related Subjects

Related Technology	Subject	RAI
GSTRM (GESTR-Mechanical)	Uncertainties in LHGR limit evaluations	SRXB-A-65
PANACEA, ISCOR	Uncertainties applied to LHGR	SRXB-A-24
PANACEA, ISCOR	Application of nodal uncertainties to nodal exposure to MAPLHGR and LHGR values	SRXB-A-30
PANACEA, ISCOR	Does LHGR limit in 3D simulator include decrease with exposure	SRXB-A-31
PANACEA, ISCOR	Application of nodal uncertainties and increases with exposure	SRXB-A-32
PANACEA, ISCOR	Describe how core monitoring system calculate pin wise power parameters	SRXB-A-33
PANACEA, ISCOR	Effect on pin power due to neighboring bundles	SRXB-A-39

2.4.3 Adequacy of Existing Treatment and Alternate Approach

The standard GE methodology for determining LHGR limits includes conservative consideration for, and provides reasonable assurance of adequate margin to address, the power and void reactivity uncertainties in question.

2.5 LOCA RELATED NODAL POWER LIMITS

The purpose of the maximum average planar linear heat generation rate (MAPLHGR) limits is to assure adequate protection of the fuel during a postulated loss-of-coolant accident (LOCA) with the defined operation of the emergency core cooling system (ECCS).

2.5.1 Fuel Parameters That Affect LOCA Related Nodal Power Limits

The fuel parameters identified previously, i.e., the local pin power peaking and [[ ]], are factors, to differing extents, in the development of LHGR limits. The fuel parameters ultimately determine the local power, which is the subject of the MAPLHGR, a local limit. The void reactivity coefficient is not a factor in the ECCS-LOCA analysis.

2.5.2 Treatment of Fuel Parameter Uncertainties

The ECCS-LOCA analysis follows the NRC-approved SAFER/GESTR application methodology documented in Volume III of NEDE-23785-1-PA [Reference 22]. The analytical models used to perform ECCS-LOCA analyses are documented in Volume II of NEDE-23785-1-PA [Reference 23] together with NEDE-30996P-A [Reference 24] and NEDC-32950P [Reference 25].

When SAFER/GESTR methodology is applied, the hot bundle is initialized with a [[ ]] In addition, a [[ ]] In order to ensure that the SAFER analysis is bounding for all exposures, the hot rod of the hot bundle is placed at the exposure corresponding to the [[ ]] In addition to these analytical conservatisms, margin to the MAPLGHR limits is maintained during plant operations.

Total bundle power is also important to the severity of the ECCS-LOCA analysis. [[

]] Furthermore, the ECCS-LOCA basis target MCPR is set lower than the OLMCPR so that the OLMCPR is not set by the ECCS-LOCA analysis (i.e., it is set by the AOO analysis).

Pin power peaking for the hot rod is set to a [[

]] to further ensure that the ECCS-LOCA results are bounding.

Lastly, the axial power profile [[

]]

The above considerations indicate that significant conservatisms related to initial local pin and bundle powers exist in the GE SAFER/GESTR ECCS-LOCA methodology.

In addition to the above conservatisms, the Licensing Basis peak cladding temperature (PCT) determined by the methodology described above must be greater than the Upper Bound PCT. The Licensing Basis PCT includes application of Appendix K modeling assumptions and plant variables uncertainties. The Upper Bound PCT in the SAFER/GESTR methodology adjusts the nominal PCT to account for modeling and plant variable uncertainties (at 95% probability). The 95% probability PCT includes an uncertainty of [[ ]] on the LHGR.

Additional detail regarding the LOCA/ECCS analyses can be found in the accepted VYNPS RAI response shown in Table 2-13 below.

Table 2-13 LOCA/ECCS Related Subjects

Related Technology	Subject	RAI
SAFER	Information supplied regarding PCT difference in VYNPS LBLOCA analysis	SRXB-A-10

The SAFER/GESTR methodology assumes a bounding post-LOCA core power decay and, thus, core kinetics are not modeled. The average and hot bundle void profile is determined by SAFER at the limiting initial conditions described above as well as at the post-LOCA conditions. Uncertainties in predictions of void reactivity have no impact in the SAFER/GESTR methodology. The overall SAFER/GESTR methodology is designed to maximize the PCT.

2.5.3 Adequacy of Existing Treatment and Alternate Approach

The conservatism of the present ECCS-LOCA methodology used to determine MAPLGHR limits adequately considers the effects of the uncertainties in local and bundle power and provides adequate and reasonable assurance that those limits provide adequate margin to protect the fuel.

2.6 STABILITY

Thermal-hydraulic stability analyses are performed to assure that the SLMCPR is protected in the event of a thermal-hydraulic instability event. Specific analyses are associated with each of the long-term stability solutions that have been licensed and implemented in the U.S. These long-term solutions include Option I-D, Option II, Option III, and Enhanced Option I-A.

10CFR50, Appendix A, General Design Criterion (GDC) 10 requires that the reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

10CFR50, Appendix A, GDC 12 requires that the reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

2.6.1 Fuel Parameters That Affect Stability

The fuel parameters identified previously, i.e., the local pin power peaking, void reactivity coefficient, and [[ ]], affect stability performance to differing extents.

2.6.2 Treatment of Fuel Parameter Uncertainties

The treatment of the fuel parameter uncertainties for each of the long-term stability solutions listed above is provided in the following discussion.

2.6.2.1 Option I-D

Option I-D has (1) "prevention" elements and (2) a "detect & suppress" element. The prevention portion of the solution includes separate administratively controlled exclusion and buffer regions, which are evaluated for every reload. The detect-and-suppress portion of the solution is a flow-biased APRM flux scram trip that prevents oscillations of significant magnitude. This scram ensures the Fuel Cladding Integrity SLMCPR is protected for the dominant core wide mode of coupled thermal-hydraulic/neutronic reactor instability.

Stability analyses for both the EPU and fuel cycle specific conditions are performed to define the exclusion and buffer regions as well as to confirm that the scram setpoints meet the design basis. With respect to power distribution uncertainties of the nuclear simulator data, the results pertaining to the exclusion region may be slightly affected, but this is not considered to have any safety significance for reasons described below. The power distribution uncertainties of the nuclear simulator data are considered in the determination of the limiting bundle conditions and therefore have insignificant impact on the flow-biased APRM flux scram trip setpoint and the SLMCPR protection. An increase to the void reactivity used in the GE stability analysis models (the frequency domain code ODYSY and the time-domain code TRACG) may also affect the predicted results. However, the current stability models have been used to model actual instability events, and the decay ratio acceptance criteria have been established consistent with the uncertainty as documented in the approved licensing reports. Furthermore, recent instability events at two domestic BWRs have also been evaluated with the stability models and shown to meet the previously established criteria. This provides high confidence that the GE methodology is adequately simulating recent fuel designs and fuel power densities. Therefore, no adjustment to stability models or analysis is necessary due to potential void reactivity uncertainties.

Exclusion Region Calculation

The NRC-approved ODYSY methodology (NEDC-32992P-A) is used in the exclusion region calculation for every reload [Reference 26]. The calculation of the exclusion region boundary is based on a very conservative core wide decay ratio ([[ ]]) that may be influenced by the core wide axial power distribution calculation. [[

]] An additional protection feature includes a cycle-specific buffer region, which is 5% in rated core power or 5% in rated core flow, beyond the exclusion region. Manual monitoring of the decay ratio is required while operating in the buffer region.

The decay ratio calculation includes a cycle-specific confirmation that core wide oscillation is the predominant reactor instability mode and that regional mode instability is not probable. The dominance of the core-wide mode oscillation is confirmed for every reload at the most limiting state point on the EPU power/flow map. The calculation to confirm that the regional mode of instability is not likely to be affected by uncertainties in power distribution because it considers the limiting bundle power. [[

]] Therefore, reasonable potential local or bundle power distribution uncertainties do not affect the confirmation that regional oscillations are not likely for plants with the Option I-D stability solution.

Detect and Suppress Calculation

The detect and suppress evaluation for Option I-D plants is performed under the approved LTR basis (NEDO-32465-A) [Reference 27]. The flow-biased APRM scram setpoints are initially established with conservative margin such that they are found applicable to future fuel cycles during reload confirmation calculations. The calculation of the scram setpoints is based on the limiting fuel bundle being at the Operating Limit MCPR (OLMCPR) and the SLMCPR not being exceeded during the instability oscillation.

The detect and suppress calculation requires the use of the DIVOM (which is defined as the Delta CPR over Initial MCPR Versus the Oscillation Magnitude) curve. Per the BWROG Guideline, Plant-Specific Core-Wide Mode DIVOM Procedure Guideline, [Reference 28] a plant and cycle-specific DIVOM evaluation is used to establish the plant specific relationship between the Hot Channel Oscillation Magnitude (HCOM) and the relative change in MCPR such that the initial MCPR value corresponds to the OLMCPR and the limiting MCPR value remains above the SLMCPR. [[

]]

[[

]] The scram setpoint analytical limit is established such that the hot channel power is maintained below acceptable values.

Bypass Voiding

The following discussion provides an assessment of the impact of bypass voiding on the effectiveness of the flow-biased APRM scram to provide SLMCPR protection for Option I-D. The primary effect of voiding in the bypass region on the neutron detectors (LPRMs and TIPs) is to reduce the detector response, assuming the same power in the adjacent fuel. This reduction is due to a decrease in the moderation caused by the presence of voids, which decreases the thermal neutron flux incident on the detectors for the same neutron flux generated in the adjacent fuel.

There is also the potential for some additional noise in the neutron flux signal, but that has a minor impact on steady state operation. These impacts are greatest for the highest elevation LPRM (D level) where the highest bypass voiding occurs.

For the Option I-D stability solution, the APRM flow-biased scram is used to mitigate stability transients. The analytical limit for the scram setpoint is based on assuring that the scram occurs before power oscillations become large enough to cause the MCPR to approach the SLMCPR. High bypass voids can potentially reduce the APRM reading, and so the margin to scram would increase and this could be non-conservative from the stability mitigation point of view since it would take higher amplitude oscillations to initiate an APRM scram.

The worst-case impact is at natural circulation (following a two recirculation pump trip) when the bypass voids are highest. An evaluation was performed at this condition for the Vermont Yankee plant (49.4% power and 31.3% core flow). [[

]]

The flow-biased APRM scram setpoint analytical limits are initially established with conservative margin such that they are found applicable to future fuel cycles during reload confirmation calculations. The calculation of the scram setpoint analytical limits is based on the limiting fuel bundle being at the OLMCPR and the SLMCPR not being exceeded during the power oscillation. The detect and suppress evaluation for Vermont Yankee Cycle 24 under EPU conditions was reevaluated to assess the impact of bypass voiding on the safety margins. The detect and suppress calculation assumes a flow runback along the rated licensing rodline to natural circulation flow. The flow-biased APRM trip analytical limit at natural circulation is 53.7% of rated power. [[

]] Hence, the SLMCPR is fully protected for Option 1-D plants, including the effects of bypass voiding.

The noise due to bypass voids slightly increases the overall APRM neutron noise at off-rated conditions where the voids may be significant. However, the impact of this noise on the APRM scram setpoint is negligible because the setpoint (derived from the analytical limit by considering noise and other instrument errors) is based on the normal (no void) noise at rated conditions (~2% of rated power), and this bounds the increased noise at off-rated conditions because the decrease in normal noise at off-rated conditions is more than the increase due to bypass voiding.

Additional detail can be found in the accepted VYNPS response for RAIs SRXB-A-44 and SRXB-A-55.

An assessment of the impact of the 40% void depletion history assumption on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] A similar assessment can be made for the axial and radial power distributions. Therefore, based on these assessments and those provided above, no adjustment to stability models or analysis is necessary due to potential void coefficient or power distribution uncertainties.

An assessment of the impact of extrapolating beyond 70% voids on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] Therefore, no adjustment to stability models or analysis is necessary due to potential void coefficient uncertainties.

There may be differences in bypass voiding between GE and non-GE fuel due to their geometric and lattice differences, however the impact on stability is insignificant because of the need for thermal-hydraulic compatibility of the fuel types in the core.

2.6.2.2 Option II

Option II has (1) a "prevention" element and (2) a "detect & suppress" element. The prevention portion of the solution includes an administratively controlled exclusion region, which is evaluated for every reload. The detect-and-suppress portion of the solution is a quadrant-based flow-biased APRM flux scram trip that prevents oscillations of significant magnitude. This scram ensures the Fuel Cladding Integrity SLMCPR is protected for both the core wide and regional modes of coupled thermal-hydraulic/neutronic reactor instability. Option II differs from Option I-D in that it has no buffer region and the quadrant-based APRM is able to detect both regional and core-wide mode oscillations.

Stability analyses for both the EPU and fuel cycle specific conditions are performed to define the exclusion region as well as to confirm that the scram setpoints meet the design basis. With respect to power distribution uncertainties of the nuclear simulator data, the results pertaining to the exclusion region may be slightly affected, but this is not considered to have any safety significance for reasons described below. The power distribution uncertainties of the nuclear simulator data are considered in the determination of the limiting bundle conditions and therefore have insignificant impact on the flow-biased APRM flux scram trip setpoint and the SLMCPR protection. An increase to the void reactivity used in the GE stability analysis models (the frequency domain code ODYSY and the time-domain code TRACG) may also affect the predicted results. However, the current stability models have been used to model actual instability events, and the decay ratio acceptance criteria have been established consistent with the uncertainty as documented in the approved licensing reports. Furthermore, recent instability events at two domestic BWRs have also been evaluated with the stability models and shown to meet the previously established criteria. This provides high confidence that the GE methodology is adequately simulating recent fuel designs and fuel power densities. Therefore, no adjustment to stability models or analysis is necessary due to potential void reactivity uncertainties.

Exclusion Region Calculation

The NRC-approved ODYSY methodology [Reference 26] is used in the exclusion region calculation for every reload. The calculation of the exclusion region boundary is based on a very conservative core wide decay ratio ([[ ]]) that may be influenced by the core wide axial power distribution calculation. [[

]]

Detect and Suppress Calculation

The detect and suppress evaluation for Option II plants is performed under the approved LTR basis [Reference 27]. The flow-biased APRM scram setpoints are initially established with conservative margin such that they are found applicable to future fuel cycles during reload confirmation calculations. The calculation of the scram setpoints is based on the limiting fuel bundle being at the OLMCPR and the SLMCPR not being exceeded during the instability oscillation.

The detect and suppress calculation requires the use of the DIVOM curve. Per the BWROG Guideline, "Plant-Specific Regional Mode DIVOM Procedure Guideline" [Reference 29], a plant- and cycle-specific DIVOM evaluation is used to establish the plant specific relationship between the HCOM and the relative change in MCPR such that the initial MCPR value corresponds to the OLMCPR and the limiting MCPR value remains above the SLMCPR. [[

]]

[[

]] The scram setpoint analytical limit is established such that the hot channel power is maintained below acceptable values.

Bypass Voiding

The bypass voiding discussion provided in Section 2.6.2.1 for Option I-D is fully applicable to Option II because both stability solutions use the flow-biased APRM scram to provide SLMCPR protection.

2.6.2.3 Option III

Option III is a "detect & suppress" solution that combines closely spaced Local Power Range Monitor (LPRM) detectors into Oscillation Power Range Monitor (OPRM) "cells" to detect either core-wide or regional (local) modes of reactor instability. The detect and suppress evaluation for Option III plants is performed under the approved LTR basis [Reference 27]. The OPRM scram setpoints are established such that the SLMCPR is not exceeded during the instability oscillation.

The examination of core and fuel stability behavior begins with fuel assumed to be at the OLMCPR and terminates once power oscillations cause fuel critical power to reach the SLMCPR. Therefore, if any uncertainties are increased and applied to the SLMCPR, they are directly incorporated into the stability methodology. As discussed before in relation to nodal and core reactivity, uncertainties or biases in depletion isotopics at high exposure and void conditions from prediction, which might have a postulated effect on the void reactivity coefficient, would manifest themselves in separately observable differences in local and core power and reactivity. The variation of void reactivity coefficient across the GE BWR fleet encompasses significant variations in bundle and core exposures and void fraction and is well behaved. The effect of the void reactivity coefficient on instability events is well understood via existing code qualification parametric studies. Large unknown uncertainties in the void reactivity coefficient would be noticeable and be manifest as an inability to reasonably model instability events. The existing GE thermal-hydraulic stability models reasonably and adequately model the magnitude and period of industry thermal-hydraulic instability events. Both the GE stability codes (frequency domain code ODYSY and time-domain code TRACG) model past events relatively well, including the recent thermal-hydraulic instability events at two domestic BWRs. This demonstrates the accuracy of the void model in the GE methodology and provides high confidence in the simulation of recent fuel designs and fuel power densities. Because the transient analysis results (delta/initial) are not affected and the difference between OLMCPR and SLMCPR remains unchanged, the stability envelope will not be affected.

Key inputs to the stability-based OLMCPR analysis are the DIVOM slope and HCOM. These inputs would not be affected by an increase in the OLMCPR or the SLMCPR. Key HCOM inputs are LPRM to OPRM assignments, total scram delay time, RPS trip logic, and averaging/conditioning filter cutoff frequencies. A new HCOM is required only if one of these key (but unrelated to OLMCPR or SLMCPR) parameters changes.

Further, a 5-10% uncertainty in radial peaking factor is applied in this analysis, primarily to address variations in bundle peaking from initial rod pattern selection. This relatively large radial peaking factor reasonably encompasses the small (<~1%) increase in bundle power uncertainty (described above) for the SLMCPR determination, in particular because the stability analysis is otherwise conservative for plant specific conditions or settings.

Per the BWROG Guideline, “Plant-Specific Regional Mode DIVOM Procedure Guideline” [Reference 29], a plant- and cycle-specific DIVOM evaluation is used to establish the plant specific relationship between HCOM and the relative change in MCPR such that the initial MCPR value corresponds to the OLMCPR and the limiting MCPR value remains above the SLMCPR. [[

]]

[[

]] The scram setpoint analytical limit is established such that the hot channel power is maintained below acceptable values.

Bypass Voiding

The following discussion provides an assessment of the impact of bypass voiding on the effectiveness of the OPRM scram to provide SLMCPR protection for Option III. The primary effect of voiding in the bypass region on the neutron detectors (LPRMs and TIPs) is to reduce the detector response, assuming the same power in the adjacent fuel. This reduction is due to a decrease in the moderation caused by the presence of voids, which decreases the thermal neutron flux incident on the detectors for the same neutron flux generated in the adjacent fuel. There is also the potential for some additional noise in the neutron flux signal, but that has a minor impact on steady state operation. These impacts are greatest for the highest elevation LPRM (D level) where the highest bypass voiding occurs.

For the Option III stability solution, the OPRM scram is used to mitigate stability transients. The scram setpoint is based on assuring that the scram occurs before power oscillations become large enough to cause the MCPR to approach the SLMCPR. High bypass voids can potentially reduce the OPRM reading, and so the margin to scram would increase and this could be non-conservative from the stability mitigation point of view since it would take higher amplitude oscillations to initiate an OPRM scram.

The worst-case impact is at natural circulation (following a two recirculation pump trip) when the bypass voids are highest. An evaluation was performed at 49.4% power and 31.3% core flow for a BWR/4 with 764 fuel assemblies at 120% OLTP MELLLA operation. [[

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The D and C level LPRM detectors may also indicate additional noise due to the void bubbles in the bypass region. The frequency of this noise is inversely related to the bubble transit time across the LPRM detector (~ 2 inches). For a typical bypass flow velocity at natural circulation of 0.4 ft/sec, the noise frequency is ~2.4 Hz. This noise due to bypass voids has a negligible impact on the ability of the Option III detection algorithms to detect instability oscillations because the noise is high frequency (~2.4 Hz) and is effectively filtered out by the double pole Butterworth "cut-off" filter (~1 Hz) in the OPRM equipment.

An assessment of the impact of the 40% void depletion history assumption on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] A similar assessment can be made for the axial and radial power distributions. Therefore, based on these assessments and those provided above, no adjustment to stability models or analysis is necessary due to potential void coefficient or power distribution uncertainties.

An assessment of the impact of extrapolating beyond 70% voids on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] Therefore, no adjustment to stability models or analysis is necessary due to potential void coefficient uncertainties.

There may be differences in bypass voiding between GE and non-GE fuel due to their geometric and lattice differences, however the impact on stability is insignificant because of the need for thermal-hydraulic compatibility of the fuel types in the core.

2.6.2.4 Enhanced Option I-A

Enhanced Option I-A (EIA) is a "prevention" solution that automatically prevents reactor operations within an Exclusion Region by modifying the flow-biased APRM flux scram function to contain this region. This scram ensures the Fuel Cladding Integrity SLMCPR is protected for both the core wide and regional modes of coupled thermal-hydraulic/neutronic reactor instability. Reactor operations within a Restricted Region are automatically restricted by modifying the flow-biased APRM control rod block function to contain this region. An administratively controlled Monitored Region provides additional protection outside of the Restricted Region.

Stability analyses for both the EPU and fuel cycle specific conditions are performed to define the stability region boundaries as well as to confirm that the scram setpoints meet the design basis. With respect to power distribution uncertainties of the nuclear simulator data, the results pertaining to the region boundaries may be slightly affected, but this is not considered to have any safety significance for reasons described below. The power distribution uncertainties of the nuclear simulator data are considered in the determination of the limiting bundle conditions and therefore have insignificant impact on the flow-biased APRM flux scram trip setpoint and the SLMCPR protection. An increase to the void reactivity used in the GE stability analysis model (the frequency domain code ODYSY) may also affect the predicted results. However, the current stability model has been used to model actual instability events, and the decay ratio acceptance criteria have been established consistent with the uncertainty as documented in the approved licensing reports. Furthermore, recent instability events at two domestic BWRs have also been evaluated with the stability model and shown to meet the previously established criteria. This provides high confidence that the GE methodology is adequately simulating recent fuel designs and fuel power densities. Therefore, no adjustment to stability models or analysis is necessary due to potential void reactivity uncertainties.

Region Boundary Calculations

The NRC-approved ODYSY methodology [Reference 30] is used in the region boundary calculations for every reload. The calculation of the region boundaries is based on conservative decay ratio criteria that may be influenced by the core wide axial power distribution calculation. [[

]]

Bypass Voiding

The bypass voiding discussion provided in Section 2.6.2.1 for Option I-D is fully applicable to EIA because both stability solutions use the flow-biased APRM scram to provide SLMCPR protection. In addition, the EIA solution makes use of a 40% flow clamp such that a scram is initiated if core flow falls below 40% of rated. There is less bypass voiding at 40% flow than at natural circulation, so bypass voiding is less significant for EIA than for Option I-D.

2.6.3 Adequacy of Existing Treatment and Alternate Approach

The uncertainties in power distribution calculation and void reactivity do not significantly affect the safety margin in the stability analysis. Additional detail can be found in the accepted VYNPS response for RAIs SRXB-A-13, SRXB-A-14, and SRXB-A-15.

2.7 LICENSED EXPOSURE

GE fuel designs are licensed to a [[ ]] exposure limit (i.e., 70 GWd/MTU for GE14). [Reference 31] This is equivalent to a GE14 rod average exposure of [[ ]], although an explicit rod average exposure limit is not specified for GE14 or other GE fuel designs. This exposure limit is specified and applied in the process computer to assure that fuel is not operated beyond its analyzed basis. In this application, the best estimate value of the [[ ]] exposure condition is monitored against the specified exposure limit.

2.7.1 Fuel Parameters That Affect Pellet Exposure

The fuel parameters and associated uncertainties identified previously (i.e., the local pin power peaking, void reactivity coefficient, [[ ]]) are included in the development of the LHGR Operating Limits, and the fuel exposure limit. These fuel parameters ultimately determine both the local power and local exposure.

2.7.2 Treatment of Fuel Parameter Uncertainties

The fuel rod thermal-mechanical performance consideration of greatest interest at exposures near the [[ ]] exposure limit is the fuel rod internal pressure. [[

]] therefore, no additional conservatism in local exposure monitoring is required to maintain fuel integrity.

2.7.3 Adequacy of Existing Treatment and Alternate Approach

In summary, the GE standard fuel thermal-mechanical analysis basis considers and provides adequate margin for uncertainties in local and bundle power and exposure. Additional supporting information is provided in the response to SRXB-A-65.

3.0 EXTENSION OF SAFETY PARAMETER BASES TO THE MELLLA+ OPERATING DOMAIN

3.1 INTRODUCTION

Since the early 1980s, the BWR fleet has commonly used an operating strategy known as spectral shift operation. Spectral shift refers to promoting Pu-239 buildup early in the cycle by favoring a "harder" neutron energy spectrum (i.e., increasing voids). This is achieved by overemphasizing the bottom peak in the core axial power shape. The overemphasized bottom peak is attained through reduced core flow, or control rod patterns, or through the enrichment and burnable poison distributions designed into the fuel, or through combinations of all these tactics. Reducing flow to promote spectral shift is generally favored over tactics such as power shaping with control rods.

MELLLA+ operation allows the reactor to be at full power down to 80% of core rated flow [Reference 36]. Like Extended Power Uprate, (EPU), these conditions increase the amount of steam voids in the core. The void amount is a direct function of the power to flow ratio. Raising the average bundle power (EPU) or lowering the flow (MELLLA+) have the same affect, and for the most part raise similar technical issues. This section addresses those technical issues unique to MELLLA+ operation.

3.2 CRITICAL POWER

3.2.1 Safety Limit Critical Power Ratio (SLMCPR)

The approach for the SLMCPR evaluation applied to MELLLA+ operating conditions is the same (with respect to the process) as described under Section 2.2.1. This process was modified in 2004 as part of the resolution to a Part 21 on SLMCPR [Reference 32]. The MELLLA+ operating domain has an additional high power state point that is considered in the evaluation. The current design process for determining the cycle-specific SLMCPR considers the highest licensed power level at two flow points, rated flow and the lowest licensed flow at 100% power (e.g., ~80% flow for MELLLA+ operation). These power/flow state points are considered at (minimum) three exposure points in the cycle, for a total of 6 evaluation points. The SLMCPR determined using this approach is appropriately conservative to cover the MELLLA+ power/flow operating conditions [Reference 33].

3.2.2 Operating Limit Critical Power Ratio (OLMCPR)

MELLLA+ evaluation procedures require consideration of OLMCPR values for each limiting corner of the power flow map. If changes are required to account for OLMCPR at different flow points, this change is reflected in the process computer algorithm for MFLCPR (Ratio of bundle critical power to OLMCPR) for each bundle. The same conservatisms apply for the nuclear inputs to the transient evaluations. The sensitivities remain the same as those evaluated at the full power conditions.

3.3 SHUTDOWN MARGIN

It should be noted that the data in Section 2.3 supports a 2σ demonstration margin criteria of 0.38% $\Delta k/k$. This is done by showing that the same core 1σ spread for the [[ ]] cores is [[ ]]. The cores comprising this dataset are all high energy, modern fuel, spectral shift operation. Relative to steady state methods, MELLLA+ operation is a method of spectral shift operation. The [[ ]] from the spectral shift, high energy cores is less than the [[ ]] from early cores reported in Reference 22 for earlier versions of PANACEA and essentially the same as the [[ ]] for the current version of PANACEA reported in [Reference 34] for a broader, fleet-wide statistical assessment of cold eigenvalues for plants covering a range of operating conditions, but without a large representation of high energy density cores (such cores were not prevalent at that time). The similarity in the cold eigenvalue variation for the various populations indicates that the methods have maintained fidelity in cold eigenvalue prediction, even as core and fuel advances have been made.

3.4 FUEL ROD THERMAL MECHANICAL PERFORMANCE

One of the benefits of MELLLA+ operation is that it supports spectral shift operation, wherein the flow is reduced early in the cycle to promote a bottom peaked axial power shape. Spectral shift operation has the potential to increase axial peaking lower in the core at BOC, then in the upper portion of the core near EOC. The fuel rod thermal-mechanical analyses explicitly address the variation in the axial power distribution that may occur as a result of spectral shift operation, and therefore the specified LHGR Operating Limits and exposure limit are directly applicable to MELLLA+ operation.

3.5 LOCA Related Nodal Power Limits

There are no differences in the ECCS-LOCA methodology between EPU and MELLLA+ except that for MELLLA+ the ECCS-LOCA analyses are performed for at least two additional state points. MELLLA+ ECCS-LOCA analyses will include calculations for the rated power/MELLLA+ boundary point and the low flow point on the MELLLA+ boundary at which the off-rated flow dependent LHGR or MAPLHGR setdown begins to apply. The Licensing Basis PCT is based on the analyzed state point with the highest PCT using Appendix K assumptions.

3.6 Stability

The GE BWR Detect and Suppress Solution – Confirmation Density (DSS-CD) (NEDC-33075P, Revision 6) is the only licensed stability solution for operation in the MELLLA+ domain [Reference 35]. DSS-CD is a “detect & suppress” solution and represents an evolutionary step from Stability Solution Option III (see Section 2.6.2.3). DSS-CD introduces an enhanced detection algorithm, the Confirmation Density Algorithm (CDA), which reliably detects the inception of power oscillations and generates an early power suppression trip signal prior to any significant oscillation amplitude growth and MCPR degradation.

TRACG analysis is performed to demonstrate significant margin to the SLMCPR for the generic OPRM CDA setpoints. Conservative multipliers are applied to the TRACG results in the assessment of the CPR margin for limiting instability scenarios. These multipliers accommodate the uncertainties in power distribution and void reactivity. The DSS-CD LTR defines a generic applicability envelope for MCPR margin such that a similar increase in the SLMCPR and the OLMCPR will not affect the applicability of DSS-CD.

In summary, the DSS-CD has been designed for the MELLLA+ domain and uncertainties in power distribution calculation and void reactivity are accounted for in the significant safety margin in the stability analysis.

Bypass Voiding

The following discussion provides an assessment of the impact of bypass voiding on the effectiveness of the OPRM scram to provide SLMCPR protection for DSS-CD. The primary effect of voiding in the bypass region on the neutron detectors (LPRMs and TIPs) is to reduce the detector response, assuming the same power in the adjacent fuel. This reduction is due to a decrease in the moderation caused by the presence of voids, which decreases the thermal neutron flux incident on the detectors for the same neutron flux generated in the adjacent fuel. There is also the potential for some additional noise in the neutron flux signal, but that has a minor impact on steady state operation. These impacts are greatest for the highest elevation LPRM (D level) where the highest bypass voiding occurs.

For the DSS-CD stability solution, the OPRM scram is used to mitigate stability transients. The scram setpoint is based on assuring that the scram occurs before power oscillations become large enough to cause the MCPR to approach the SLMCPR. High bypass voids can potentially reduce the OPRM reading, and so the margin to scram would increase and this could be non-conservative from the stability mitigation point of view since it would take higher amplitude oscillations to initiate an OPRM scram.

The worst-case impact is at natural circulation (following a two recirculation pump trip) when the bypass voids are highest. An evaluation was performed at this condition for the highest power density BWR type (~60% power and ~30% core flow) with 120% uprated MELLLA+ operation. [[

]]

The D and C level LPRM detectors may also indicate additional noise due to the void bubbles in the bypass region. The frequency of this noise is inversely related to the bubble transit time across the LPRM detector (~ 2 inches). For a typical bypass flow velocity at natural circulation of 0.4 ft/sec, the noise frequency is ~2.4 Hz. This noise due to bypass voids has a negligible impact on the ability of the DSS-CD detection algorithm to detect instability oscillations because the noise is high frequency (~2.4 Hz) and is effectively filtered out by the double pole Butterworth "cut-off" filter (~1 Hz) in the OPRM equipment.

An assessment of the impact of the 40% void depletion history assumption on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] A similar assessment can be made for the axial and radial power distributions. Therefore, based on these assessments and those provided above, no adjustment to stability models or analysis is necessary due to potential void coefficient or power distribution uncertainties.

An assessment of the impact of extrapolating beyond 70% voids on stability can be summarized as follows. As stated in Section 2.2.2.2, [[

]] Therefore, no adjustment to stability models or analysis is necessary due to potential void coefficient uncertainties.

There may be differences in bypass voiding between GE and non-GE fuel due to their geometric and lattice differences, however the impact on stability is insignificant because of the need for thermal-hydraulic compatibility of the fuel types in the core.

3.7 LICENSED EXPOSURE

As noted in Section 3.4, spectral shift operation has the potential to increase axial peaking lower in the core at BOC, then in the upper portion of the core near EOC. The fuel rod thermal-mechanical analyses explicitly address the variation in the axial power distribution that may occur as a result of spectral shift operation, and therefore the specified LHGR Operating Limits and exposure limit derived from the fuel rod thermal-mechanical analyses are directly applicable to MELLLA+ operation.

4.0 LICENSING APPLICATION

4.1 OVERVIEW

The purpose of the Applicability of GE Methods to Expanded Operating Domains Licensing Topical Report (LTR) is to provide a licensing basis that allows the NRC to issue Safety Evaluations (SEs) for Constant Pressure and Extended Power Uprate (CPPU, EPU) applications and the MELLLA+ LTR. The SE for the Applicability of GE Methods to Expanded Operating Domains LTR would approve the use of GE's methods for extended power uprates (EPU or CPPU) and MELLLA+ operating domain expansion until final resolution of the Methods RAIs.

The Applicability of GE Methods to Expanded Operating Domains LTR is for temporary application and it is expected that it would be necessary for only a limited number of utility license applications until the NRC's review of the Methods RAIs is complete. GE anticipates that a limited number of future license applications, associated with extended power uprate and MELLLA+, will reference the Applicability of GE Methods to Expanded Operating Domains LTR. GE intends to resolve the Methods RAIs as soon as practical and thereby eliminate the need for referencing the Applicability of GE Methods to Expanded Operating Domains LTR in the long term.

4.2 APPLICABILITY

The Applicability of GE Methods to Expanded Operating Domains LTR basis is applicable to current GE BWR product lines licensed with GE nuclear and safety analysis methods. The Methods LTR is applicable to plants that include current GE and non-GE legacy fuel designs. The Methods LTR is applicable to plants seeking NRC approval for CPPU and EPU power uprates, and MELLLA+ operating domain expansion, including currently licensed operating domains and operational flexibility features. The Methods LTR is applicable to plants applying licensed GE Stability Solutions.

Each GE technology code has an associated “application statement” defining the application range. The application of these codes complies with the limitations, restrictions and conditions specified in the approving NRC SER for each code.

The parameters establishing the Applicability of GE Methods to Expanded Operating Domains applicability envelope are:

Parameter	Generic Value
BWR Product Line	BWR/2-6
Fuel Product Line	GE and non-GE fuel designs using square arrays of fuel rods, including 7x7, 8x8, 9x9, and 10x10 designs
Licensing Methodology	GE Nuclear and Safety Analysis Methods
Operating Domain	CPPU, EPU, with MELLLA+ including currently licensed operating domains (e.g., ELLA, MELLLA) and operational flexibility features
Maximum Rated Power Level	120% OLTP
Stability Solution	GE Stability Solutions

The evaluations documented in this report, demonstrating the acceptability of the margins associated with the Applicability of GE Methods to Expanded Operating Domains, encompass the above applicability envelope parameters. The plant specific application process will confirm that operations proposed by the plant specific license amendment meet the Applicability of GE Methods to Expanded Operating Domains LTR applicability envelope requirements.

4.3 PLANT SPECIFIC APPLICATION PROCESS

Each plant seeking to apply the Methods LTR must provide information supporting the application that demonstrates that the plant parameters are within the applicability definition in Section 4.2.

5.0 SUMMARY AND CONCLUSION

Except for the change in additional SLMCPR margin required per NEDC-33173P Revision 0 and its Safety Evaluation (SE), all other Limitations and Conditions of the Revision 0 SE remain applicable to Revision 2 of NEDC-33173P.

Safety Limit Critical Power Ratio (SLMCPR)

Confirmatory gamma scan data (References 38, 39, and 40) has been provided for 10x10 fuel designs at original licensed and power uprate conditions which demonstrate the adequacy of the power distribution uncertainties for the SLMCPR process. It has also been demonstrated that no adjustments or additional justification is required for mixed core applications.

Operating Limit Critical Power Ratio (OLMCPR)

Adequate conservatism in the analyses that establish the OLMCPR is demonstrated. Therefore, no additional margin to the OLMCPR is required.

Shutdown Margin (SDM)

The Technical Specification (TS) limit for the SDM of 0.38 % $\Delta k/k$ is not increased for CPPU or EPU and MELLLA+ applications. The uncertainty does not increase to a degree that warrants an increase in the TS limit. GE normally provides 1% SDM in the core design.

Fuel Rod Thermal-Mechanical Performance

Adequate overall modeling uncertainties are included within the current design basis for generation of the LHGR Operating Limits and exposure limit. Therefore, no changes are required in the LHGR Operating Limits and exposure limit.

LOCA Related Nodal Power Limits

The conservatisms applied in the calculation of the limit in the ECCS-LOCA calculations provide justify the adequacy of current methodology for application in CPPU or EPU and MELLLA+ applications. Therefore, no additional margin is applied to the MAPLHGR limit.

Stability

The effectiveness of the neutron monitoring systems and detect and suppress methodologies is not significantly affected by postulated increases in bypass voiding for CPPU or EPU applications including MELLLA+.

Licensed Exposure

Adequate overall modeling uncertainties are included within the current design basis for generation of the LHGR Operating Limits and exposure limit. Therefore, no changes are required in the LHGR Operating Limits and exposure limit.

6.0 REFERENCES

1. Letter from George Stramback (GE) to NRC, Submittal of GE Proprietary Licensing Topical Report NEDC-33006P, Revision 1, General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus, August 2002, MFN 02-050, August 23, 2002. (The Acceptance version of NEDC-33006P is included as Reference 36.)
2. Entergy letter to U.S. Nuclear Regulatory Commission, Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263, Extended Power Uprate " BVY 03-80, September 10, 2003.
3. Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 24, Extended Power Uprate - Response to Request for Additional Information," BVY 05-024, March 10,2005.
4. Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 30, Extended Power Uprate - Response to Request for Additional Information," BVY 05-072, August 1,2005.
5. Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 32, Extended Power Uprate - Response to Request for Additional Information," BVY 05-083, September 10,2005.
6. Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 34, Extended Power Uprate - Response to Request for Additional Information," BVY 05-086, September 18,2005.
7. Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 35, Extended Power Uprate - Response to Request for Additional Information," BVY 05-088, September 28,2005.

8. GE Nuclear Energy, ”Constant Pressure Power Uprate Licensing Topical Report,” NEDC-33004P-A, Revision 4, July 2003.

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38. GE Hitachi Nuclear Energy, NEDC-33173P, Supplement 2 Part 1, Licensing Topical Report, Applicability of GE Methods to Expanded Operating Domains – Power Distribution Validation for Cofrentes Cycle 13, August 2009.

39. GE Hitachi Nuclear Energy, NEDC-33173P, Supplement 2 Part 2, Licensing Topical Report, Applicability of GE Methods to Expanded Operating Domains – Pin-by-Pin Gamma Scan at FitzPatrick October 2006, August 2009.

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APPENDIX A VERMONT YANKEE REACTOR SYSTEMS BRANCH QUESTIONS

Appendix A includes a profile of the questions from the NRC Reactor Systems Branch that were recently addressed on the VYNPS EPU docket. Some of the RAIs are not related to GE methods and some are questions seeking specific VYNPS information. The following table presents the VYNPS reference letters and associated RAI responses.

Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 24, Extended Power Uprate - Response to Request for Additional Information," BVY 05-024, March 10,2005.	Attachment 3 – SRXB-A-6
Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 30, Extended Power Uprate - Response to Request for Additional Information," BVY 05-072, August 1,2005.	Attachment 1 – Revised SRXB-A-6 Attachment 9 – SRXB-A-7 thru SRXB-A-58
Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 32, Extended Power Uprate - Response to Request for Additional Information," BVY 05-083, September 10,2005.	Attachment 4 – Revised SRXB-A-17 Attachment 5 – SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70
Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 34, Extended Power Uprate - Response to Request for Additional Information," BVY 05-086, September 18,2005.	Attachment 2 – SRXB-A-66 Data CD Attachment 3 – Supplement to SRXB-A-64 Attachment 4 – SRXB-A-65 and 67 Attachment 6 – SRXB-A-71
Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, License No. DPR-28 (Docket No. 50-271), Technical Specification Proposed Change No. 263 - Supplement No. 35, Extended Power Uprate - Response to Request for Additional Information," BVY 05-088, September 28,2005.	Attachment 1 – SRXB-A-68

The RAIs are presented in two tables that follow: the first is sorted by RAI number, and the second by technology grouping. The subject column provides the subject and a few words regarding the response and resolution. There is a group of RAIs labeled "Not Methods Related" that are not relevant to the technologies addressed with the Applicability of GE Methods to Extended Operating Domains LTR.

RAI Sort By Number		
Related Technology	**Subject**	**RAI**
Steady state and transient nuclear, Steady state and transient thermal hydraulic, fuel rod mechanical	Strategy for Application of Methods to design and addition SLMCPR margin to account for lack of experimental data	SRXB-A-06
Not Methods Related	The code used for over pressure protection analysis was identified	SRXB-A-07
Not Methods Related	Criteria for single loop operation. Not design basis requirement	SRXB-A-08
PANACEA, ISCOR	Refer to SRXB-A-19 for Representative Core definition	SRXB-A-09
Not Methods Related	Information supplied regarding PCT difference in VYNPS LBLOCA analysis	SRXB-A-10
TGBLA, MCNP	Information supplied on storage safety requirements -- related to limit on lattice reactivity	SRXB-A-11
Not Methods Related	Information supplied regarding revision of CST minimum volume	SRXB-A-12
ODYSY	Clarification of Stability Solutions. Information supplied relating plant to overall stability requirements	SRXB-A-13
ODYSY	Justify that hot bundle oscillation not dependent on core design. Information supplied discussing influence of core design on hot bundle oscillation	SRXB-A-14
ODYSY	Have EPU core loadings degraded stability? Cycle decay ratios compared	SRXB-A-15
Not Methods Related	Information supplied on APRM flow biased scram set points for EPU	SRXB-A-16
Not Methods Related	Uncontrolled rod withdrawal considered an accident rather than a transient. Energy deposition limit is consistent with a transient	SRXB-A-17
Not Methods Related	Confirm that GDC-9 is applicable. Refers to template markups in BVY 05-072 Attachment 11	SRXB-A-18
Not Methods Related	Flow dependant limits are confirmed for each cycle	SRXB-A-19
Not Methods Related	Explanation given and information supplied regarding SLCS pump discharge pressure	SRXB-A-20
Not Methods Related	NRC approved evaluation model LFW events identified.	SRXB-A-21
Not Methods Related	NRC staff approved evaluation model identified for ATWS and discussion provided on EOPs	SRXB-A-22
Not Methods Related	Requested data for VYNPS supplied	SRXB-A-23
PANACEA, ISCOR	Explanation supplied for the uncertainties applied to LHGR. Refer to SRXB-A-68	SRXB-A-24
PANACEA, ISCOR	Explanation provided for increase in nodal uncertainties with elevation	SRXB-A-25
Not Methods Related	Demonstrate that core will not operate with power flow ratio greater than 50Mwt/Mlbm/hr. VYNPS evaluated to be less than 40Mwt/Mlb/hr	SRXB-A-26

RAI Sort By Number		
Related Technology	**Subject**	**RAI**
PANACEA, ISCOR	Information and discussion supplied regarding criteria for axial and nodal uncertainties	SRXB-A-27
PANACEA, ISCOR	Information and discussion of SLMCPR evaluation and monitoring supplied for axial and nodal uncertainties in safety limit analyses	SRXB-A-28
PANACEA, ISCOR ODYN, SAFER	Explanation provided for inclusion of axial and nodal uncertainties in transient and accident evaluations	SRXB-A-29
PANACEA, ISCOR	Application of nodal uncertainties to nodal exposure to MAPLHGR and LHGR values	SRXB-A-30
PANACEA, ISCOR	Does LHGR limit in 3D simulator include decrease with exposure	SRXB-A-31
PANACEA, ISCOR	Application of nodal uncertainties and increases with exposure. Refer to SRXB-6 and SRXB-31.	SRXB-A-32
PANACEA, ISCOR	Describe how core monitoring system calculate pin wise power parameters	SRXB-A-33
PANACEA, ISCOR	Justify acceptability of basing assessment of pin power accuracy on code-to-code comparisons. Alternate approach and SLMCPR procedures proposed in response to SRXB-6	SRXB-A-34
PANACEA, ISCOR	Core Follow Data Supplied	SRXB-A-35
PANACEA, ISCOR	Reasons for differences between PCTIP and axial power distributions provided	SRXB-A-36
TGBLA, MCNP	Explanation provided to justify acceptability of basing assessment of pin power accuracy on BOL conditions	SRXB-A-37
TGBLA, MCNP	Explanation provided for use of different uncertainties for GE14 and later designs. Refer to response to SRXB-6	SRXB-A-38
PANACEA, ISCOR	Explanation of effect on pin power due to neighboring bundles provided with explicit results for 10x10 lattices	SRXB-A-39
SLMCPR	Provided confirmation that current channel bow uncertainties are included in SLMCPR evaluations	SRXB-A-40
SLMCPR	Provide uncertainty analysis for 3D MONICORE	SRXB-A-41
SLMCPR	Provided explanation of R-factor uncertainty procedures	SRXB-A-42
SLMCPR	Justification of Inlet Sub cooling Uncertainties provided	SRXB-A-43
PANACEA, ISCOR	Discussion of bypass voiding on instrumentation provided	SRXB-A-44
SLMCPR	Explanation provided regarding why axial TIP not included in SLMCPR	SRXB-A-45
TGBLA, MCNP	Explanation provided regarding Cross Sections for High void operation. Refer to generic EPU and MELLLA+ studies.	SRXB-A-46
TGBLA, MCNP	Plots of isotopic concentrations provided	SRXB-A-47
TGBLA, MCNP	Information provided on the isotopic influence on void coefficient	SRXB-A-48
GEXL	Double Hump Power distributions for GEXL accounted for in SLMCPR calculations	SRXB-A-49

RAI Sort By Number		
Related Technology	**Subject**	**RAI**
GEXL	Power flow ranges for GEXL shown to be adequate	SRXB-A-50
TGBLA, MCNP	Discussion provided on Void reactivity coefficients for transients and accidents, including ATWS and SBO. Refer to SRXB-A-6	SRXB-A-51
Void and pressure drop correlations	Pressure Drop data base information provided, reference made to generic MELLLA+ report	SRXB-A-52
Void and pressure drop correlations	Void fraction measurement data made through Safety Limit Document reference	SRXB-A-53
Void and pressure drop correlations	Are void fraction uncertainties included in water density? Explanation provided	SRXB-A-54
Instrument effects	Effect high void fractions on instrument response during transients. Effects of bypass voids on instrument response explained	SRXB-A-55
Instrument effects	Explanation provided for impact of instrument random noise during plant maneuvers	SRXB-A-56
Not Methods Related	More detailed explanation provided for Reactivity events	SRXB-A-57
ODYN	Explanation of uncertainties in power during transients	SRXB-A-58
Not Methods Related	Clarified the single loop operation of shutdown cooling (SDC) in the VYNPS Appendix R analysis.	SRXB-A-59
Not Methods Related	Explanation provided for equilibrium and representative cycle core terms	SRXB-A-60
TGBLA, MCNP	Explanation provided on the effect of EPU on spent fuel storage Refer to SRXB-A-11	SRXB-A-61
Not Methods Related	Explained expression in TS 3.4.3. Information provided supporting the value of 1.29 at EPU conditions.	SRXB-A-62
TGBLA, MCNP	Describe transients used to determine MCPR	SRXB-A-63
PANACEA, ISCOR	Information provided for maximum bundle power and power density before and after EPU	SRXB-A-64
GSTRM (GESTR-Mechanical)	Uncertainties in LHGR limit evaluations	SRXB-A-65
TGBLA, MCNP	CASMO/TGBLA code comparisons	SRXB-A-66
PANACEA, ISCOR	Shutdown margin verification and qualification Data and procedure provided	SRXB-A-67
TGBLA, MCNP	Void reactivity coefficients -- provide more information than response to SRXB-A-51	SRXB-A-68
Void and pressure drop correlations	Explanation and information provided regarding Void fraction uncertainties	SRXB-A-69
Void and pressure drop correlations	Explanation provided regarding acceptable to exceed correlations range. Refer to SRXB-A-55	SRXB-A-70
TGBLA, MCNP	Clarification and more detail on response to SRXB-A-57	SRXB-A-71

RAI Sort by Related Technology		
Related Technology	**Subject**	**RAI**
GSTRM (GESTR-Mechanical)	Uncertainties in LHGR limit evaluations	SRXB-A-65
GEXL	Double Hump Power distributions for GEXL accounted for in SLMCPR calculations	SRXB-A-49
GEXL	Power flow ranges for GEXL shown to be adequate	SRXB-A-50
Instrument effects	Effect high void fractions on instrument response during transients. Effects of bypass voids on instrument response explained	SRXB-A-55
Instrument effects	Explanation provided for impact of instrument random noise during plant maneuvers	SRXB-A-56
Not Methods Related	Criteria for single loop operation. Not design basis requirement	SRXB-A-08
Not Methods Related	Information supplied regarding revision of CST minimum volume	SRXB-A-12
Not Methods Related	Information supplied on APRM flow biased scram set points for EPU	SRXB-A-16
Not Methods Related	Uncontrolled rod withdrawal considered an accident rather than a transient. Energy deposition limit is consistent with a transient	SRXB-A-17
Not Methods Related	Confirm that GDC-9 is applicable. Refers to template markups in BVY 05-072 Attachment 11	SRXB-A-18
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Not Methods Related	Requested data for VYNPS supplied	SRXB-A-23
Not Methods Related	Demonstrate that core will not operate with power flow ratio greater than 50Mwt/Mlbm/hr. VYNPS evaluated to be less than 40Mwt/Mlb/hr	SRXB-A-26
Not Methods Related	More detailed explanation provided for Reactivity events	SRXB-A-57
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PANACEA, ISCOR	Application of nodal uncertainties and increases with exposure. Refer to SRXB-6 and SRXB-31.	SRXB-A-32
PANACEA, ISCOR	Describe how core monitoring system calculate pin wise power parameters	SRXB-A-33
PANACEA, ISCOR	Justify acceptability of basing assessment of pin power accuracy on code-to-code comparisons. Alternate approach and SLMCPR procedures proposed in response to SRXB-6	SRXB-A-34
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SLMCPR	Explanation provided regarding why axial TIP not included in SLMCPR	SRXB-A-45
Steady state and transient nuclear, Steady state and transient thermal hydraulic, fuel rod mechanical	Strategy for Application of Methods to design and addition SLMCPR margin to account for lack of experimental data	SRXB-A-06
TGBLA, MCNP	Information supplied on storage safety requirements -- related to limit on lattice reactivity	SRXB-A-11
TGBLA, MCNP	Explanation provided to justify acceptability of basing assessment of pin power accuracy on BOL conditions	SRXB-A-37
TGBLA, MCNP	Explanation provided for use of different uncertainties for GE14 and later designs. Refer to response to SRXB-6	SRXB-A-38
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