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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]

This LTR is consistent with and based on the approach used for the VYNPS extended power uprate review. It is intended to be referenced by near-term license applications for Extended Power Uprate, Constant Pressure Power Update, and the MELLLA+ operating domain expansion. A temporary additional SLMCPR margin of 0.02 is proposed, consistent with that accepted for VYNPS, without the provision for decreasing the additional margin for a specific plant application. The range of applicability includes any expanded operating range up to 120% of Original Licensed Thermal Power and including the MELLLA+ operating domain expansion. The approach in the enclosed LTR is to be implemented on a temporary basis pending the resolution of the NRC's RAIs regarding GE methods. GE is committed to the activities necessary to demonstrate the adequacy of GE's methods.

The treatment of the uncertainties in the safety limit development is discussed and the additional SLMCPR margin is supported. 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

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)
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]

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. The SE for the Applicability of GE Methods to Extended Operating Domains LTR would approve the use of GE's methods, including the use of a temporary additional SLMCPR margin of 0.02 as described in the Methods LTR, for expanded operating domains bounded by EPU or CPPU power uprates and MELLLA+ until final resolution of the NRC RAIs regarding GE's analytical methods [References 8 and 9].

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 1, 10, 11, and 12]. 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 [Reference13]. 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 increased uncertainties are established and the effect of the increase is evaluated. The adequacy of the existing margin, and, as applicable, augmented 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 and, if necessary, the resulting margin adjustments.

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 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.

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% Δk/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, and, as applicable, augmented margin for 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 14 and 15]. 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 which 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 17]. 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 18] 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
[[			
			]]

The potential effect of larger pin power uncertainty on the SLMCPR has been considered. First, in lieu of an arbitrary increase in the uncertainty, a review of [[

]] In the determination of SLMCPR, the use of additional pin power uncertainty so derived, i.e., [[

]], providing real additional critical power margin relative to GE's standard methodology and addressing local peaking uncertainty concerns.

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 19]. 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 [References 14, NEDC-32601P-A and 15, NEDC-32694P-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. [[

]]

BWRs have always operated at void fractions higher than 70% with some of the earlier gamma scan data from fuel exceeding 80% void fractions so that the effect of void fraction is included in confirmation of local and bundle power peaking uncertainty and, thus, not a significant concern. Instead, the largest differences in bundle power are the result of depletion and are not the result of differing product lines, composition, or core power. This key aspect is already addressed in the current NRC approved value [[

]] Therefore, the procedure of using the original gamma scan data to determine a conservative bound on the uncertainty is adequate and reasonable.

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

[[

]] This additional critical power margin provides adequate additional assurance of safety and is developed consistent with current NRC-approved bundle power uncertainty methodology.

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 13]. 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 14], 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 [[

]].

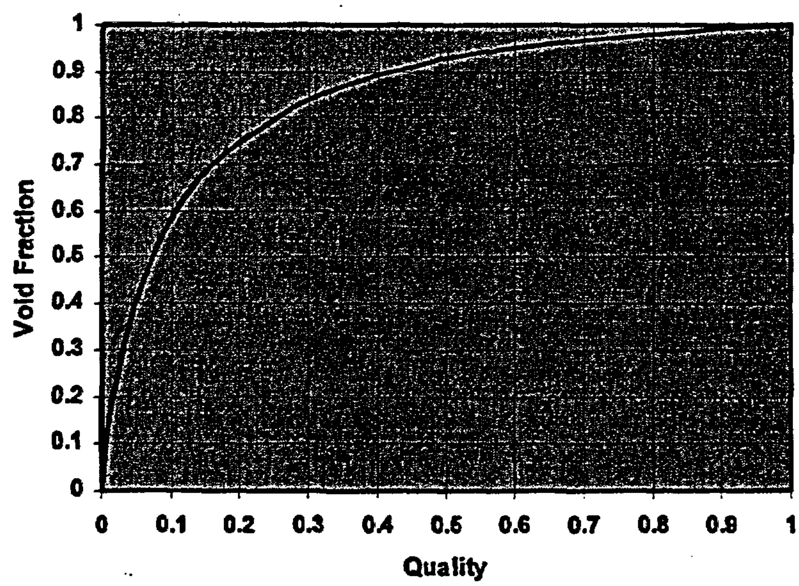
A void fraction of [[ ]] is relatively high and typical of the conditions where boiling transition will occur in a BWR fuel bundle. Also, since the OLMCPR is determined such that boiling transition will not occur, it is highly unlikely that a void fraction of [[ ]] will be exceeded (e.g., perhaps momentarily 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 $X = \frac{\dot{m}_g}{\dot{m}_f + \dot{m}_g}$.

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 16] 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 [See Appendix B] 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 and Alternate Approach

The use of alternative, more conservative, values for uncertainties in the local peaking factor [[ ]] results in an increase in the SLMCPR relative to that calculated with current GE standard methodology. [[ ]] 0.02 ΔCPR effect on SLMCPR based on the conservatively increased local peaking [[ ]] uncertainties. The approach for the SLMCPR evaluation applied to uprated operating conditions involves a two-step process. First, the SLMCPR is evaluated following the standard (cycle specific) process. Second, this evaluation is repeated with the increased uncertainties discussed in Section 2.2.1.2. The final SLMCPR is determined as the greater of the standard evaluation with an additional 0.02 ΔCPR (added to the safety limit), or the SLMCPR calculated with normal approved GE methods and the increased uncertainties. This approach accounts for any potential unique situations or designs and provides additional reasonable assurance of safety with respect to SLMCPR. No other uncertainties warrant an increase in the SLMCPR margins or considerations in the evaluation process.

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.

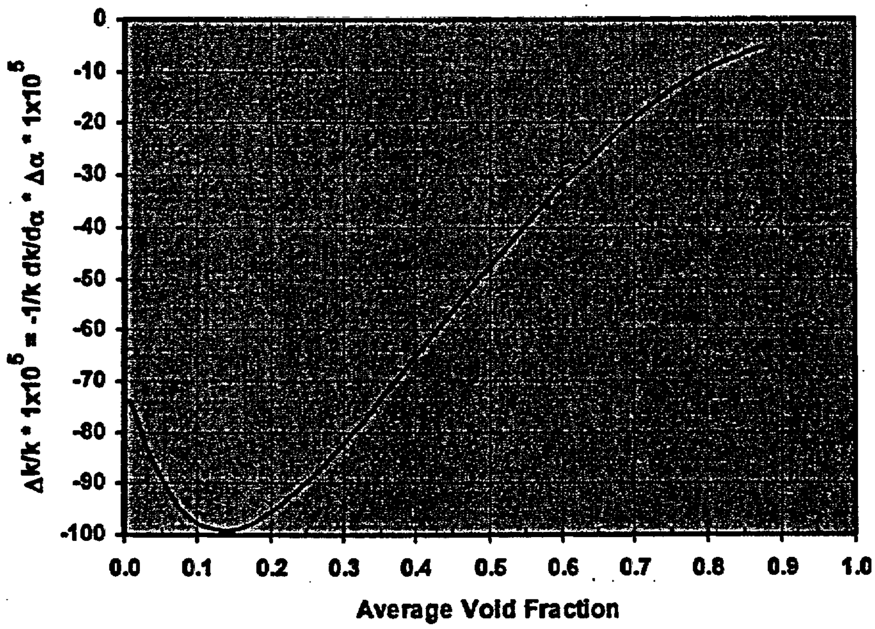
2.2.2.1 Fuel Parameters That Affect OLMCPR

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 $\Delta\alpha \approx \left(\partial\alpha/\partial X\right)\Delta X$ tends to be larger at low void fraction, so that the effective feedback $\Delta k/k \approx 1/k\left(\partial k/\partial\alpha\right)\Delta\alpha$ 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 ($\Delta X = 0.001$) using a representative void coefficient over a range of void fraction values.

Figure 2-3 Reactivity Change for a Small Quality Perturbation (ΔX = 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 20, 21, and 22] 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 [See Appendix B].

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 [See Appendix B]. 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 23] 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 [See Appendix B]) 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 [See Appendix B] 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 22]
- 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 17] 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 19] 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% Δk/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 [See Appendix B] 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.

[[

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]]

[[

]]

Table 2-11 Summary of Uncertainty Components for LHGR Evaluations

Component	NEDE-32601	Revised
[[		
		]]

[[

]].

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 24]. The analytical models used to perform ECCS-LOCA analyses are documented in Volume II of NEDE-23785-1-PA [Reference 25] together with NEDE-30996P-A [Reference 26] and NEDC-32950P [Reference 27].

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 28]. 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 29]. 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 30] 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). [[

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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 [See Appendix B].

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 28] 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 29]. 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 31], 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 29]. 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. If the current SLMCPR is increased by 0.02, the overall effect on the stability based OLMCPRs (note these values are determined at OPRM amplitude setpoints from 1.05 to 1.15 or 1.20) would be that they would increase by the ratio of the new SLMCPR to the old SLMCPR. But the acceptance criteria for selecting the appropriate OPRM setpoint, i.e., the transient OLMCPR, would also increase. Consequently, the OPRM setpoint would remain essentially unchanged if there were a change in SLMCPR and OLMCPR.

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 31], 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. [[

]]

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 32] 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 [See Appendix B].

2.7 LICENSED EXPOSURE

GE fuel designs are licensed to a [[ ]] exposure limit (i.e., 70 GWd/MTU for GE14). [Reference 33] 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 [See Appendix B].

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 1]. 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 34]. 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 35].

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% Δk/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 36] 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 5) is the only licensed (SER pending) stability solution for operation in the MELLLA+ domain [Reference 37]. 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

The evaluations presented in Sections 2 and 3 demonstrate that for CPPU, EPU, or MELLLA+ license amendment requests, an operational restriction in bundle critical power ratio via an increase in the SLMCPR of 0.02 ΔCPR is sufficient to provide additional and reasonable assurance of safety. No additional operational restrictions are required for CPPU or EPU applications and no other operational restrictions are required for MELLLA+ applications.

Safety Limit Critical Power Ratio (SLMCPR)

An adjustment to the SLMCPR of 0.02 ΔCPR is proposed to provide additional and reasonable assurance of safety for CPPU or EPU including MELLLA+ conditions. The standard, cycle specific evaluation (but with increased uncertainties) will be performed to assure that the adjustment is adequate. The adjustment to SLMCPR accounts for potential increases in the power distribution uncertainties, pending the acquisition of confirmatory gamma-scan data for 10x10 fuel designs. The adjustment will be removed and standard 1σ uncertainties applied considering updated data as it becomes available. This adjustment is also applicable to non-GE fuel designs in CPPU or EPU and MELLLA+ 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 % Δk/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

Consistent with the SLMCPR treatment of uncertainties, increases in the assumed pin and bundle power distribution uncertainties are applicable to the power distribution aspects of the thermal-mechanical calculations. However, 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 additional SLMCPR margin noted above and conservatisms in detect and suppress methodologies for the GE stability options justify that no additional margin is necessary. 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

As discussed regarding LHGR, increases in the assumed pin and bundle power distribution uncertainties are also applicable to the power distribution aspects included in the thermal-mechanical calculations. However, 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.

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.

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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. Appendix B includes a copy of the relevant VYNPS RAIs referenced by the Applicability of GE Methods to Extended Operating Domains LTR. 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

RAI Sort By Number		
Related Technology	Subject	RAI
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
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

RAI Sort By Number		
Related Technology	Subject	RAI
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
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

RAI Sort By Number		
Related Technology	Subject	RAI
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
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	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
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
Not Methods Related	Explained expression in TS 3.4.3. Information provided supporting the value of 1.29 at EPU conditions.	SRXB-A-62

RAI Sort by Related Technology		
Related Technology	Subject	RAI
Not Methods Related	The code used for over pressure protection analysis was identified	SRXB-A-07
Not Methods Related	NRC staff approved evaluation model identified for ATWS and discussion provided on EOPs	SRXB-A-22
Not Methods Related	Information supplied regarding PCT difference in VYNPS LBLOCA analysis	SRXB-A-10
Not Methods Related	NRC approved evaluation model LFW events identified.	SRXB-A-21
ODYN	Explanation of uncertainties in power during transients	SRXB-A-58
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
PANACEA, ISCOR	Refer to SRXB-A-19 for Representative Core definition	SRXB-A-09
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
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	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
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

RAI Sort by Related Technology		
Related Technology	Subject	RAI
PANACEA, ISCOR	Information provided for maximum bundle power and power density before and after EPU	SRXB-A-64
PANACEA, ISCOR	Shutdown margin verification and qualification Data and procedure provided	SRXB-A-67
PANACEA, ISCOR ODYN, SAFER	Explanation provided for inclusion of axial and nodal uncertainties in transient and accident evaluations	SRXB-A-29
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
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
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
TGBLA, MCNP	Discussion provided on Void reactivity coefficients for transients and accidents, including ATWS and SBO. Refer to SRXB-A-6	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
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TGBLA, MCNP	Clarification and more detail on response to SRXB-A-57	SRXB-A-71
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

RAI Sort by Related Technology		
Related Technology	Subject	RAI
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

APPENDIX B ACCEPTED VYNPS RAI RESPONSES

The following table presents the VYNPS reference letters and associated RAI responses. In Appendix B, the Cover Letter, Affidavit and the designated Attachments are included from each Supplement.

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 – (4 Non-P) SRXB-A-6 Attachment 5 – Affidavit
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 – (2 Non-P) Revised SRXB-A-6 Attachment 9 – (10 Non-P) SRXB-A-7 thru SRXB-A-58 Attachment 12 – Affidavit
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 – (6 Non-P) SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70 Attachment 9 – Affidavit
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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 Attachment 3 – Affidavit

Entergy

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Vermont Yankee
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March 10, 2005

Docket No. 50-271
BVY 05-024
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 24
Extended Power Uprate – Response to Request for Additional Information

References:

1) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," December 21, 2004

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, Technical Specification Proposed Change No. 263 – Supplement No. 23, Extended Power Uprate – Response to Request for Additional Information," BVY 05-017, February 24, 2005

This letter responds to NRC's request for additional information (RAI) of December 21, 2004 (Reference 1) regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 2) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

Reference 3 provided Entergy's response to 15 of the 18 individual RAIs included in Reference 1. This submittal provides responses to two of the remaining RAIs. Entergy is in the process of preparing a response to the last remaining RAI and anticipates submitting that response by March 16, 2005.

Subsequent to the receipt of the RAI, discussions were held with the NRC staff to further clarify the RAIs. In certain instances the RAIs may have been modified based on clarifications reached during the telecons. The information provided herein is consistent with those clarifications.

Attachment 1 is Entergy's response to RAI SPSB-C-35. Attachment 2 provides the "Exhibits" referenced in the response.

Attachment 3 is Entergy's response to RAI SRXB-A-6. Because the response to RAI SRXB-A-6 contains proprietary information as defined by 10CFR2.390, Attachment 3 has been designated in its entirety as proprietary information. A non-proprietary version of Attachment 3, suitable for public disclosure, is provided as Attachment 4 to this letter with the proprietary information redacted. An affidavit that constitutes a request for withholding of the proprietary information in Attachment 3 from public disclosure in accordance with NRC regulations is provided by the owner of the proprietary information (General Electric Company (GE)) as Attachment 5. The proprietary information in Attachment 3 is designated by double underline within double square brackets. In each case, the superscript notation, "$^{(3)}$", refers to paragraph (3) of the affidavit, which provides the basis for the proprietary determination. The proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information contained in the response was provided to Entergy in a GE transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in the enclosed response such that the affidavit remains applicable. GE requests that the enclosed proprietary information be withheld from public disclosure in accordance with the provisions of 10CFR2.390 and 10CFR9.17.

There are no new regulatory commitments contained in the responses to the RAIs.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 10, 2005.

Sincerely,

WFMaguire

William F. Maguire
General Manager, Plant Operations
Vermont Yankee Nuclear Power Station

Attachments (5)

cc: Mr. Richard B. Ennis, Project Manager
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Mr. David O'Brien, Commissioner
VT Department of Public Service
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Montpelier, Vermont 05620-2601

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 24

Extended Power Uprate

Response to RAI SRXB-A-6

REDACTED VERSION

Total number of pages in Attachment 4 (excluding this cover sheet) is 23.

Reactor Systems Branch (SRXB)

Boiling Water Reactors and Nuclear Performance Section (SRXB-A)

RAI SRXB-A-6

Table 1-1 in Attachment 6 of the application dated September 10, 2004 [*sic*], lists all the nuclear steam system codes used for the EPU request. Section 1.2.2 of Attachment 6, "Computer Codes," indicates that the VYNPS application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC safety evaluation report (SER) that approved each code, with exceptions as noted in Table 1-1.

Similarly, review the fuel vendor's analytical methods and code systems used to perform the safety analyses supporting the VYNPS EPU application and provide the following information:

(a) Confirm that the steady state and transient neutronic and thermal-hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges.

(b) Confirm that for the EPU conditions, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

(c) Confirm that the assessment database and the assessed uncertainty of models used in all licensing codes that interface with and/or are used to simulate the response of VYNPS during steady state, transient or accident conditions remain valid and applicable for the EPU conditions.

Response to RAI SRXB-A-6

GE's NRC-approved neutronic and thermal-hydraulic methods and code systems were submitted originally with a database of performance demonstrations that spanned the plants and operations of the BWR fleet. The breadth of demonstration established the applicability ranges for the coupled sets of methods. The review and approval of these methods was, in whole or in part, based on the performance demonstration given in those submittals. Periodic updates of methods performance provided to NRC represent further evidence that methods continue to meet performance expectations. The current applicability of methods to VYNPS EPU conditions is based [[

]]

Finally, the conclusion that the methods are applicable to EPU at VYNPS, based on the results provided in Parts 6(a) and 6(b), is documented in Part 6(c).

Response to Part 6(a):

NRC-approved or industry-accepted computer codes and calculational techniques have been applied for the power uprate analyses for VYNPS: TGBLA, PANACEA, ISCOR, ODYN, TASC, TRACG, STEMP, SAFER, ODYSY, LAMB, and GESTR. The application of each of these codes complies with the limitations, restrictions, and conditions specified in the approving NRC SER where applicable for each code. Moreover, GE routinely updates the staff on how NRC approved methods perform. A demonstration of continued acceptable performance of these codes for the expected phenomenological conditions for the VYNPS power uprate is pertinent here, too.

[[

]]

Safe BWR core operation is assured by operating within the Technical Specification limits for LHGR, MAPLHGR, MCPR, Hot Reactivity (Reactivity Anomaly), and Cold Reactivity (Shutdown Margin Demonstration). [[

]]

The numbered paragraphs below discuss [[

]]

These comparisons show that the [[

NON-PROPRIETARY INFORMATION

]]
are summarized in Table 6-2. Inspection of Tables 6-1, 6-3 and 6-4 and Figures 6-1 through 6-6, further demonstrates the fact that VYNPS EPU operation is consistent [[

NON-PROPRIETARY INFORMATION

]]

Applicability to transient methods

The transient, accident and stability analyses are classified into the three broad areas.

d. Transient events
e. LOCA
f. Stability

The transient area in this case involves the events affecting the core. [[ ]]

The reactivity events are analyzed with the steady state tools and the results presented regarding steady-state methods in this response are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events.

The pressurization events result in higher pressures and a momentary increase in core flow for VYNPS, which reduces the void fractions and increases the power generation. The Peach Bottom tests are used as a basis for the transient model validation for the limiting pressurization events. The model bias and uncertainties have been defined for these events and are applied for the transient analysis. The core conditions are bounded by the GEXL and void-quality correlations for these events. Further, the GEXL correlation has been qualified through full scale thermal-hydraulic testing for transient conditions by simulation of the limiting pressurization events. These response conditions keep the core within the bounding range of the other plants indicated in the examples for their transient conditions.

The depressurization and flow reduction events are not the limiting events and result in a reduction of power, which is the reason they are not limiting. The reduction in power results in a decrease in void fraction. When flow decreases occur such as a pump trip, there is still a reduction in power and no challenge to the thermal limits due to the reduction in power. [[ ]]

LOCA events start from the outset with a sharp drop in reactor power (reactor trip occurs almost immediately), core flow and pressure. The post-LOCA thermal hydraulic conditions are within the qualification basis of the SAFER-GESTR code models. These models have been qualified against data obtained in numerous small and full-scale experiments and tests. The GEXL and void-quality correlation are applied in the TASC code to determine dryout times. The post-LOCA thermal hydraulic conditions are within the qualification basis for these models. There is no significant dependence upon the steady-state correlations of the void-quality correlation for the limiting calculations. The GEXL and void-quality correlation applications are within the range of application.

The pump trip transients lead to flow reductions that impact stability. [[

]]. The development of the void-quality correlation (NEDE-21565, J.A. Findlay & G.E. Dix, "BWR Void Fraction and Data", 1977) established a maximum range of [[ ]] voids. [[

]]

Response to Part 6(b):

[[

]]

BWRs are designed so that they can be shut down in the cold condition (68°F) with the single strongest control blade completely withdrawn. In order to qualify the 3D simulator to accurately predict the cold shutdown margin, cold critical startup configurations are analyzed. In all cases, enough control blades were withdrawn at a given water temperature for the reactor to be critical or on a large positive period. [[

]] The RMS difference between predicted nuclear design basis based on past experience and the actual measured [[

NON-PROPRIETARY INFORMATION

]]

Response to Part 6(c):

The response to Part 6(a) has shown that the expected performance of VYNPS with EPU does not exceed [[ ]] Thus, VYNPS with EPU is not expected to operate with any of these [[ ]] exceeding the values [[ ]]

The results presented in Part 6(b) demonstrate that for several cycles of the [[ ]] the GE methods provide the same level of fidelity [[ ]] The uncertainties in the methods predictions are consistent with those previously developed and reported in the approved licensing topical reports for these methods.

Therefore, since the methods continue to be valid [[ ]] it is concluded that the models themselves continue to remain "valid and applicable" for EPU at VYNPS.

Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

August 1, 2005

Docket No. 50-271
BVY 05-072
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 30
Extended Power Uprate – Response to Request for Additional Information

References:

1) 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

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, Supplement No. 24 – Response to Request for Additional Information," BVY 05-024, March 10, 2005

3) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The major aspects of this submittal are:

1) An update to Entergy's response to request for additional information (RAI) item SRXB-A-6 regarding certain analytical methodologies of General Electric (GE) that are used for the design and evaluation of VYNPS' fuel. The prior response to SRXB-A-6 was provided with Entergy's letter of March 10, 2005 (Reference 2) and is being superseded by this submittal.

2) An executive overview summarizing Entergy's understanding of the key issues remaining to provide reasonable assurance of steam dryer integrity at EPU conditions and also summarizing the framework for Entergy's response to those issues.

3) Responses to a significant number of those RAIs requested by NRC letter of July 27, 2005 (Reference 3). The remaining RAIs that pertain to the steam dryer and piping/nozzle stress evaluations are not included, but will be transmitted as a separate submittal by August 4, 2005.

GE Analytical Methods

In its letter of March 10, 2005, Entergy had proposed in its response to RAI SRXB-A-6 a means of addressing the NRC staff's questions regarding GE methods. The response was consistent with the Methods Interim Process proposed by GE in its letter of March 25, 2005 (MFN 05-005). Although Entergy remains confident that the concepts originally advanced in the response to RAI SRXB-A-6 are valid, an alternate, VYNPS-specific approach is provided by this letter. Entergy is revising and superseding the prior response to SRXB-A-6 with this submittal.

The alternate approach, discussed in the revised response to RAI SRXB-A-6 (Attachment 1), considers those core operating parameters and associated limits that could be impacted if all the uncertainties in methodology postulated by the staff were present during EPU operation, and then evaluating what, if any, operating restrictions should be imposed to compensate for this theoretical condition by providing additional safety margins to the affected limits. Using this approach Entergy has determined that a change of 0.02 to the safety limit minimum critical power ratio (SLMCPR) provides sufficient additional conservatism and adequate margin to address the postulated uncertainties in GE's methodology. Entergy is therefore proposing a license condition for EPU operation that imposes this additional 0.02 SLMCPR restriction until such time that the generic issues associated with GE analytical methods are adequately resolved with respect to VYNPS.

The alternate approach also describes Entergy's basis for confirming the adequacy of existing margin to accommodate the postulated uncertainties and assessing their impact on each of the remaining affected core operating parameters and associated limits. In addition, actual VYNPS operational experience with regard to core thermal limits is provided in the revised response to RAI SRXB-A-6.

Steam Dryer Analyses

Attachment 3 provides an overview of Entergy's understanding of the fundamental issues left to be resolved in order to provide reasonable assurance that steam dryer integrity will be maintained at EPU conditions. These issues are drawn from 129 individual questions posed by the NRC staff. Attachment 3 provides a restatement of

Entergy's overall approach to the steam dryer integrity issue and the framework of Entergy's strategy in addressing the remaining fundamental issues so that the answers to individual questions can be reviewed in that context. Attachment 5 provides responses to questions associated with computational fluid dynamics and steam dryer loads at EPU conditions. The remainder of the steam dryer-related RAIs are in review and are expected to be submitted by August 4, 2005.

Response to Requests for Additional Information

Attachments 4, 5, 7, 8, and 9 respond to individual RAIs, according to NRC review branch. Of the 200 individual RAIs requested by the NRC in Reference 3, 107 which pertain primarily to uncertainties in the acoustic circuit model, Scale Model Test benchmark adequacy, and applicability of the insights gained from the Quad Cities 2 instrumented dryer tests will be addressed in a future submittal, expected to be provided by August 4, 2005.

The revised response to RAI SRXB-A-6, as well as other responses to Reactor Systems Branch RAIs, (Attachments 1 and 9) contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with provisions of that regulation. Attachments 1 and 9 are considered to be Proprietary Information in their entirety. Attachments 2 and 10 are non-proprietary versions of Attachments 1 and 9, respectively. Affidavits supporting the proprietary nature of the documents are provided as Attachment 6 (for Attachment 1), and as Attachment 12 (two affidavits for Attachment 9). "Exhibits," which provide supporting information to certain RAI responses are included in Attachment 11.

This submittal provides a substantial portion of the information needed to support the preparation of the NRC's safety evaluation report for EPU and is therefore being submitted in advance of the responses to the remaining questions. In compiling and analyzing the information for this submittal, Entergy remains convinced that the VYNPS can be safely operated at up to 120% CLTP. It is our understanding that an audit of the underlying details supporting elements of this submittal will be conducted on or about August 22, 2005. Entergy anticipates that the nature of the audit will be confirmatory and respectfully requests that additional requests for information, if any, be communicated as soon as practical.

The following attachments are included in this submittal:

Attachment	Title
1	Revised Response to RAI SRXB-A-6 (proprietary version)
2	Revised Response to RAI SRXB-A-6 (non-proprietary version)
3	Overview of Steam Dryer Issues
4	Responses to RAIs EEIB-A-1 through EEIB-A-5 (no proprietary information)
5	Responses to RAIs EMEB-B-18 through EMEB-B-149, non-inclusive (non-proprietary version)
6	Affidavit for Attachment 1

7	Responses to RAIs SPSB-C-47 through SPSB-C-52 (no proprietary information)
8	Responses to RAIs SPLB-A-25 through SPLB-A-29 (no proprietary information)
9	Responses to RAIs SRXB-A-7 through SRXB-A-58 (proprietary version)
10	Responses to RAIs SRXB-A-7 through SRXB-A-58 (non-proprietary version)
11	RAI Response Exhibits (10)
12	Two affidavits for Attachment 9
13	New Regulatory Commitments (2)

There are two new regulatory commitments contained in this submittal that are incorporated into the responses to RAIs EEIB-B-1 and EEIB-B-5 regarding actions associated with the postulated station blackout event. They are summarized in Attachment 13.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings (or audits of design files) at your earliest convenience.

If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 2, 2005.

Sincerely,

Robert J. Wanczyk

Robert J. Wanczyk
Director, Nuclear Safety Assurance
Vermont Yankee Nuclear Power Station

Attachments (13)

cc: (see next page)

cc: Mr. Richard B. Ennis, Project Manager
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Mr. Samuel J. Collins (w/o attachments)
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USNRC Resident Inspector (w/o attachments)
Entergy Nuclear Vermont Yankee, LLC
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Mr. David O'Brien, Commissioner (w/o proprietary information)
VT Department of Public Service
112 State Street – Drawer 20
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BVY 05-072
Docket No. 50-271

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 30

Extended Power Uprate

Response to Request for Additional Information

Revised Response to RAI SRXB-A-6

NON-PROPRIETARY VERSION

Total number of pages in Attachment 2
(excluding this cover sheet) is 23.

REVISED RESPONSE TO RAI SRXB-A-6

PREFACE

This attachment provides a revised response to the NRC staff's request for additional information (RAI) dated December 21, 2004,[1] regarding RAI SRXB-A-6. The response provided below supersedes the response provided by Entergy in its letter dated March 10, 2005.[2]

In its letter of March 10, 2005, Entergy had proposed in its response to RAI SRXB-A-6 a means of addressing the NRC staff's questions regarding GE methods. The response was consistent with the Methods Interim Process proposed by GE in its letter of March 25, 2005 to the NRC staff[3]. Although Entergy remains confident that the concepts originally advanced in the response to RAI SRXB-A-6 are valid, an alternate, VYNPS-specific approach appears to offer the most efficient path to resolving the NRC staff's concerns. Therefore, Entergy is revising and superseding the prior response to SRXB-A-6 with this submittal.

The alternate approach, discussed in the revised response to RAI SRXB-A-6 below, considers those core operating parameters and associated limits that could be impacted if the uncertainties in methodology postulated by the NRC staff were present during EPU operation, and then evaluating what, if any, operating restrictions should be imposed to compensate for this theoretical condition by providing additional safety margins to the affected limits. Using this approach Entergy has determined that a change of 0.02 to the safety limit minimum critical power ratio (SLMCPR) provides sufficient additional conservatism and adequate margin to address the postulated uncertainties in GE's methodology. Entergy is therefore proposing a license condition for EPU operation that imposes this additional 0.02 SLMCPR restriction until such time that the generic issues associated with GE's analytical methods are adequately resolved.

The alternate approach also describes Entergy's basis for confirming the adequacy of existing margin to accommodate the postulated uncertainties and assessing their impact on each of the remaining affected core operating parameters and associated limits. In addition, actual VYNPS operational experience with regard to core thermal limits is provided in the revised response to RAI SRXB-A-6.

---

1 U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," December 21, 2004

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, Supplement No. 24 – Response to Request for Additional Information," BVY 05-024, March 10, 2005

3 GE Nuclear Energy (George B. Stramback) letter to U.S. Nuclear Regulatory Commission (Herbert Berkow), "Methods Interim Process (TAC No. MC5780)," MFN 05-005, March 25, 2005

Reactor Systems Branch (SRXB)
Boiling Water Reactors and Nuclear Performance Section (SRXB-A)
(The RAI is stated below as provided in NRC's letter of December 21, 2004.)

RAI SRXB-A-6

Table 1-1 in Attachment 6 of the application dated September 10, 2004 [*sic*], lists all the nuclear steam system codes used for the EPU request. Section 1.2.2 of Attachment 6, "Computer Codes," indicates that the VYNPS application of these codes complies with the limitations, restrictions, and conditions specified in the applicable NRC safety evaluation report (SER) that approved each code, with exceptions as noted in Table 1-1.

Similarly, review the fuel vendor's analytical methods and code systems used to perform the safety analyses supporting the VYNPS EPU application and provide the following information:

(a) Confirm that the steady state and transient neutronic and thermal-hydraulic analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges.

(b) Confirm that for the EPU conditions, the calculational and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions.

(c) Confirm that the assessment database and the assessed uncertainty of models used in all licensing codes that interface with and/or are used to simulate the response of VYNPS during steady state, transient or accident conditions remain valid and applicable for the EPU conditions.

Revised Response to RAI SRXB-A-6

Margin in GE Analytical Methods Supporting VYNPS EPU Submittal

Summary

As part of Entergy's submittal of a license amendment request for the VYNPS extended power uprate (EPU), Entergy is proposing an alternate approach to address NRC questions related to GE's standard methodologies to facilitate NRC approval of the request. The alternate approach includes the addition of an operational restriction on bundle critical power ratio to be implemented via an adjustment of 0.02 ΔCPR to the safety limit minimum critical power ratio (SLMCPR). The following discussion provides the bases for the adequacy and additional conservatism, with respect to the aforementioned NRC questions, of the margins in pertinent safety parameters for the VYNPS EPU provided by GE's standard methodologies as further augmented by the proposed Entergy alternative operational restriction. It is intended that the operational restriction in the alternate approach would only be implemented as a condition of the

EPU License Amendment until the aforementioned NRC questions are otherwise satisfactorily resolved.

Introduction

In its review of GE's generic MELLLA+ submittal[4], the NRC has asked questions related to the adequacy, given the absence of recent gamma-scan test data, of the standard uncertainties and biases utilized in GE's bundle lattice and core simulation methodologies (see "GE Bundle Lattice and Core Simulation Methodology & Utilization of Gamma Scan and Fuel Isotopic Data" section below) for current fuel designs and operating strategies and the potential effect on safety parameters influenced by such uncertainties and biases. As noted in the "GE Bundle Lattice and Core Simulation Methodology & Utilization of Gamma Scan and Fuel Isotopic Data" section, GE has benchmarked its methods using industry standard techniques and utilized gamma scan data to retrospectively confirm the adequacy of certain elements of its methods and benchmarking. GE has provided, and continues to provide, information to the NRC supporting the adequacy of GE's methodologies for application to BWRs and BWR expanded operating domains.

The following discussion addresses the NRC questions regarding both gamma-scan and isotopic data and supports the alternate approach of addressing the NRC's questions regarding uncertainties and biases which is being proposed by Entergy as an element of the VYNPS extended power uprate (EPU) license amendment request. The alternate approach includes a proposed operational restriction in bundle critical power ratio implemented via an increase in the SLMCPR of 0.02 ΔCPR. This operational restriction provides additional margin and addresses the aforementioned NRC questions. The following discussion identifies the fuel parameters related to the gamma scan and isotopic data and addresses the effect of uncertainties in those fuel parameters to the extent each is applicable to the six pertinent safety parameters which are influenced by those fuel parameters. For each of the six pertinent safety parameters:

1. the fuel parameters which affect it are identified,
2. the treatment of fuel parameter uncertainties in the safety parameter limit development is considered, and
3. the adequacy of the existing treatment in conjunction with the alternate approach is supported.

Safety Parameters Influenced by Noted Uncertainties and Biases

GE has reviewed its methodologies to determine the uncertainties and biases which were confirmed by earlier gamma scan test data or measurements of irradiated fuel isotopics and to confirm that the existing types of uncertainties already included in GE's NRC-approved treatment of uncertainties and biases address the NRC staff questions regarding the absence of

[4] MELLLA+ is not part of Entergy's VYNPS EPU application; however, NRC requests for information on the related subject are contained in that docket and are provided here as reference.

recent confirmatory test data.

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 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., traversing in-core probes (TIP) data and shutdown margin demonstrations.

The safety parameters potentially influenced by the 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 would lead to potential non-conservatisms in resulting predictions. 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 reasonable assurance of safety.

A discussion of the adequacy of the margin existing in, and, as applicable, augmented margin for, each of these safety parameters is provided below, again based on the alternate approach being proposed by Entergy for the VYNPS EPU submittal.

SLMCPR Margin

The safety limit minimum critical power ratio (SLMCPR) is directly affected by the fuel parameters confirmed by gamma scan data. The local pin power peaking (axial and in-bundle) and [[ ]] uncertainties are factors which affect SLMCPR. SLMCPR is not affected by void reactivity coefficient uncertainties. The other safety parameters, discussed in succeeding sections, already incorporate other conservatisms which encompass the pin power/exposure, [[ ]] and void reactivity coefficient uncertainties or are not influenced by those 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 protected from boiling transition when such uncertainties are incorporated. Uncertainties in local pin power peaking and [[ ]] (and bundle power) are explicitly included in the SLMCPR determination and considered separately, then cumulatively below.

The potential effect of larger pin power uncertainty on the SLMCPR has been considered. First, in lieu of an arbitrary increase in the uncertainty, a review of [[

]] In the determination of SLMCPR, the use of additional pin power uncertainty so derived, i.e., [[

]], providing real additional critical power margin relative to GE's standard methodology and addressing local peaking uncertainty concerns.

[[ ]] is a component of the total bundle power uncertainty. The total bundle power uncertainty for application within GE's NRC-approved SLMCPR determination process consists of the component uncertainties in *Table SRXB-A-6-1* (From Table 4.2, page 4-2 in NEDC-32694P-A).

Table SRXB-A-6-1

Quantity	Uncertainty	Source
[[		
		]]

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. In 2005, GE formally provided a large amount of data for uprated plants loaded primarily with 10x10 fuel in methods related RAI responses under the MELLLA+ docket (MFN 05-029, TAC No. MC5780). Examination of these data confirms the applicability and conservatism of the original [[ ]] uncertainty documented in GE's NRC-approved topical report describing the SLMCPR methodology power distribution uncertainties (NEDC-32601P-A/NEDC-32694P-A).

[[

]]

BWRs have always operated at void fractions higher than 70% with some of the earlier gamma scan data from fuel exceeding 80% void fractions so that the effect of void fraction is included in confirmation of local and bundle power peaking uncertainty and, thus, not a significant concern. Instead, the largest differences in bundle power are the result of depletion and are not the result of differing product lines, composition, or core power. This key aspect is already addressed in the current NRC approved value [[

]] Therefore, the procedure of using the current gamma scan data to determine a conservative bound on the uncertainty is reasonable and valid.

[[

]] This additional critical power margin provides a real additional assurance of safety and is developed consistent with current NRC-approved bundle power uncertainty methodology.

The effects of [[ ]]in *Table SRXB-A-6-1* on the bundle power uncertainty for SLMCPR determination [[ ]]

[[ ]]0.02 ΔCPR effect on SLMCPR based on the conservatively increased local peaking [[ ]] uncertainties. [[ ]]is further conservative.

In summary, use of alternative, even more conservative values for uncertainties in the local peaking factor [[ ]]results in an increase in the SLMCPR for VYNPS of 0.02 relative to that calculated with current GE standard methodology and provides additional reasonable assurance of safety for VYNPS EPU with respect to SLMCPR.

OLMCPR Margin

The analysis of anticipated operational occurrences (AOOs) examines the change in critical power ratio relative to the original starting point conditions and determines the most limiting transient event. The fuel parameters identified previously, i.e., the local pin power peaking, void reactivity coefficient, and [[ ]], are factors in the evaluation of limiting AOOs. [[

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

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.

Both the ODYN and TRACG transient methodologies 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 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 are conservatively analyzed by ODYN and typically establish the limiting ΔCPRs, conservatisms existing in the process of determining OLMCPRs address NRC questions related to gamma scans and fuel isotopics as they relate to OLMCPR.

In summary, the standard GE methodologies utilized to establish the OLMCPR conservatively address uncertainties issues and provide reasonable assurance of safety for VYNPS EPU with respect to OLMCPR.

Shutdown Margin (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 indirect 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. As described in the "GE Bundle Lattice and Core Simulation Methodology & Utilization of Gamma Scan and Fuel Isotopic Data" section, 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 verification because any error in bundle or nodal power (or exposure) would tend to degrade the ability of the core simulator to establish a stable bias (in eigenvalue), a measure of the ability of the model to reliably predict core hot and cold critical conditions. While the Technical Specification for SDM is 0.38% $\Delta k/k$ 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. For VYNPS, the standard design SDM is 1.1% $\Delta k/k$ to provide additional flexibility in cycle length and operations. The uncertainty in cold critical predictive capability is considered and included in this choice of SDM requirement.

However, it is very important to note that actual SDM is a demonstrated quantity (plant verification) during plant startups or by use of local criticality confirmations. In addition, trending of hot eigenvalue (i.e., reactivity anomalies), also required by Technical Specifications and another direct confirmation of the adequacy of GE's methods with respect to fuel depletion and reactivity predictions, is performed. Because such plant verification data from power uprated plants and plants with modern fuel designs, including GE14, have continued to confirm that adequate SDM exists and that eigenvalue biases in GE's methods are stable and well understood, there is sufficient justification for the adequacy of GE's bundle lattice and core simulation methodologies and the uncertainties in the nodal and bundle power and exposure even without recent confirmatory gamma scan or fuel isotopic data.

In summary, the current design process and design goal, in combination with the existing processes of plant verification of SDM and trending of hot eigenvalues, provide reasonable assurance of adequate SDM.

LHGR Margin

For each GNF fuel design, including GE14, thermal-mechanical based linear heat generation rate (LHGR) 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 limits, 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. 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 limits. The fuel parameters ultimately determine the local power, which is explicitly addressed by the LHGR limit.

Fuel rod thermal-mechanical licensing criteria explicitly considered in the specification of LHGR limits include fuel centerline temperature, cladding plastic strain, and fuel rod internal pressure. Each of these criteria is limiting over a portion of the fuel rod lifetime. For development of the final limit curve, the peak power node is conservatively assumed [[ ]]. In addition, model and operating uncertainties are explicitly addressed in the development of limits, including an additional power uncertainty of [[ ]] power that is not specifically assigned to any cause, as well as a [[ ]] conservative power bias in the fuel rod internal pressure calculation. The uncertainty and bias also apply to exposure because, in the determination of LHGR limits, the exposure is the integrated power.

Moreover, the model uncertainties in GE's NRC-approved thermal-mechanical analysis methodology (GSTRM) are based upon temperature benchmark data and are also validated via fission gas benchmark data for which the nominal power history is produced in the steady-state core simulator. Because the large uncertainties included by this process encompass the uncertainties in local and rod power reflected in the NRC staff questions and because separate experimental benchmarking information confirms that the model uncertainties remain valid, an adjustment to provide additional LHGR margin is unnecessary.

In summary, 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.

MAPLHGR Margin

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 emergency core cooling system (ECCS). 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.

The ECCS-LOCA analysis applicable to the VYNPS EPU follows the NRC-approved SAFER/GESTR application methodology documented in NEDE-23785-1-PA Rev. 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-Of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984. The analytical models used to perform ECCS-LOCA analyses are also documented in NEDE-23785-1-PA together with NEDE-30996P-A, "SAFER Model for Evaluation of Loss-of-Coolant Accidents for Jet Pump and Non-jet Pump Plants, Volume I, SAFER – Long Term Inventory Model for BWR Loss-of-Coolant Analysis," October 1987, and NEDC-32950P, "Compilation of Improvements to GENE's SAFER

ECCS-LOCA Evaluation Model," January 2000.

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 (thus set by the AOO analysis).

Pin power peaking for the hot rod is set to a [[ ]] to further insure 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 VYNPS Licensing Basis peak clad temperature (PCT) determined by the methodology described above is 1960°F. This result includes application of Appendix K modeling assumptions. The maximum nominal PCT is about [[ ]] lower than the Appendix K value. When the nominal PCT is adjusted to account for model uncertainties (at 95% probability), the PCT (also known as the Upper Bound PCT in the SAFER/GESTR methodology) is about [[ ]] lower than the Licensing Basis PCT. The 95% probability PCT includes an uncertainty of [[ ]] on the LHGR ([[ ]]).

The SAFER/GESTR methodology assumes a bounding post-LOCA core power trajectory 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.

In summary, 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 reasonable assurance that those limits provide adequate margin to protect the fuel.

Stability

BWR thermal-hydraulic stability analyses are performed to assure that SLMCPR is protected in the event of a thermal-hydraulic instability event. The fuel parameters in question affect stability performance.

Background: VYNPS has implemented the Option I-D solution documented in "*Application of the 'Regional Exclusion with Flow-Biased APRM Neutron Flux Scram' Stability Solution (Option I-D) to Vermont Yankee", Licensing Topical Report,* GENE-637-018-0793, July 1993. Option I-D has (1) "prevention" elements, these being the Exclusion and Buffer Regions, and (2) a "detect & suppress" element, this being Safety Limit MCPR (SLMCPR) protection provided by the flow-biased APRM flux scram trip (for the dominant core wide mode of coupled thermal-hydraulic/neutronic reactor instability).

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 met for thermal hydraulic oscillations.

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, July 2001) 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. [[ ]] 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 for VYNPS 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 regional mode of instability is not likely may 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 VYNPS.

Detect and Suppress calculation: The detect and suppress evaluation for the VYNPS EPU is performed under the approved LTR basis (NEDO-32465-A, General Electric Company, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.). 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 new BWROG Guideline (GE-NE-0000-0031-6498-R0, "Plant-Specific Core-Wide Mode DIVOM Procedure Guideline," June 2005), 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.

In summary, the uncertainties in power distribution calculation and void reactivity do not significantly impact the safety margin in the stability analysis for VYNPS.

Margin to Licensed Fuel Exposure

GE fuel designs are licensed to a peak pellet exposure (i.e., 70 GWd/MTU for GE14). This is equivalent to a GE14 rod average exposure of ~61.4 GWd/MTU, but there is not an explicit rod average exposure limit for GE14 or other GE fuel designs. This limit is used to assure that fuel is not operated beyond its analysis basis. 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 limits, and, thus, the fuel exposure limit. The fuel parameters ultimately determine the local power, which is explicitly addressed by the LGHR limit.

Fuel rod internal pressure is the limiting licensing criterion at end-of-life for GE fuel designs. The fuel cladding creep rate is a function of cladding temperature and in turn of LHGR. As discussed previously, the LHGR limits for GE14 are deliberately conservative with respect to local rod power, assume a conservative pellet swelling rate uncertainty, and are also specified such that the margin to the criterion for limiting pellet-cladding gap increase due to rod internal pressure is actually smaller several GWd/MTU before end-of-life than at the peak pellet exposure (end-of-life) limit. Thus, existing uncertainties and margins in GE's NRC-approved fuel thermal-mechanical methodology adequately address the NRC question regarding local peaking uncertainty with respect to the licensed fuel exposure limit.

In summary, the GE standard fuel thermal-mechanical analysis basis considers and provides adequate margin for uncertainties in local and bundle power.

Additional Margin Summary

If it is desirable to address NRC questions regarding the adequacy of GE's standard uncertainties in local power/exposure, [[ ]], or void reactivity coefficient for EPU conditions in the absence of recent confirmatory gamma scan and fuel isotopic data via an alternate approach of incorporating additional margin in appropriate safety parameters, the evaluation above provides the basis for a determination that an operational restriction implemented via an adjustment to the SLMCPR of 0.02 ΔCPR provides additional and reasonable assurance of safety for VYNPS at EPU conditions. Significant conservatisms already exist in the processes for determination of the other safety parameters, i.e., OLMCPR margin, SDM, LHGR, MAPLHGR, thermal-hydraulic stability protections, and fuel (peak pellet) exposure, to address the NRC staff questions, and adjustments or operational restrictions for these are, thus, not required.

GE Bundle Lattice and Core Simulation Methodology & Utilization of Gamma Scan and Fuel Isotopic Data

Summary

GE's bundle lattice and core simulation codes, TGBLA and PANACEA, are best-estimate methods with uncertainties and biases in inputs and outputs of those codes addressed by the conservative treatment, previously approved by the NRC, of uncertainties and biases propagation in GE's calculations of conservative limits for various fuel safety parameters. The bundle lattice methods have been benchmarked, using industry standard practice, against Monte Carlo calculations for all GE fuel types. These benchmarks have been further confirmed for certain GE fuel types, retrospectively, with gamma-scan data available to GE. The core simulator methodology has been benchmarked, again using industry standard practice, against the operating plant instrumentation, e.g., traversing in-core probes (TIPs). [[

]] Operating plant data are continuously utilized to evaluate the accuracy of predictions of the bundle lattice and core simulator methodologies on both a plant-specific and BWR fleet-wide basis, and such trending is periodically (approximately annually) reviewed with the NRC staff in fuel technology update meetings.

In accordance with its understanding of previous NRC-approved licensing topical reports and NRC-issued safety evaluations for GE's methods, GE has evaluated and reflected the accuracy of its methodologies as it has introduced new fuel designs and operating strategies. GE believes that its bundle lattice and core simulator methodologies, including the associated uncertainties and biases utilized by GE, in combination with its NRC-approved treatment of uncertainties and biases, are adequately predicting the performance and assuring the safety of BWRs at up to and including 120% EPU conditions.

Qualification Process

GE utilized rod gamma scans, i.e., measurements of gamma emissions from certain fission product isotopes in individual irradiated BWR fuel rods, to further confirm the ability of its benchmarked methods to adequately predict local (fuel pin) power and exposure (i.e., burnup or depletion). GE utilized bundle gamma scans, i.e., scans of entire BWR fuel bundles, to confirm an appropriate value for uncertainty related to the [[ ]]. GE utilized irradiated fuel rod isotopic measurements, i.e., radiochemistry determination of inventory of certain fission and activation products, which are necessarily limited in number due to the difficulty in obtaining such measurements, in lattice physics code development but not as part of code benchmarking.

GE evaluates methods on multiple geometrical bases. The process of monitoring operational core parameters provides an up-to-date (hourly) evaluation of steady-state core reactivity control and provides a way to evaluate the core simulator eigenvalue bias. Comparison of calculated to measured TIP signals provides confirmation of the three-dimensional field of flux/power on a very timely basis (monthly) but with a resolution scale that only reflects the coarse mesh resolution of the three-dimensional simulator. Natural noise in the TIP instrumentation conservatively results in a fundamental contribution of 1% to the evaluated comparison (NEDC-32694P-A, page A7). Bundle-wise or pin-wise gamma scans allow for a better resolution in space but result in a poor temporal comparison because the present concentration of the typically measured fission product (^{140}Ba) requires an integration of the power history for the prior sixty days. Moreover, because of limitations, gamma scans may only be achievable once per cycle for operating power reactors. Bundle gamma scans usually entrain an experimental uncertainty of 1% (1σ) in the measured values while rod gamma scans entrain an uncertainty of 2% (1σ).

Because the injection of experimental error of non-routine benchmarking may confound physical phenomena of interest and for purposes of more timely and comprehensive evaluation, it is meaningful to compare production lattice physics methods (TGBLA) to Monte Carlo methods whose efficacy has been established through comparison to critical benchmarks. Assuming adequate trials have been considered, the local accuracy provides significant insight for examination of relative local pin peaking accuracy. If the local power is being produced correctly, the subsequent depletion of the fuel is occurring at the correct rate and location. Furthermore, assuming the nominal production lattice physics code produces stable core eigenvalue behavior (evaluated in the operational core follow examination), use of the depleted isotopic compositions from the deterministic code for comparisons to Monte Carlo later in the life of the fuel is both meaningful and produces further insight into modeling accuracy. The conclusion is that it is meaningful and proper to consider comparisons between TGBLA and Monte Carlo methods in evaluation of methods accuracy.

In summary, the GE standard fuel lattice and core simulator methodology qualification process utilizes a large volume of contemporaneous operating plant data supported by available confirmatory, retrospective gamma scan to assure high-quality best-estimate predictions of local, nodal, bundle, and core power. As discussed above, GE's NRC-approved treatment of the uncertainties in the power predictions assure conservative limits for the safety parameters influenced by the local, nodal, bundle, and core power.

VYNPS Core Follow / Required Technical Specification Surveillance Information

In addition to the above arguments related to GE analytical methods, VYNPS and Global Nuclear Fuels personnel perform core follow and required Technical Specification surveillance activities in the effort to ensure the VYNPS core is operating as expected.

CASMO-4/SIMULATE-3 Overview

The CASMO-4 and SIMULATE-3 codes are part of the Studsvik Scandpower Core Master

System. CASMO-4 is a multigroup transport theory code which feeds cross sections and discontinuity factors into SIMULATE-3. SIMULATE-3 is an advanced two-group nodal diffusion code with the ability to perform pin-power reconstruction. The code package is primarily used to independently verify vendor calculations and confirm the core is behaving as predicted during the cycle. Entergy receives the most up-to-date versions of the codes when available.

Entergy, through the core follow procedure, uses the CASMO-4/SIMULATE-3 package to trend the online performance of key core parameters. The key parameters, indicated in the core follow procedure, include MFLCPR, MFLPD, MAPRAT and gamma TIPS.

Thermal Limits

3D-MONICORE™ (3DM) is the plant adaptive online software. At the heart of 3DM is the PANACEA-11 (P11) software engine that runs in shape adaptive mode to calculate core state points. Offline non-adaptive P11 is used by Global Nuclear Fuels (GNF) engineers to show the agreement between the offline and online comparison during core follow. Offline P11 and SIMULATE-3 are used to model online data provided by 3DM to ensure future predictions with the codes are correct.

MFLCPR (Maximum Fraction Limiting Critical Power Ratio)

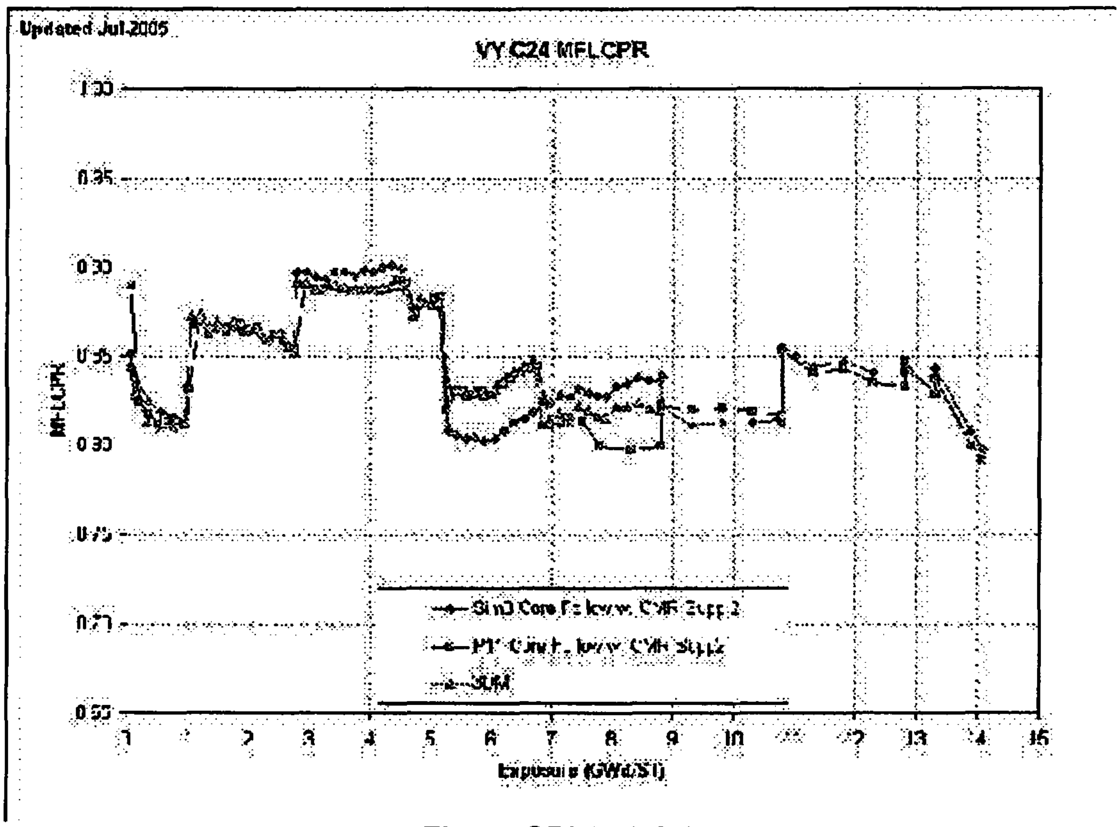

Figure SRXB-A-6-1
Cycle 24 MFLCPR

SIMULATE-3 MFLCPR is calculated through Entergy's own in-house code JAFCPR 2.1 which is fed power distributions and core parameters from SIMULATE-3. The code uses a similar approach to P11's CPR routine giving high confidence in its accuracy to calculate MFLCPR.

SIMULATE-3 shows good agreement with GNF methods as shown in *Figure SRXB-A-6-1* and there is substantial margin where the codes do not agree. The data points beyond the last 3DM point represent the as designed expected values for each thermal limit, in this case the Cycle 24 (current cycle) Cycle Management Report (CMR), Supplement 2.

MFLPD (Maximum Fraction Limiting Power Density)

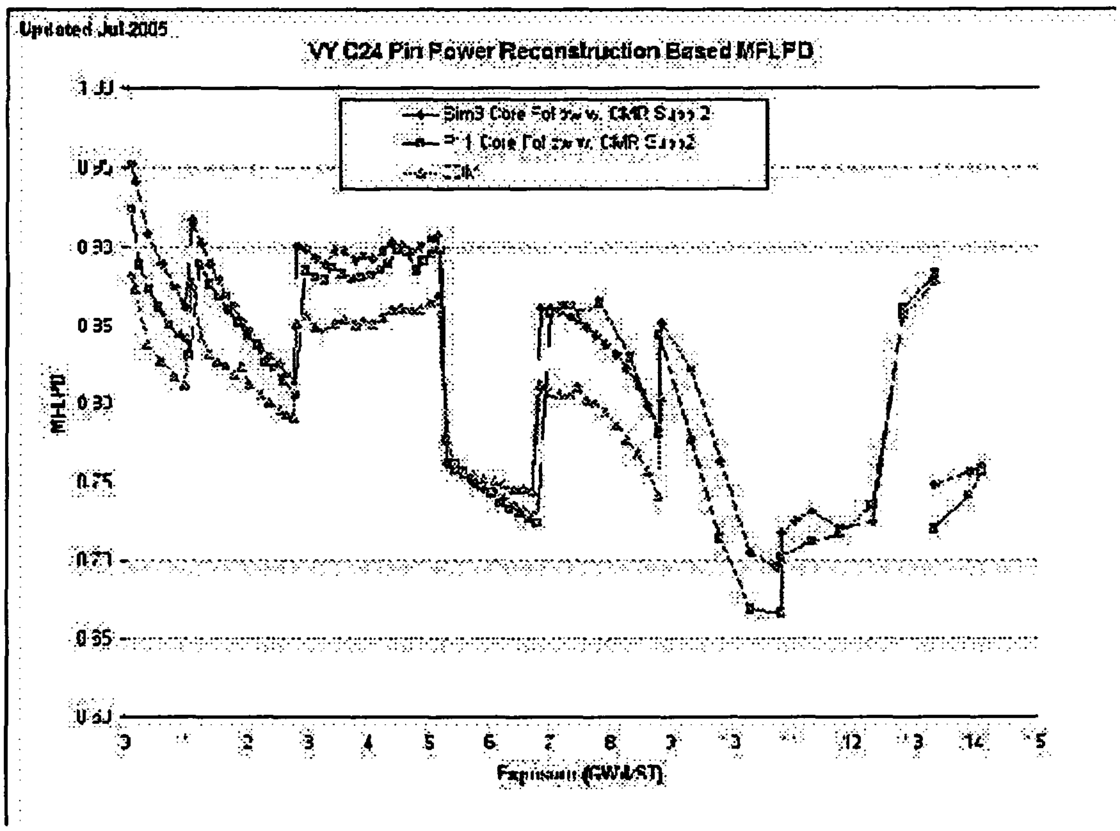

Figure SRXB-A-6-2
Cycle 24 Pin Power Reconstruction Based MFLPD

SIMULATE-3 uses its own pin power reconstruction module to determine MFLPD. As in *Figure SRXB-A-6-2*, in most cases, SIMULATE-3 and P11 calculate a larger and more conservative MFLPD than is representative in actual 3DM online operations, but the two offline codes agree relatively well.

MAPRAT (Maximum Average Planer Linear Heat Generation Ratio)

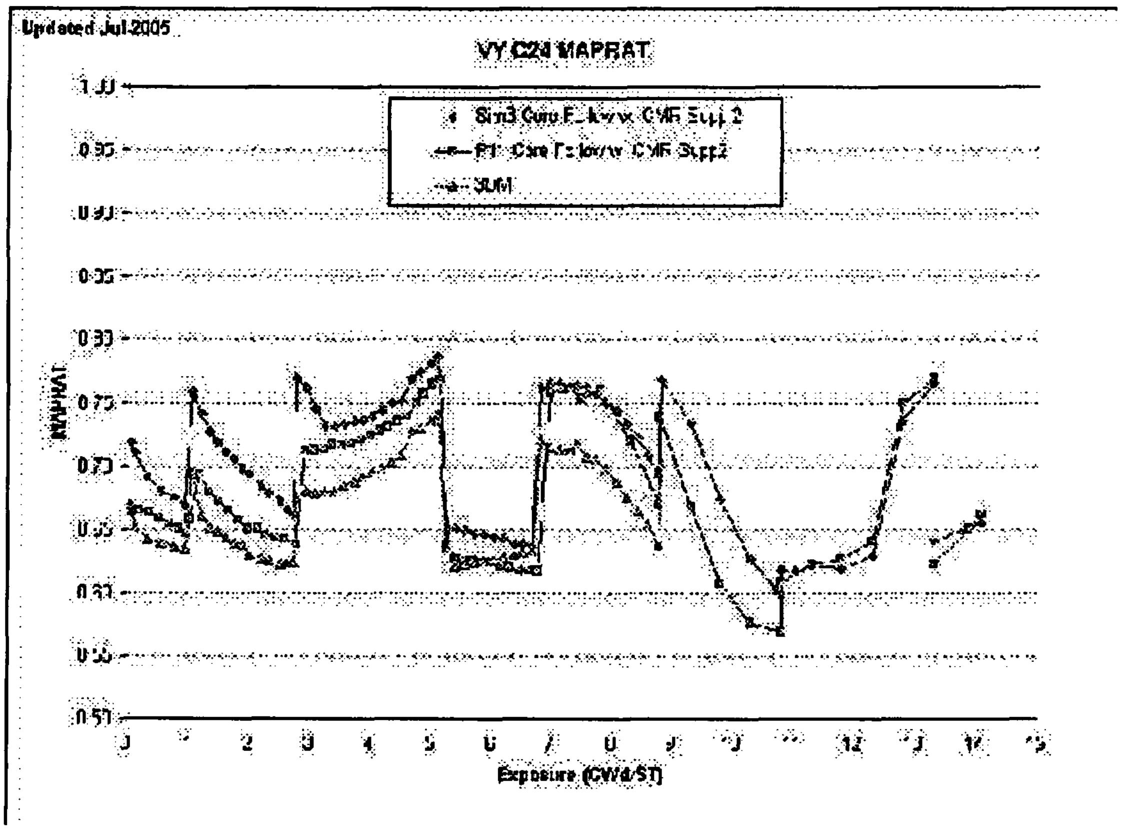

Figure SRXB-A-6-3 – Cycle 24 MAPRAT

MAPLHGR is a nodal parameter and requires no additional SIMULATE-3 module for its calculation. SIMULATE-3 and P11 have a fairly consistent bias over 3DM as shown in *Figure SRXB-A-6-3*. MAPRAT is typically the least limiting thermal limit at VYNPS.

Gamma TIPS

SIMULATE-3 TIPS are generated to confirm the accuracy of the model relative to the plant TIP data set. SIMULATE-3 TIP comparisons to 3DM are created when TIP data becomes available. The TIPS are produced in a Studsvik Scandpower post processing code known as S3post.

The following TIP output, *Figure SRXB-A-6-4*, depicts the comparison for VYNPS Cycle 24. The dashed line represents SIMULATE-3 and the solid line, 3DM. The radial, axial, and nodal RMS error values are calculated for each combined core average TIP. Any larger than expected deviations are reviewed with plant reactor engineers, nuclear engineering analysis (NEA) engineers, and the NEA manager, per the core follow procedure, to investigate the discrepancy and, if necessary, take action. The low RMS errors provide a high confidence that the VYNPS SIMULATE-3 model is correctly calculating the power distribution.

Technical Specification Reactivity Anomaly Surveillance

Technical Specification 3/4.3.E requires that at least every equivalent full power month, the critical rod configuration is compared to the expected configuration. These configurations are required to be within 1% $\Delta k/k$. The comparisons are performed using the eigenvalue calculated at that statepoint. As can be seen in *Figure SRXB-A-6-5*, for VYNPS Cycle 24 (current cycle), the hot eigenvalue has compared well with the predicted eigenvalue, as well as the GNF core follow eigenvalue. Agreement between these eigenvalues provides confidence that the actual plant operation follows core design.

It should be noted that VYNPS Cycle 24 has not yet ended and data is provided only through a partial cycle.

VERMONT YANKEE CYCLE 24 SIMULATE TO 3DM TIP COMPARISON - JULY 05

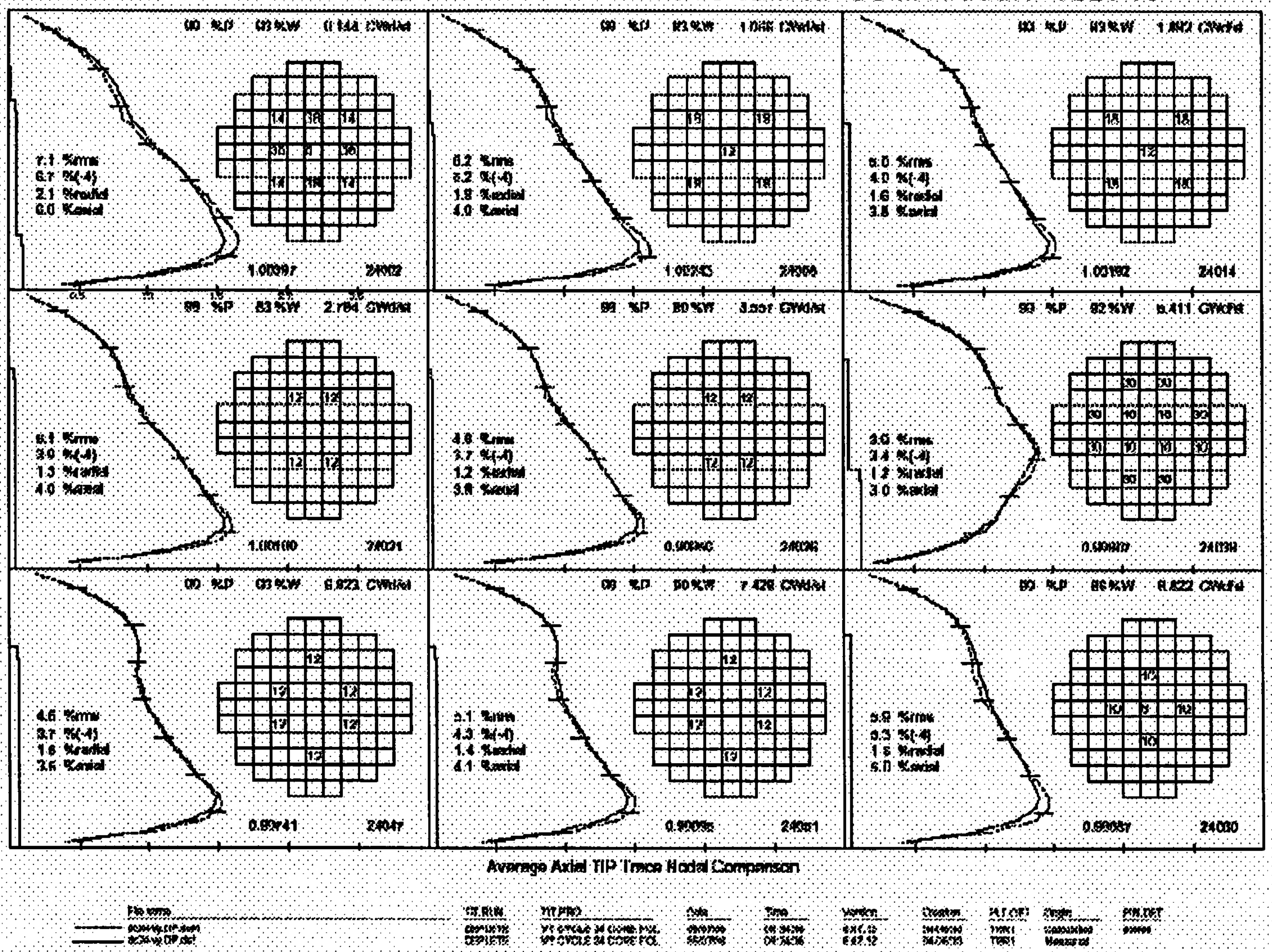

Figure SRXB-A-6-4
Cycle 24 TIP Comparison Data

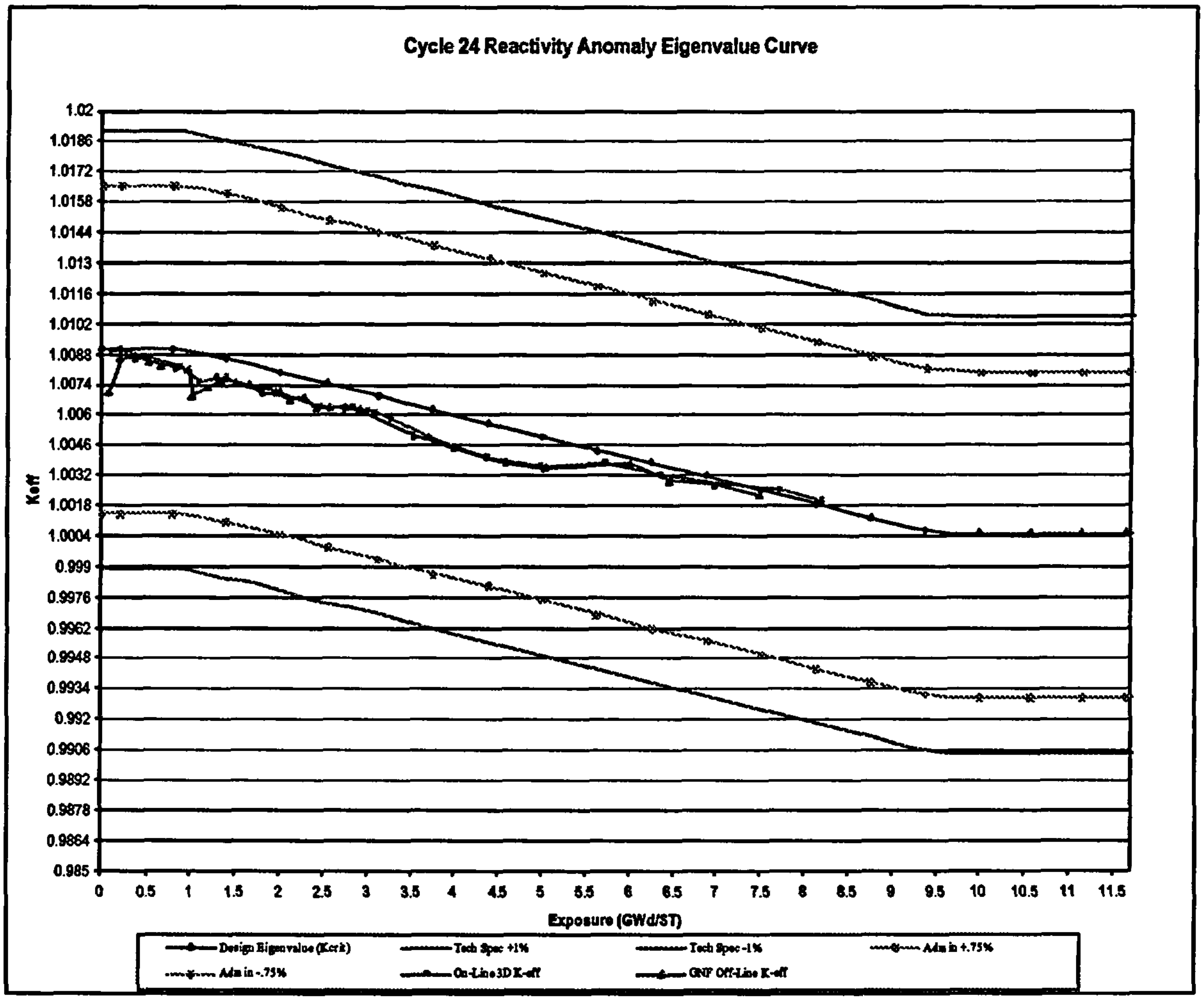

Figure SRXB-A-6-5
Cycle 24 Reactivity Anomaly Eigenvalue Curve

Conclusion

SIMULATE-3 gives high confidence through independent means that GNF methods are adequate to model the VYNPS reactor core. The model also gives high confidence that future design cycles are valid and sufficiently accurate to model EPU conditions.

The Technical Specification Reactivity Anomaly surveillance provides confidence that the actual plant operation follows core design.

BVY 05-072
Docket No. 50-271

Attachment 10

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 30

Extended Power Uprate

Response to Request for Additional Information

Responses to Reactor Systems Branch RAIs

NON-PROPRIETARY VERSION

Total number of pages in Attachment 10 (excluding this cover sheet) is 126.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides responses to the NRC Reactor Systems Branch's (SRXB) individual requests for additional information (RAIs) in NRC's letter dated July 27, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. In certain instances the intent of certain individual RAIs may have been modified based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The individual RAIs are re-stated as provided in NRC's letter of July 27, 2005.

Reactor Systems Branch (SRXB)
Boiling Water Reactors and Nuclear Performance Section (SRXB-A)

RAI SRXB-A-7

Table 1-1 of the VYNPS Power Uprate Safety Analysis Report (PUSAR) (i.e., Attachment 4 of the application dated September 10, 2003), lists computer codes used for CPPU for transient analysis. Please clarify which code was used for the over-pressure protection analysis.

Response to RAI SRXB-A-7

The over-pressure protection analysis was performed using version 10 of the ODYN code[2], which is applicable to plants that use variable speed pumps for recirculation flow control.

RAI SRXB-A-8

Section 3.10.1 of the VYNPS PUSAR discusses the shutdown cooling (SDC) analysis for CPPU. However, SDC with single loop operation was not discussed in the PUSAR. Please clarify which criteria apply to SDC with single loop operation, and whether the criteria are satisfied at CPPU conditions.

Response to RAI SRXB-A-8

[[

]] The reactor shutdown cooling under abnormal conditions with either loss of the normal suction flow path or loss of one SDC loop, is considered a licensing basis, not a design basis.

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

[2] "Qualification of the One-Dimensional Core Transient Model (ODYN) for Boiling Water Reactors," NEDO-24154-A, Vols. 1 – 3, August 1, 1986, NEDC-24154P-A Supplement 1, Volume 4, February 2000

There is no licensing basis or design basis requirement for single loop SDC analysis because VYNPS has no commitment to Reg. Guide 1.139, or Technical Specification requirement that specifies SDC operation using single RHR loop.

Evaluations have been performed for certain events (i.e., an Appendix R fire) that assume only the capability of one loop of RHR shutdown cooling. The SDC capability is evaluated at power uprate conditions as part of the specific event scenario in which SDC is credited, using assumptions appropriate for the specific event. Section 3.10.1 of the VYNPS PUSAR provides evaluation of the RHR SDC design basis consistent with the design and licensing basis for the VY SDC mode of operation.

RAI SRXB-A-9

Section 2.2 of the PUSAR states that a representative cycle core was used for the CPPU evaluation. Please define the VYNPS "representative" cycle core and discuss which GE fuel type is limiting from the standpoint of fuel thermal limits.

Response to RAI SRXB-A-9

A Power Uprate Representative Equilibrium Cycle (PUREC) core design was generated at LPU/100% rated core flow conditions representing a full-core loading of GE14 fuel and an 18-month equilibrium operating cycle. In addition, refer to the response to NRC RAI SRXB-A-19.

RAI SRXB-A-10

Please provide the following additional information regarding the VYNPS LBLOCA analysis for the CPPU:

a) Describe the VYNPS limiting single failure LBLOCA event for the current licensing basis conditions and for EPU conditions, respectively. Typically, the events are the same; but if the events are different for VYNPS, then explain the reasons. Also, describe the type of reactor core that was assumed for the EPU analysis (i.e., whether the core was assumed to be loaded with the same kind of GE fuel, or a mixed-core was assumed). If it was a mixed-core, then describe which GE fuel types used, their proportions and burnup level, etc.

b) The peak cladding temperature (PCT) changes due to CPPU are typically within 20 °F; but for the VYNPS EPU, it was determined to increase by 50 °F. Discuss the reasons behind such a comparatively large increase of the PCT, and why VYNPS is an exception in this regard.

Response to RAI SRXB-A-10

a) At both current licensing basis and EPU conditions, the limiting large break LOCA case for VYNPS is the maximum recirculation line break with a DC power source (battery) failure. This case results in the large break with the least amount of ECCS available.

Consistent with the approved SAFER/GESTR-LOCA methodology, the VYNPS EPU ECCS-LOCA analysis assumed an equilibrium core loading. This approach is acceptable because [[

]]

b) The <20°F impact due to EPU is the expected average result across the BWR fleet. However, the impact on a given plant may be larger due to plant specific characteristics. A comparison of the VYNPS Appendix K LOCA response for the two operating conditions show almost identical responses with the vessel water level for the EPU case reflooding the bottom of the core only slightly later (i.e., about 4 seconds). This is expected for the EPU case which initializes with higher voids (resulting in less initial vessel inventory) and less subcooling (resulting in more inventory loss due to flashing from depressurization). At the time of reflooding the cladding heatup rate for VYNPS is about [[ ]] which results in the [[ ]] difference in PCT. This rate is high because [[

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RAI SRXB-A-11

As shown in Supplement No. 4, Attachment 4 (NRC Review Standard RS-001, BWR Template Safety Evaluation (SE) as revised for VYNPS), Section 2.8.6, "Fuel Storage," draft General Design Criterion (GDC) 66 is applicable to the NRC's review of the affect of the proposed EPU on new and spent fuel storage. This GDC requires prevention of criticality in fuel storage systems by physical systems or processes, preferably utilizing geometrically safe configurations. The NRC staff did not find any discussion on criticality of new and spent fuel storage in the licensee's submittals. Please provide this information.

Response to RAI SRXB-A-11

For EPU, Vermont Yankee is not changing its licensing basis requirements for new or spent fuel storage, which are listed in Technical Specification 5.5. These include the following:

a) The new fuel storage facility shall be such that the effective multiplication factor (Keff) of the fuel when dry is less than 0.90 and when flooded is less than 0.95.

b) The Keff of the fuel in the spent fuel storage pool shall be less than or equal to 0.95.

c) Spent fuel storage racks may be moved (only) in accordance with written procedures which ensure that no rack modules are moved over fuel assemblies.

d) The number of spent fuel assemblies stored in the spent fuel pool shall not exceed 3353.

e) The maximum core geometry infinite lattice multiplication factor of any segment of the fuel assembly stored in the spent fuel storage pool or the new fuel storage facility shall be less than or equal to 1.31 at 20°C.

RAI SRXB-A-12

In Section 7.2.1, Step # 3, NUMARC 87-00, Revision 1, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," it is indicated that the minimum permissible usable gallons of water in the condensate storage tank (CST) should be recorded in the plant Technical Specifications (TSs). In Supplement No. 25, Attachment 2, Section 2.1, the licensee stated that in order to ensure that at least 100,000 gallons of usable CST inventory is available during an SBO, the minimum administrative limit for CST level identified in procedure OP 0150, "Conduct of Operations and Operator rounds," will be increased. The current CST minimum volume in TS 3.5.E.1.b is 75,000 gallons. Please justify why the TSs do not need to be revised, consistent with the recommendations in NUMARC 87-00.

Response to RAI SRXB-A-12

The purpose of Section 7.2.1 of NUMARC 87-00, Rev. 1 is to ensure that adequate condensate inventory is available for decay heat removal during an SBO event for the required coping duration. Procedural Step 3 of Section 7.2.1 provides one measure for assessing the adequacy of condensate inventory requirements and is not a recommendation to revise the TSs.

VYNPS relies on administrative controls (i.e., plant procedures) to meet the SBO condensate inventory requirements. Step 5 of Section 7.2.1 provides the criteria for establishing additional water sources in addition to the TS limit value of the condensate storage tank (CST). VYNPS's credit of the administratively controlled additional CST volume meets the four criteria specified in Step 5 of NUMARC 87-00, Rev. 1 and therefore is an acceptable means to demonstrate that acceptable water is available to support the coping period.

10CFR50.36(c)(2)(ii) establishes criteria for Technical Specification limiting conditions for operation. The CST inventory required for copying with an SBO event does not satisfy these criteria because it does not involve:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. An SBO event is not a design basis accident or transient event that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety. SBO risk is assessed to be low.

Furthermore, in accordance with 10CFR50.36(c)(2)(iii), a licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995, to satisfy the criteria in paragraph (c)(2)(ii) of this section.

RAI SRXB-A-13

As discussed in PUSAR Section 2.4, "Stability," VYNPS currently operates under the Option I-D solution. Please provide a clarification for the following areas:

a) The current flow-biased average power range monitor (APRM) scram provides automatic detection and suppression of core wide instability. Provide the technical basis that supports the conclusion that regional mode reactor instability is not probable under EPU conditions.

b) Describe any alternative method to provide automatic detection and suppression of any mode of instability other than through the current flow-biased APRM scram.

c) Describe how the dominance of the core-wide mode oscillations is maintained under the EPU conditions. Specifically, describe how the effects on axial and power distributions (which change for EPU core loadings) have been taken into account in the new calculations to ensure the dominance of the core-wide mode. Are there any negative effects on stability of the EPU core loadings?

Response to RAI SRXB-A-13

a) As stated in NEDO-31960-A[3], the Option I-D solution is applicable to plants that have relatively tight fuel inlet orificing and relatively small diameter cores. For plants with tight inlet orificing, the probability of regional oscillations is very low and the expected mode of oscillation is core-wide. In addition, plants with small diameter cores are also less likely to experience regional oscillations because of the strong preference of the fundamental mode of the neutronics. Even though the implementation of EPU at VYNPS does not change the fuel inlet orificing or the core diameter, the EPU core design might drive a higher core decay ratio due to higher power at extended operation. The APRM flow-biased flux scram line has been redesigned with three slopes for ARTS/MELLLA operation and rescaled for EPU operation such that the SLMCPR is protected against thermal-hydraulic instability events. [[

]] Calculations will continue to be performed for each VYNPS reload cycle to demonstrate that the probability of regional oscillations is very low.

b) Response: There is no other method to provide automatic detection and suppression of any mode of instability for Option I-D plants other than through the current flow-biased APRM scram.

[3] NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," Licensing Topical Report, November 1995

c) The dominance of the core-wide mode oscillation can be demonstrated by calculating the core and channel decay ratio at the most limiting power/flow state point (i.e., the intersection of Natural Circulation Line and MELLLA boundary). If the calculated channel decay ratio is = 0.56, the dominance of the core-wide mode of oscillation is demonstrated.

A limiting axial power shape with a 2.00 magnitude axial power shape peaked at node 3 is applied to the hottest channel and this gives a relatively conservative axial power shape in the hot channel decay ratio evaluation.

[[

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Calculations will continue to be performed for each VYNPS reload cycle to demonstrate that the hot channel decay ratio criterion is met and thus the probability of regional mode oscillations is very low.

RAI SRXB-A-14

Provide the technical basis that supports a conclusion that the hot bundle oscillation magnitude portion of the detect-and-suppress calculation is not dependent upon the core and fuel design.

Response to RAI SRXB-A-14

As stated in NEDO-32465-A[4], the hot channel oscillation magnitude (HCOM) is dependent on plant-specific factors. Some of the parameters that affect the hot bundle oscillation magnitude are: core size, LPRM assignments, trip setpoints, growth rate, harmonic power distributions (contours), LPRM failures, trip overshoot, and oscillation frequency. The only parameter that could potentially be affected by the core and fuel design would be the contours. However, the EPU power distributions are not significantly different from the pre-EPU power distributions to have a significant effect on the harmonic power distributions and hence, the hot bundle oscillation magnitude. Please note that the HCOM values have been revised for Cycle 24 EPU operation to demonstrate that the APRM flow-biased flux scram line provides adequate SLMCPR protection against thermal-hydraulic instability events.

RAI SRXB-A-15

The hot channel decay ratio provided for EPU is very close to acceptable criteria limits. In addition, the core-wide decay ratio is not provided. Have the proposed EPU core loadings degraded the stability performance significantly? Provide a table of hot channel and core-wide decay ratios at the most limiting state point for the last three cycles and the proposed EPU condition. The purpose is to evaluate the impact of the EPU on relative stability of the plant, and the applicability of Option I-D to VYNPS under these new conditions.

[4] NEDO-32465-A, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," Licensing Topical Report, August 1996

Response to RAI SRXB-A-15

The limiting hot channel and procedure core decay ratios at the most limiting state point for the last three cycles including the proposed EPU condition are provided in Table SRXB-A-15-1 below:

Table SRXB-A-15-1

Cycle	Limiting Hot Channel Decay Ratio	Procedure Core Decay Ratio
22	[[	
23		
24 (EPU)		]]

Note that the Cycle 22 decay ratios were computed with FABLE/BYPSS methodology and therefore are not directly comparable to the Cycle 23 and Cycle 24 results, which were computed with ODYSY methodology. (The limiting hot channel and procedure core decay ratios at the most limiting state point were not computed for Cycle 21.) By comparing the Cycle 23 and Cycle 24 results, it can be seen that the proposed EPU core loadings do not significantly impact the stability performance of the plant. [[

]] Therefore, the dominant mode of oscillations is core-wide and Option I-D is applicable to VYNPS under these new conditions.

RAI SRXB-A-16

It appears that the APRM flow-biased scram setpoint will be maintained at the same absolute levels (in terms of megawatts) for EPU as for CLTP. Please address the following:

a) Because the distance (in terms of megawatts (MWs)) between the most limiting power/flow operating point and the scram setpoint represents the oscillation amplitude required for scram, has this distance (i.e., the maximum oscillation amplitude) changed for EPU? Provide a graphical power/flow map representation of the new and old operating domains and the VYNPS scram setpoints, including the exclusion region. Note that the most limiting condition in terms of the oscillation amplitude is not necessarily the most unstable point, but the one that results in the largest amplitude.

b) If the above distance (i.e., the oscillation amplitude required for scram) has changed, is the CLTP scram setpoint still conservative for the EPU?

c) Has the resolution of the recent DIVOM (delta critical power ratio (CPR) over initial minimum CPR versus oscillation magnitude) 10 CFR Part 21 notification had any effect on VYNPS implementation of Option I-D? What DIVOM correlation is used to justify the EPU scram line? Is it a plant-specific or generic correlation? Please provide details.

Response to RAI SRXB-A-16

a) The rated rodline has been redefined for EPU and the APRM scram line has been rescaled for EPU. Therefore, as can be seen in Figures SRXB-A-16-1 and SRXB-A-16-2, the difference in power at natural circulation between the rated rodline and the APRM scram setpoint decreases from 15.3% of rated power for pre-EPU to 4.6% of rated power for EPU. This decrease in the power difference occurs for core flows higher than natural circulation as well. The corresponding hot channel oscillation magnitude (HCOM) is reduced for EPU. This is offset somewhat by a larger value of DIVOM for EPU. The net effect is an improvement in margin to the SLMCPR.

b) The APRM scram setpoint has been rescaled for EPU and the scram setpoint is conservative for EPU. Calculations will continue to be performed for each VYNPS reload cycle to demonstrate that the SLMCPR is protected against thermal-hydraulic instability events.

c) [[

]] This conservative DIVOM slope addresses the recent DIVOM 10 CFR Part 21 issue. Consistent with the resolution of the Part 21 issue, for future cycles, VYNPS plans to use a plant- and cycle-specific core-wide mode DIVOM curve in accordance with the BWROG Plant-Specific Core-Wide Mode DIVOM Procedure Guideline (GE-NE-0000-0031-6498-R0, June 2, 2005).

Vermont Yankee EPU, Power/Flow Map

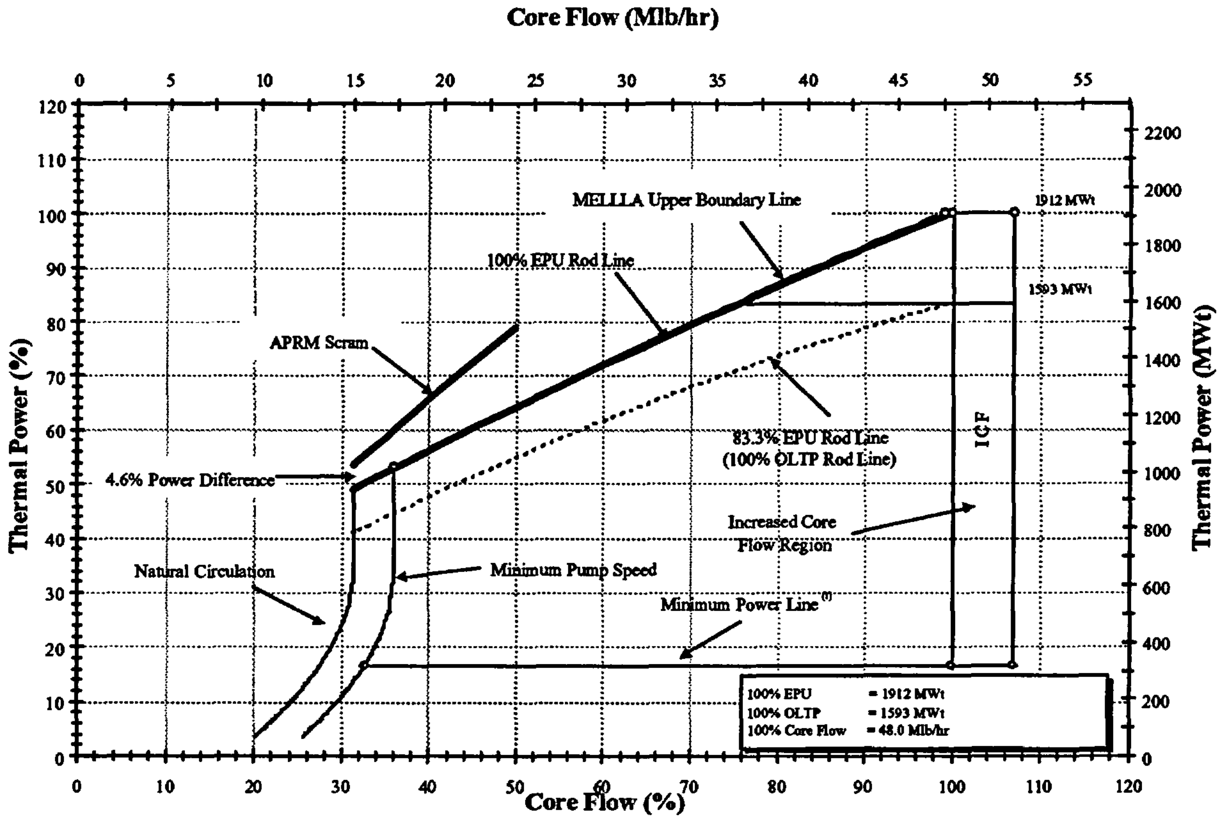

Figure SRXB-A-16-1

Vermont Yankee Pre-EPU, Power/Flow Map

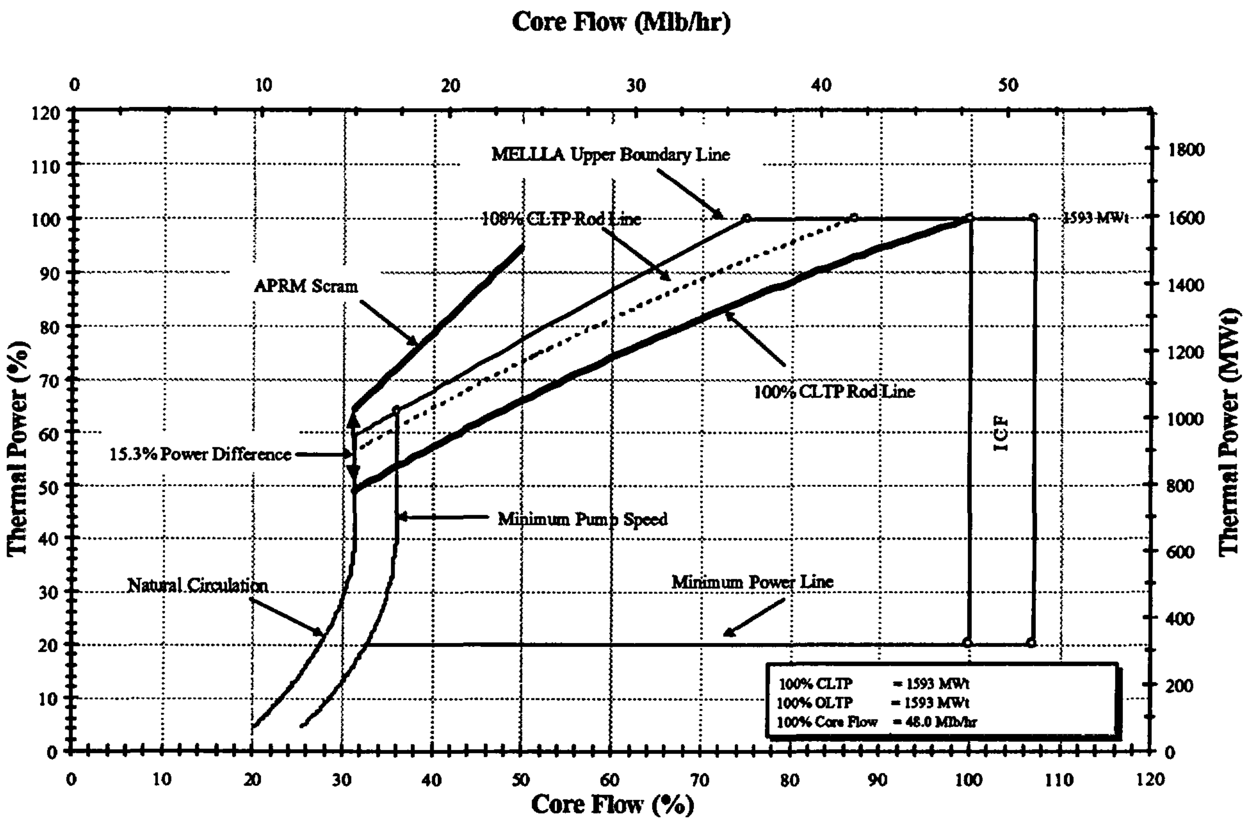

Figure SRXB-A-16-2

RAI SRXB-A-17

In Supplement 4, Attachment 5, Matrix 8, page 13, note for SE Section 2.8.5.4.1, there is an explanation for uncontrolled control rod withdrawal from a subcritical or low power startup condition. In this explanatory section, this event is considered as an accident and a fuel enthalpy of 170 calories/gram is given as the acceptance criterion. However, in SRP Section 15.4.1, this event is considered as a transient, not as an accident, and hence specified acceptable fuel design limit criteria is applied. Why is this event considered as an accident rather than a transient?

Response to RAI SRXB-A-17

This event is indeed considered to be a transient consistent with the SRP. For these low power conditions, the acceptance criterion for transients is the cladding failure threshold of 170 cal/gram, which is much lower than the acceptance criterion of 280 cal/gram for accidents such as a Control Rod Drop event. The approved GESTAR fuel Licensing Topical Report (LTR) NEDE-24011-P-A-14-US refers to NEDO-10527 (GESTAR Reference S-12), which documents both the criteria and methods used in the RWE transient. The analysis of the RWE transient is documented in NEDO-23842.

RAI SRXB-A-18

Review Standard RS-001, BWR Template SE for Sections 2.8.5.1, 2.8.5.2.1, 2.8.5.2.2, 2.8.5.2.3, 2.8.5.3.1, 2.8.5.4.3, 2.8.5.5 and 2.8.5.6.1, guides the NRC staff to reach a conclusion regarding reactor coolant pressure boundary (RCPB) pressure limits not being exceeded. However, the revised template reflecting the VYNPS licensing basis (provided in Supplement Nos. 4 and 8) does not include any acceptance criteria in the "Regulatory Evaluation" portion of each of these SE sections related to the RCPB. Please confirm that draft GDC-9, "Reactor Coolant Pressure Boundary," is applicable to these sections and provide a markup of the SE template accordingly.

Response to RAI SRXB-A-18

Draft GDC-9, "Reactor Coolant Pressure Boundary," is applicable to Review Standard RS-001, BWR Template SE for Sections 2.8.5.1, 2.8.5.2.1, 2.8.5.2.2, 2.8.5.2.3, 2.8.5.3.1, 2.8.5.4.3, 2.8.5.5 and 2.8.5.6.1. See the SE template markups for these sections as Exhibit SRXB-A-18-1 in Attachment 11.

RAI SRXB-A-19

The following question relates to the review for SE template Section 2.8.5.4.3, "Startup of a Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in Core Flow Rate."

VYNPS UFSAR Section 14.5.6 states that: "[f]low dependent operating limits, MCPR(F) [minimum critical power ratio (MCPR) flow-dependent limit], LHGRFAC(F) [linear heat generation rate (LHGR) flow-dependent multiplication factor] and MAPFAC(F) [maximum average planar linear heat generation rate (MAPLHGR) flow-dependent multiplication factor] are developed to ensure that core thermal limits are not violated for the limiting flow increase transients." This UFSAR section also states that "[t]hese flow-dependent limits are generic ARTS [APRM and rod block monitor TS] program limits and are derived from a conservative treatment of a two recirculation pump slow flow runout event. The validity of the flow-dependent limits for the core flow increase transients was reconfirmed for the GE14 fuel introduction." Confirm that validity of the flow-dependent limits were verified for the EPU operating conditions.

Response to RAI SRXB-A-19

It has been shown that the operating limit MCPR is not significantly affected and that the power and flow dependent limits are unaffected by CPPU[5]. The licensed core flow and the conservative control rod line are unchanged with EPU. Consistent with the CPPU methodology, the rated and off-rated MCPR operating limits are established for each fuel cycle as part of the reload analysis.

RAI SRXB-A-20

With respect to PUSAR Section 6.5, "Standby Liquid Control System":

a) The results of the licensee's anticipated transients without scram (ATWS) analyses at EPU conditions determined that the calculated peak vessel bottom pressure is 1490 psig as shown in PUSAR Table 9-5. However, the standby liquid control system (SLCS) pump discharge pressure value proposed for the surveillance test is only 1325 psig (reference proposed revision to Surveillance Requirement (SR) 4.4.A.1). Clarify why this test pressure is acceptable.

b) PUSAR Section 6.5 states that because of the increase in SLCS pump discharge pressure under EPU conditions, the surveillance test pressure in SR 4.4.A.1 will be increased from 1320 psig to 1325 psig. What is the SLCS discharge relief valve setpoint under EPU conditions? Taking relief valve setpoint tolerance into consideration, how much margin is there to prevent the relief valve from lifting?

Response to RAI SRXB-A-20

a) The peak pressure referenced above (1490 psig at the vessel bottom) occurs very early in the transient event. Depending on the event, the first peak generally occurs within the first 10 to 50 seconds (see Figure SRXB-A-20-1). The SLCS pump discharge test pressure of 1325 psig is based on the peak reactor pressure that occurs during SLCS operation. This pressure is 1292 psia (1277 psig), occurring during the PRFO BOC event.

b) The minimum SLCS pump relief valve nominal setpoint for EPU is 1400 psig. Based on the 1325 psig discharge test pressure, there is a minimum of 75-psi margin. This margin provides allowance for SLCS pump relief valve setpoint drift and for SLCS pump

[5] "Constant Pressure Power Uprate", NEDC-33004P-A, Revision 4, July 2003

pressure pulsations. The GE recommendation for relief setpoint margin is 75 psi.

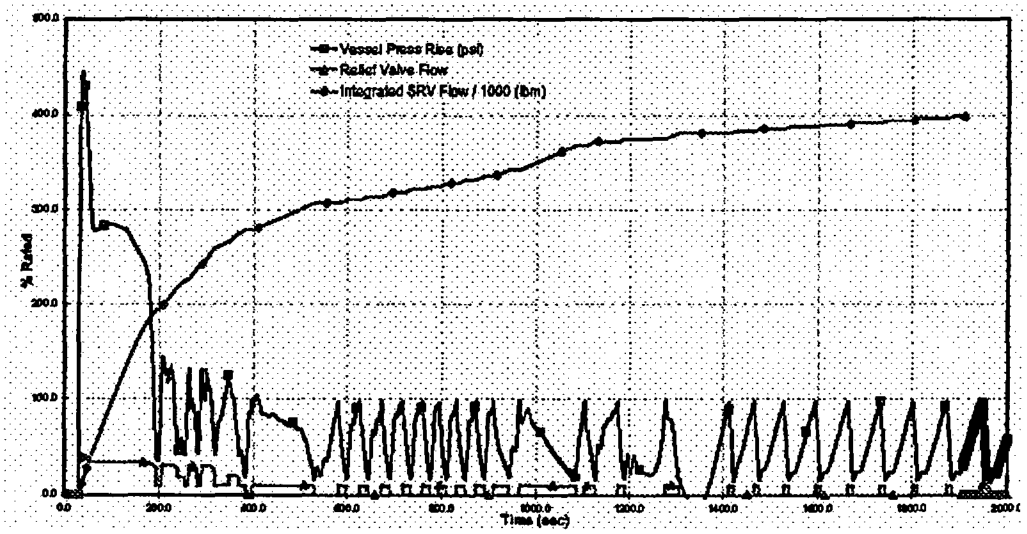

Figure SRXB-A-20-1
Vermont Yankee Long-Term PRFO Transient Response at LPU and BOC

RAI SRXB-A-21

With respect to PUSAR Section 9.1, "Anticipated Operational Occurrences," identify the staff approved evaluation model used for the plant-specific loss of feedwater flow event analysis.

Response to RAI SRXB-A-21

The SAFER code was used in the analysis of long-term Loss-of-Feedwater events.

RAI SRXB-A-22

With respect to PUSAR Section 9.3.1, "Anticipated Transients Without Scram:"

a) Identify the staff-approved evaluation model used for the plant-specific ATWS analysis.

b) Confirm that operator actions specified in the VYNPS emergency operating procedures are consistent with the generic Emergency Procedure Guidelines/Severe Accident Guidelines insofar as they apply to the operator actions for ATWS. Specify the time delay used in the ATWS analysis for starting of the SLCS pumps.

Response to RAI SRXB-A-22

a) The ATWS reactor transient analysis was performed using version 10 of the ODYN code

(Ref. 1), which is applicable to plants that use variable speed pumps for recirculation flow control. The GE computer model STEMP04 was used for the suppression pool heatup calculation. The analytical models of STEMP have been accepted by the NRC in previous applications (Ref. 2) and other ATWS analyses.

b) The VYNPS Emergency Operating Procedures (EOPs) are consistent with Revision 2 of the Boiling Water Reactor Owners' Group (BWROG) Emergency Procedure and Severe Accident Guidelines (EPGs/SAGs), insofar as they apply to the operator actions for ATWS. With respect to the EPGs, plant specific setpoints, limits, equipment, and operating characteristics are substituted, as necessary, to ensure fidelity to the plant.

As discussed in the VYNPS PUSAR section 9.3, plant-specific analyses were performed to ensure the ATWS acceptance criteria of peak vessel bottom pressure less than 1500 psig (the ASME Service Level C limit), peak suppression pool temperature less than 281 degrees F (the wetwell shell design temperature), and peak containment pressure less than 62 psig (110% of drywell design pressure), were met. The limiting events with respect to these criteria are the Pressure Regulator Failure – Open (PRFO) and Main Steam Isolation Valve Closure (MSIVC) events. Each event was analyzed at beginning-of-cycle and end-of-cycle conditions.

As noted in the NRC SER to the Constant Pressure Power Uprate (CPPU) Licensing Topical Report (CLTR), NEDC-33004P-A, boron injection from SLC is assumed to start at the later of either (1) reaching the boron injection initiation temperature (BIIT) or (2) two minutes after the ATWS recirculation pump trip on either low reactor level or high reactor pressure. In the ATWS analyses of VYNPS at EPU conditions, the SLC initiation is assumed to occur two minutes after the recirculation pump trip on high reactor pressure.

In addition to boron injection, a number of other operator actions (consistent with the EPGs) are assumed in the VYNPS EPU ATWS analyses. These operator actions are assumed to occur at the same time or later than the timing assumed for the pre-uprate ATWS analyses, consistent with the NRC SER for NEDC-33004P-A.

The ATWS analyses methodology assumes operator action to reduce feedwater flow to the reactor in order to decrease reactor water level. This action occurs at the later of either reaching the BIIT or 90 seconds after the MSIV closure. In the ATWS analyses of VYNPS at EPU, this event was assumed to be initiated by operator action at the BIIT.

Finally, the ATWS analytical methodology assumes operator action to initiate torus cooling. For VYNPS the time at which operators initiate torus cooling was increased from the 10 minutes assumed in the pre-uprate ATWS analysis to 15 minutes at EPU conditions. This assumption, while increasing margin for operator action, is analytically more conservative because it allows additional torus water heat-up prior to initiating cooling.

In summary, operator actions assumed in the VYNPS EPU ATWS analyses are consistent with the operator actions in the EPGs and documented in the NRC SER for the CLTR.

RAI SRXB-A-23

Supplement No. 24, Attachment 3, Table 6-4, "Metric Summary for VYNPS (120%)" presents the predicted maximum bundle powers and bundle power-to-flow ratios with exposure for the projected uprated conditions. In support of the staff's review of the LOCA analyses, please provide the following information specific to VYNPS:

a) For the peak power fuel assemblies, provide the limiting axial power distributions and radial peaking factors. For different exposures, select bundles with limiting axial power peaking operating with bottom peaked, double-hump or mid-peaked, and top peaked axial power distributions. Please assure that the axial power distribution corresponding to the exposure with the highest hot bundle exit void fraction is also provided.

b) Include in the selected bundles, the power distribution and peaking corresponding to the maximum powered bundle selected for the cycle state point of 13.184 gigawatt days per standard ton. Table 6-4 also shows that the bundle is operating at 7.51 MW. Please provide the corresponding predicted bundle operating conditions, including axial power distribution, void fraction distribution and bundle nodal exposure.

c) Please also include the bundle inlet mass flow rate and inlet temperature.

Response to RAI SRXB-A-23

The requested data are provided in the following tables at selected points during the cycle as well as 13.184 GWd/ST. A range of bottom-peaked, mid-peaked, and top-peaked axial shapes are included.

[[

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RAI SRXB-A-24

Supplement No. 24, Attachment 3, Figure 6-6, presents the peak LHGR [[ ]] Are bundles (pins) setting the peak limit for non-GE14 fuel? What uncertainties are applied to the peak LHGR to account for the calculational uncertainties?

Response to RAI SRXB-A-24

The peak LHGR values for Cycle 18 of Plant A are indeed close and sometimes [[ ]] These peak LHGR values came from a simulation of Cycle 18 operation and are a result of the over-calculation of peak LHGR by the predictive off line nuclear methods. The limits in question are for GE14 fuel. In actual operation, the peak LHGR never exceeded the limit value (see response to RAI-5 of MFN 05-029 TAC NO. MC5780). The uncertainties applied to the peak LHGR value are discussed in the alternate approach and the response to RAI SRXB-A-41.

RAI SRXB-A-25

Explain the reason for the increase in the [[ ]] The RAI response proposed void fraction weighting]] Justify why the nodal uncertainties for [[ ]] in order to establish the uncertainties that should be applied to VYNPS bundle powers and thermal limits.

Response to RAI SRXB-A-25

It is assumed that in the first part of the question, by "nodal uncertainties," is meant pin power peaking uncertainties. The data referred to are not used to establish the power allocation factor uncertainty, but were used to confirm the pin power uncertainty obtained from Monte Carlo comparisons, manufacturing uncertainties, and channel bow uncertainties. The overall derived uncertainty is [[ ]] The model differences have been evaluated and are correctly accounted for in NEDE-32601-P-A. The influence of pin power peaking on the R-factor uncertainty is discussed further in the response to RAI SRXB-A-41. The second part of the question refers to application of nodal uncertainties to VYNPS bundle power uncertainties. The process computer application of the PANAC core simulator model adapts the solution such that the axial shape is consistent with the measured TIP distribution. The radial component of the error is used to evaluate the uncertainty in the process computer bundle power. The practice of using the average RMS over a number of plants was approved by the staff in NEDE-32601 and NEDE-32694. This average is a best estimate of the uncertainty in the bundle power. In most of the US applications, the bundle RMS is [[ ]] assumed in the SLMCPR evaluation. This procedure, along with periodic updates to assure the accuracy is not changing, is adequate to ensure the plant operates within limits.

RAI SRXB-A-26

[[ ]] Demonstrate that for the 20% uprate condition for the entire operating domain, VYNPS would not operate with core power/flow ratio greater than [[ ]].

Response to RAI SRXB-A-26

The uprated power for VYNPS is 1912 Mwt, and the 100% flow value is 48 Mlbm/hr. Hence at the 100% power/flow point, the power/flow metric has a value of 39.8. The vast majority of operating history will be accumulated at this point.

With regard to the nodal and axial uncertainties, they are relevant only for the predictive capability of the 3-D Simulator. In monitoring mode, the solution is adapted to the measured TIP and LPRM readings (see response to RAI SRXB-A-28). The monitor nodal uncertainty depends on the radial uncertainty, (See NEDE-32694, response to question II.5) which is shown to [[ ]]. The nodal uncertainty in the monitoring mode is [[ ]]

RAI SRXB-A-27

State what criteria are used to establish that the axial and nodal uncertainties are acceptable and do not reflect degradation of the neutronic methods predictions of the nodal and axial power distribution and peaking.

Response to RAI SRXB-A-27

Axial power distributions have an impact on both the expected thermal margins for a particular nuclear design, as well as the accuracy of the cycle length prediction. The alternate approach discusses the relationship between methods accuracy and design margin. If the methods errors fall outside the design margins, the design will not meet its requirements. Also, if the methods uncertainty of the process computer model is significantly larger than those assumed in the SLMCPR and LHGR limits, the safety limits have to be revised.

Inaccuracy in axial power distribution may also indicate a fundamental problem in the fuel or poison depletion model. Most of the time, these effects show up when a significant change is made either in the power density, discharge exposure, or cycle length for a particular plant. Core tracking calculations are performed in the predictive mode in order to continually assess the accuracy of the nuclear methods and to verify the design margin used in the design of future operating cycles. Further, the RMS difference of all predictive to measured TIP responses results in an average difference of less than [[ ]] for the reference BWRs, indicating that axial power distributions are also predicted adequately. The radial RMS is checked to see if the average over a number of plants exceeds the [[ ]] assumed in the SLMCPR analysis. Criteria for the nodal RMS are not reflected in any licensing analysis, but generally any nodal RMS values over [[ ]] observed consistently require further explanation and review of the nuclear methods accuracy.

RAI SRXB-A-28

State where the axial and nodal uncertainties are accounted for in the thermal limits calculations and safety limit analyses.

Response to RAI SRXB-A-28

The relationship between thermal limits, safety limits and methods accuracy is discussed in the alternate approach section. A number of questions have related to the impact of axial or nodal uncertainties on thermal limits. In order to clarify all of the answers, it is important to emphasize the dual role of the nuclear methods. The 3D simulator, PANACEA, is used in both the predictive and monitoring mode.

In the monitoring mode, the simulator is part of the plant process computer. The plant process computer accepts core operating conditions, and in-core instrument reading from TIPs and local power range monitors (LPRM) and converts them into a three dimensional map of linear heat generation rates and critical power ratios. The reactor operator relies on the process computer output to ensure that the LHGR and Safety Limits are not violated during operation. The uncertainties input to the SLMCPR and limiting LHGR are those applied in the monitoring mode. In order to increase the accuracy of the process computer, GE developed an adaptive mode for the 3-D simulator. The adaptive mode for the original model (PANAC10) is described in NEDE-32694-PA and NEDC-32773 for the current model (PANAC11). In the adaptive mode, adjustments are made to the local diffusion parameters such that the axial distribution of TIP and LPRM readings calculated by the simulator are identical to the measured values. Hence in the monitoring mode, the axial RMS difference is zero, and the nodal TIP RMS is equal to the radial TIP RMS. The SLMCPR involves uncertainties in axially integrated quantities and is a function of the radial TIP RMS. The LHGR uncertainty is related to the nodal RMS, which is the same as the radial RMS. The LHGR uncertainty also includes the uncertainty in the TIP and LPRM signals.

The predictive mode is used in fuel and operating strategy design and as input to transient analysis. In the predictive mode there are no in-core instruments to normalize to, so there may be differences between the real and calculated axial power distributions. In the predictive mode, differences between the projected operating results and the actual operating history is accounted for by using design margin. In operation, the ratio between operating MCPR or

LHGR must never exceed 1.0. The designer is constrained to design the core such that there is a given amount of margin between the expected limiting value and the safety limit. Core tracking calculations, such as those presented in the answers to RAI-5, RAI-25, and RAI-27 of MFN 05-029 TAC NO. MC5780, are performed in the predictive mode in order to continually assess the accuracy of the nuclear methods and to verify the design margin used in the design of future operating cycles. The radial RMS is checked to see if the average over a number of plants exceeds the [[ ]] assumed in the SLMCPR analysis. Criteria for the nodal RMS are not reflected in any licensing analysis, but generally any nodal RMS values over [[ ]] observed consistently for several cycles require further explanation and review of the nuclear methods accuracy.

RAI SRXB-A-29

State whether or not the axial and nodal uncertainties are accounted for in the initial steady-state conditions calculations for the transients and accident conditions (ECCS-LOCA). For example, the axial power peaking and distribution affect the response to LOCA and transients. Therefore any under-predictions in the axial nodal powers could change the plants response. If nodal and axial uncertainties are not applied, justify how potential under-predictions in the axial and nodal powers are accounted for.

Response to RAI SRXB-A-29

The conservatism of the inputs to transient evaluations is assured by assuming an operating strategy yielding the most severe transient impact. The primary mechanism for shutting down a transient event is control blade scram. Control blade scram effectiveness is minimized when the power distribution is peaked at the top of the core. GE design procedures require a bounding analysis, which minimizes the effectiveness of the scram function for input to transient analyses. These bounding analyses result in greater penalties to the transient behavior than any caused by misprediction of axial power by the nuclear methods. This bounding analysis is further described in the alternate approach section and in the response to RAI SRXB-A-48.

RAI SRXB-A-30

Explain how the axial and nodal uncertainties are applied to the nodal exposures, the operating MAPLHGR (MAPRAT), and the operating LHGR (PKLHGR), to account for the nodal inaccuracies of the steady-state neutronic method and code systems.

Response to RAI SRXB-A-30

The alternate approach section contains a discussion of the relationship of nuclear and other uncertainties on the determination of LHGR and MAPLHGR limits. The LHGR limits include an allowance for monitoring uncertainties. In addition the fuel rod in question is assumed to operate on limits for its entire history, thereby maximizing the generation of fission product gasses and susceptibility to transient events. The peak clad temperature (MAPLHGR) evaluation conservatively [[ ]]. For VYNPS, the axial power distribution is assumed to be a cosine shape. (See reply to RAI SRXB-A-23)

RAI SRXB-A-31

[[

]] Comparing against a fixed limit value assumes that the selected maximum powered bundles are limited to low exposures bundles. State whether the ratio of the peak LHGR to the LHGR limit monitored in the core simulator (e.g. 3D MONICORE) includes the decrease of the LHGR limit with exposure.

Response to RAI SRXB-A-31

The metrics tabulated in Table 6-2 and Figure 6-6 of the original response to RAI SRXB-A-6 are meant to illustrate the highest LHGR in the experience base and were not intended to be compared against the limits curve. For VYNPS, the thermal-mechanical limits are incorporated into the monitoring system directly as a function of rod type (product line, local Gd concentration, and maximum Gd concentration anywhere in the rod) and local (pin) exposure. Therefore, the decrease of the LHGR limit with exposure is accounted for.

RAI SRXB-A-32

State if any uncertainty is applied to the LHGR limit in the core monitoring system and if the uncertainty value is increased with exposure as the limit decreases. This is important since the peak reactivity and nodal powers increase with exposure, when the Gadolinium (Gd) burns out (>8 GWD/ST) and the thermal-mechanical limit decreases with exposure at approximately 15 GWD/ST.

Response to RAI SRXB-A-32

The treatment of LHGR uncertainties is discussed in the alternate approach section and the response to RAI SRXB-A-31.

RAI SRXB-A-33

Describe how the core monitoring system and offline calculations calculate the peak nodal LHGR and the corresponding pin-wise peak LHGR, MAPLHGR and the accumulated exposure.

Response to RAI SRXB-A-33

The three dimensional power distribution, adapted to the TIP and LPRM response, is determined by the 3-D simulator. The method for the PANAC11 system used at VYNPS is similar to the one described in NEDC-32694P-A and is described in specific detail in NEDC-32773P. The nodal power, p_{node}, represents the average power in a roughly 6 inch cube at a particular place in the core. For a given fuel type, a set of infinite lattice pin peaking factors are stored as a function of exposure, history water density and water density. The infinite lattice peaking factors are then corrected for fast and thermal flux gradients across the node in question. Fluxes and powers from adjacent nodes are used to construct the correction factors. Pin powers are then established for each pin segment in each node in the core. Similarly, a pin exposure distribution is established. Limiting pin power values are established for each pin in the node, depending on the type of fuel rod and the local pin exposure. The maximum calculated-to-limiting power ratio is established for the node in question and also for the entire core. The MAPLHGR is related to the average pin power in the node. The average pin power can also be evaluated and compared to a limit usually constant with exposure but is a function of bundle mechanical configuration.

RAI SRXB-A-34

Since no gamma scans or isotopic inventory measurements are available for the current fuel designs (GE14) as operated, justify why it is acceptable to base the assessment of the predictions of the nodal powers and the pin powers on code-to-code comparisons.

Response to RAI SRXB-A-34

The process computer nodal power can be thought of as the product of three quantities. 1) The average radial four bundle power associated with a given TIP string, 2) an axial distribution obtained from the TIP response, and 3) a bundle power allocation factor giving the power distribution for the four bundles surrounding the TIP string. Factor 1) has been extensively qualified for recent applications [[ ]] Factor 2) comes directly from plant instrumentation. Qualification of factor 3), the power allocation factor has been performed to this point with the results of bundle gamma scan data. GE recognizes that the gamma scans were carried out on older designs and applications, and is currently acting to obtain additional gamma scan data. The new gamma scan data will form the basis for a revised value for the power allocation uncertainty. Until these new data are obtained it is proposed that the current standard deviation of [[ ]] be increased to [[ ]] For additional information on the proposed uncertainties for the SLMCPR, refer to the alternate approach section.

RAI SRXB-A-35

Provide VYNPS-specific core follow data for operation at the current licensed thermal power. Include axial individual TIP measured versus calculated data for limiting conditions in terms of four-bundle nodal powers. Provide the individual TIP data for different power peaking (top-peaked, bottom-peaked and mid-peaked or double-hump). Include the corresponding four bundle predicted conditions (e.g., void fractions, nodal exposures).

Response to RAI SRXB-A-35

It is requested that the measured and calculated TIP readings surrounding hot bundle locations be examined in more detail so as to exclude effects of statistical averaging on the comparisons using plant instrumentation. The following figures and tables provide the information requested which demonstrate the accuracy of the nuclear methods for VYNPS for Cycles 23 and 24. These data are the off-line calculated instrument readings compared to the measured instrument readings from the plant. The power and void fraction data associated with these figures and tables are from the off-line calculations using TGBLA and PANAC.

For each plant cycle, the following information is provided for one or more points in each cycle.

1. A summary through the cycle of TIP radial, axial and nodal core average RMS differences between calculated and measured TIPs. In addition, the summary indicates the radial, axial and nodal differences for the TIP instrument that is in the hot channel. In this case, the "hot channel" is defined as the instrumented location with the highest power bundle. Cycle points where more detailed data are provided is also indicated.
2. For each selected cycle exposure point (noted as 'B' for BOC, 'M' for MOC, or 'E' for EOC), a map by TIP string of the radial and axial TIP RMS difference across the core is provided. If the TIP instrument is "Failed", the TIP string was declared non-operational by the plant process computer and may be ignored. The failed TIP strings are marked in these maps.
3. For each cycle exposure point selected for more detailed study, a plot of each measured and calculated TIP instrument reading as a function of axial height is provided. The TIP plots are arranged on a core-wide map to give the relative position of the TIP within the core.
4. For each cycle exposure point selected for more detailed study, a numerical table with the nodal powers, exposures and void fractions corresponding to the four bundles surrounding the instrument adjacent to the hot channel and one additional channel is provided. The choice of the additional channel is arbitrary. An attempt was made to find a string with a different axial shape than the one adjacent to the hot channel.
5. All of the TIP comparisons exclude node 1, 24, and 25. The TIP strings are 144 inches long and hence, go to node 24. Node 24 has been eliminated because on some plants there is TIP string hardware that distorts the signal. The power is very low in this node and does not affect the overall power distribution. Node 1 is eliminated because it usually is loaded with natural uranium, the power is low, and it contains the effects of the lower tie plate.

The overall statistics for the VYNPS core follow shows good power distribution agreement between the measured and calculated data. The average Radial RMS is [[ ]] for Cycle 23 and [[ ]] for Cycle 24. The nodal RMS agreement is excellent, [[ ]] for Cycle 23 and [[ ]] for Cycle 24.

Tables SRXB-A-35-2 and SRXB-A-35-14 contain eigenvalue and thermal limit tracking data for Cycles 23 and 24 respectively. The eigenvalue behavior is consistent with that observed in all of the benchmark plants. The only exception to this consistent eigenvalue behavior is Cycle 11 of Plant E. This exception is not due to power uprate, but is due to an initial reload with a very large batch fraction (close to 50%).

VYNPS Cycle 23

Table SRXB-A-35-1 VYNPS TIP Comparison Data as a Function of Cycle 23 Exposure

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Table SRXB-A-35-2 VYNPS Cycle 23 Summary Data

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Table SRXB-A-35-3 VYNPS Cycle 23 and Cycle 24 Cold Critical Data

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Table SRXB-A-35- 4 VYNPS Four Bundle Powers and TIP String Comparisons at BOC

[[

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Figure SRXB-A-35-1 VYNPS Cycle 23 TIP String Comparisons at BOC

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Table SRXB-A-35-5 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 12] at BOC

[[

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Table SRXB-A-35-6 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 5] at BOC

[[

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Table SRXB-A-35-7 VYNPS Four Bundle Powers and TIP String Comparisons Cycle 23 MOC

[[

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Figure SRXB-A-35-2 VYNPS Cycle 23 TIP String Comparisons at MOC

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Table SRXB-A-35-8 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 16] at MOC

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Table SRXB-A-35-9 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 12] at MOC

[[

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Table SRXB-A-35-10 VYNPS Four Bundle Powers and TIP String Comparisons Cycle 23 EOC

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Figure SRXB-A-35-3 VYNPS Cycle 23 TIP String Comparisons at EOC

[[

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Table SRXB-A-35-11 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 6] at EOC

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Table SRXB-A-35-12 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 17] at EOC

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VYNPS Cycle 24

Table SRXB-A-35-13 VYNPS Cycle 24 TIP Comparison Data as a Function of Cycle Exposure

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]]

Table SRXB-A-35-14 VYNPS Cycle 24 Summary Data

[[

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Table SRXB-A-35-15 VYNPS Four Bundle Powers & TIP String Comparisons at Cycle 24 BOC

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Figure SRXB-A-35-4 VYNPS Cycle 24 TIP String Comparisons at BOC

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Table SRXB-A-35-16 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 11] at BOC

[[

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Table SRXB-A-35-17 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 5] at BOC

[[

]]

Table SRXB-A-35-18 VYNPS Four Bundle Powers & TIP String Comparisons Cycle 24 MOC1

[[

]]

Figure SRXB-A-35-5 VYNPS Cycle 24 TIP String Comparisons at MOC1

[[

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Table SRXB-A-35-19 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument[String 11] at MOC1

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Table SRXB-A-35-20 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 8] at MOC1

[[

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Figure SRXB-A-35-21 VYNPS Four Bundle Powers & TIP String Comparisons Cycle 24 MOC2

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Figure SRXB-A-35-6 VYNPS Cycle 24 TIP String Comparisons at MOC2

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Table SRXB-A-35-22 VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 7] Cycle 24 MOC2

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Table 35-23 SRXB-A-VYNPS Four Bundle Nodal Power & Void Fraction Comparisons for Hot Channel Instrument [String 16] Cycle 24 MOC2

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RAI SRXB-A-36

The individual axial TIP data shows that the [[

]] Explain how it can be ascertained that the PCTIP peaking being closer to specific bundle power peaking is not due to variability in the positions of the TIPs which may reflect the response of specific bundles and not necessarily the maximum powered bundles. Provide an evaluation that establishes the reason that the measured PCTIP reflects the response of specific bundles.

Response to RAI SRXB-A-36

It is important to point out that what is measured in these comparisons is the gamma TIP response and not the axial power. The gamma TIP axial response is close to the axial power distribution, but it is not an exact match. The gamma TIP response is proportional to the gamma flux at the instrument location. Most of the gamma rays come from the fission process, but some come from capture gamma rays emitted when a neutron is captured in the Gadolinium poison, so the source of gamma rays is not directly proportional to the fission energy, and varies with bundle exposure. Also, the TIP response varies with void fraction, because the gamma rays interact with the electrons in the water. This variation of TIP response with void fraction and exposure is not large, but will cause the axial TIP response to be different that the axial power. The 3-D simulator contains a gamma tip response function derived from explicit Monte Carlo gamma transport calculations, providing an accurate estimate of the relationship between the instrument response and the bundle power density adjacent to the TIP string.

RAI SRXB-A-37

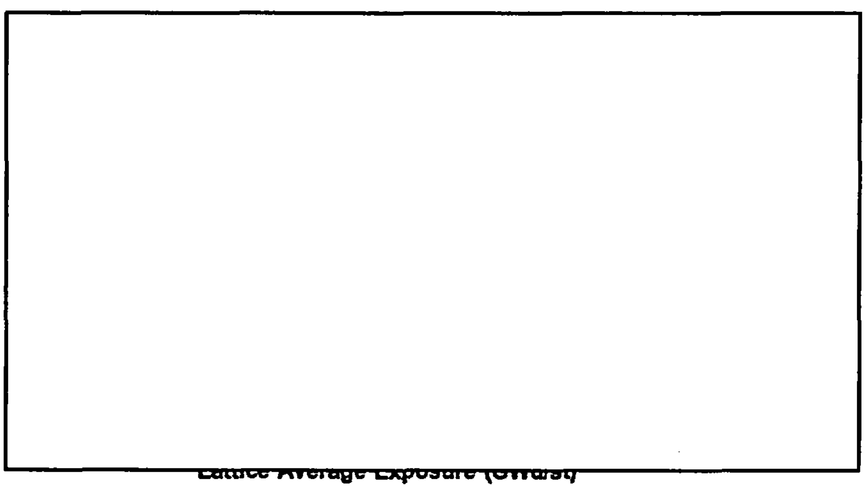
Section 3.1.1 Model Uncertainty of NEDC-32601P-A discusses the method used to determine the accuracy of TGBLA in computing the fuel pin peaking factors. The accuracy of the TGBLA model was established by comparing its peaking factor distributions with Monte Carlo (MCNP) benchmark results. Table 3-1 presents the RMS differences between the MCNP and TGBLA rod differences for 8x8, 9x9 and 10x10 lattice designs

]] Confirm that this is the case. Specifically, MCNP is not a depletion code and there are no biases assumed for MCNP calculations to demonstrate that the impact of the infinite pin power peaking with exposure is accounted for. Justify why it is acceptable to establish infinite lattice pin power peaking at BOL conditions to determine the uncertainties associated with pin power peaking as the bundles deplete under hard spectrums with fuel designs and loading that do not reflect the current core conditions.

Response to RAI SRXB-A-37

The comparisons presented in NEDE-32601P all came from TGBLA/MCNP comparisons performed at beginning of life. Since that time, a number of studies, using a "snapshot" approach have shown that the maximum TGBLA/MCNP differences occur at zero exposure. In the snapshot approach, the TGBLA model is used to evaluate the isotopic abundances for each pin in the lattice, which are transferred into the MCNP model. The pin power differences between TGBLA and MCNP are then compared for an isotopic snapshot for a given exposure. The results for a GE14 lattice design with a very high enrichment and Gadolinium content is shown in Figure 37-1. The snapshot process is conservative for evaluating RMS vs exposure because higher power pins will deplete faster than lower power pins resulting in lower U235 inventory, resulting in lower power for the next exposure. Hence power differences tend to even out with increasing exposure. This reduction in RMS with exposure was observed in the TGBLA/LANCER studies presented to the staff. The peak RMS at beginning of life conditions is not surprising because the presence of poison rods makes the transport calculation more difficult. Fewer poison rods simplify the TGBLA evaluations. Studies on low enrichment lattices with no poison rods show TGBLA/MCNP RMS values of less than [[ ]] for all void fractions.

Figure SRXB-A-37-1 Pin RMS vs Lattice Exposure

[[

]]

The average RMS obtained by the snapshot approach decreases with exposure. The 90% RMS has been included up to an exposure of 13 GWd/st. Design studies show that at these high void fractions, there is very little burnup because of the low powers. [[

]] The GEXL correlation relates the occurrence of dry out to the quality for a given flow condition. The quality is proportional to the axially integrated power. The R-factor relates the local quality near a given pin to the average quality over the entire lattice.

RAI SRXB-A-38

Table 3-1, "Summary of TGBLA/ MCNP Pin Power Comparisons," of NEDC-32601P-A combines the RMS differences in rod power using historical GE fuel designs (8x8, 8x8 GE9, 9x9 GE11, 10x10 GE12 and 10x10 SVEA96) [[

]] Justify why the uncertainty calculation should not be limited to fuel designs with similar characteristics and operated conditions, since pin power peaking is dependent on the fuel design, loading (enrichment, Gd loading, etc.) and operation (e.g., high void conditions).

Response to RAI SRXB-A-38

GE agrees that the infinite lattice peaking factor uncertainties should be based on up to date product lines (GE14) and designs. To this end, the TGBLA/MCNP snapshot burnup study was carried out on an aggressive application of the GE14 product. The lattice forming the basis for Figure SRXB-A-37-1 is a GE14 application with a lattice enrichment of [[ ]] Gadolinium concentration. As illustrated in the reply to RAI SRXB-A-37, the peak RMS [[ ]]. A series of TGBLA/MCNP comparisons have been carried out on GE14 lattices representative of EPU applications. The updated data set consists of 13 beginning of life lattices, and has an average pin power RMS of [[ ]]. The average value of [[ ]] is less than the [[ ]] uncertainty derived from an extensive and conservative review of pin-by-pin gamma scan data obtained by GE. This [[ ]] is referred to in the Alternate Approach.

RAI SRXB-A-39

[[

]] The fuel designs used to establish the gradient peaking [[ ]] is not discussed in the SLMCPR LTR.

a) State if similar [[ ]] analyses were performed for cores loaded with GE14 fuel and core designs that reflect the current and proposed operating conditions.

b) Demonstrate the validity of the currently used gradient peaking of [[ ]] for the current operating strategies and fuel designs (GE14). State if similar [[ ]] to establish the influence of neighboring bundles on the local peaking distribution.

c) State whether the core simulations calculations are performed at different exposures. If not, establish a methodology that accounts for the exposure dependence.

Response to RAI SRXB-A-39

a) One of the four DIF3D studies documented in NEDE-32601P-A (BWR/6 800 Bundle Equilibrium Core) was performed using the GE12 lattice configuration, which is identical to GE14. The average gradient peaking RMS obtained for this application was [[ ]], which is quite consistent with the RMS values obtained with the other cores. This application was based on an equilibrium design for a large BWR/6 with an 18 month cycle, consistent with current power uprate strategies.

b) Currently, a value of [[ ]] is used to represent the gradient uncertainty. As pointed out in part a), the 800 bundle BWR/6 example is quite representative of current applications. No studies have been carried out at void fractions larger than [[ ]]. The results in Table 3-2 of NEDE-32601P-A show [[

]]

c) The core simulations were performed on beginning of cycle configurations. A set of bundles with different exposures were used make up the core configuration. High and medium exposure lattices were placed adjacent to each other as one might see in a typical beginning of cycle core configuration. The BWR/6 core is shown below in Table SRXB-A-39-1.

Table SRXB-A-39-1 Quarter Core Map for DIF3D Benchmark

[[

]]

RAI SRXB-A-40

[[

]] State if the current plant-specific SLMCPR calculations [[ ]] uncertainty was included in the [[ ]].

Response to RAI SRXB-A-40

The [[ ]] uncertainty is included in the plant specific SLMCPR calculations and in the determination of additional margin discussed in the alternate approach section.

RAI SRXB-A-41

In the review of the SLMCPR methodology, the staff asked in RAI 5 the following, " The process computer monitors peak kw/ft and MAPLHGR. While MCPR depends primarily on the radial bundle power distribution, peak kw/ft and MAPLHGR depend on the bundle axial power distribution and, consequently are significantly more sensitive to the 3-D MONICORE replacement of the TIP/LPRM axial power distribution. Provide an uncertainty analysis for the 3-D MONICORE prediction of peak kw/ft and MAPLHGR." Update this RAI response. Justify why the peak Kw/ft uncertainty (see RAIs addressing changes in the peak and nodal uncertainties) would not change for the current operating strategies and fuel designs (GE14). Describe how the uncertainty is applied in 3D MONICORE.

Response to RAI SRXB-A-41

Table SRXB-A-41-1 is a summary of all the components included in the total pin LHGR uncertainty for the process computer. Listed are the values originally approved in NEDE-32601P-A and those proposed for application to EPU designs.

Table SRXB-A-41-1
Summary of Uncertainty Components for Process Computer LHGR Evaluations

[[

]]

The revised infinite lattice and power allocation (PAL) uncertainties are discussed in the alternate approach section. Since the publication of NEDE-32601, the [[ ]]. The PANAC11 version of the 3-D simulator contains an improved model for acting for the impacts of flux gradients on the pin power. The DIF3D studies described in RAI SRXB-A-39 show PANAC11Gradient effects uncertainty to be [[ ]]. This [[ ]] was representative of both the entire data set and the single 10x10 lattice core. The revised total uncertainty is [[ ]]. A local uncertainty of [[ ]] is assumed in the generation of the thermal mechanical limit values.

RAI SRXB-A-42

Section 3.2, "Conversion of Peaking Uncertainty to R-factor Uncertainty" of NEDC-32601P-A discusses that the R-factor represents the influence of the rod power peaking on the critical power. In addition, the R-factor methodology is described in NEDC-32505P, "An R-Factor Calculation Method for GE11, GE12, and GE13 Fuel," dated July 1999. The bundle R-factor is an input to the GEXL correlation. [[ ]] Tables 3.4a, b, and c of NEDC-32601-P-A provide the basis for the[[ ]]. Explain how the uncertainties in the lattice physics pin power data are accounted for in [[ ]] Explain how the void and exposure dependency of the uncertainties of pin peaking factors is incorporated in the R-factor methodology. Specifically, explain how the exposure dependence of pin powers uncertainties are established if the infinite lattice pin power uncertainties are established using MCNP/TGBLA code to code benchmarking based in BOL or the standard GE MCNP/TGBLA comparisons with exposure is used.

Response to RAI SRXB-A-42

Based on questions raised by the staff, the procedure for evaluating the R-factor uncertainty was modified. The modified procedure is documented in NEDE-32601-PQC and is the basis for the currently used R-factor uncertainty of [[ ]]. This approach takes into account the fact that the [[ ]] The lattice model uncertainties are based on the beginning of life MCNP/TGBLA comparisons based on the arguments presented in the response to RAI SRXB-A-37. [[ ]] The overall distribution of R-factor uncertainties is illustrated in NEDE-32601-PQC. This procedure is used to evaluate all new R-factor uncertainties based on updated model and channel bow uncertainties described in the alternate approach.

RAI SRXB-A-43

The subcooling increases for the EPU conditions in comparison with operation at rated power and lower domains. Justify why the uncertainty due to core inlet temperature (Section 2.5 NEDC-32601P-A) would not change.

Response to RAI SRXB-A-43

The basis of the core inlet temperature uncertainty is discussed in Section 2.5 of NEDC-32601P-A. In this discussion, it states that "It is concluded that the core inlet temperature uncertainty of [[ ]] specified in the (GETAB Analysis Basis) is adequately conservative to accommodate as much as a factor of [[ ]] stated in that reference. This level of conservatism is judged to be adequate to accommodate all plant specific variations."

RAI SRXB-A-44

State if VYNPS EPU conditions would result in any bypass voiding due to the high bundle power conditions. The LPRM uncertainty increases with increasing void, [[ ]] If any non-solid bypass voiding would occur during steady state, evaluate the LPRM and TIP uncertainties and justify why the current uncertainty based on zero bypass voiding remain applicable. Consider an increase in the random noise. If bypass voiding does occur during transient events (e.g., RPT) and plant maneuvers in the off-rated high power/low conditions, provide an evaluation of the impact of non-solid bypass voiding on the reliability and accuracy of the instrumentation.

Response to RAI SRXB-A-44

The design specification for limited local voiding in the bypass region at the uppermost LPRM detector for the highest power four-bundle configuration assures that the effects of voiding are insignificant for the limiting fuel operating conditions. The presence of bypass voids at the uppermost LPRM detector can result in a small decrease in measured power by thermal neutron detectors for that location due to reduced moderation of the neutron flux. However, since the power is divided among the four fuel assemblies and averaged with the larger measured power in the lower, un-voided fuel nodes, the total effect of the maximum voids was determined to be [[ ]] for the MCPR calculation.

Furthermore, it should be noted that VYNPS has installed gamma TIPS which will normalize the LPRM reading to the correct nodal power level since they are not sensitive to bypass voids, and this would essentially eliminate the impact of bypass voids for steady state operation.
The evaluation of the impact of non-solid bypass voiding on the reliability and accuracy of the instrumentation is based on a calculation of the MCPR overprediction by the process computer when the LPRM reading is not properly normalized to the correct power. The TIP system is used to both normalize the LPRM readings throughout the core, and establish the axial shape between the LPRM readings for the process computer. The TIP system is utilized periodically after core power and power distribution changes, and after LPRM sensitivity changes. [[

]]

The bypass voiding could increase the random noise at the LPRM due to passage of the bubbles around the LPRM detector as explained in the response to RAI SRXB-A-55. However, since the void level is less than 5%, the extra noise magnitude is small, and since this extra noise would add to the existing noise as the square root of the sum of the squares, the impact of the extra noise is negligible. Moreover, since the voiding is only at the D level, its effect on the bundle power measurement is insignificant.

There is no reliance on the process computer calculation of MCPR during transient conditions, and therefore no requirement exists for monitoring MCPR under local bypass voiding for conditions outside the normal steady state operating map (such as after a two pump trip). The impact of bypass voids after a two pump trip, and other effects of bypass voiding during transient conditions are addressed in the response to RAI SRXB-A-55.

RAI SRXB-A-45

Explain why the axial TIP RMS is not included in the calculation of the SLMCPR limit, when the SLMCPR limit calculation is non-adaptive. The staff understands that the core [[

]]. Therefore, explain how the axial power distribution is accounted for in the (offline) calculation that establishes the SLMCPR value.

Response to RAI SRXB-A-45

The purpose of the use of the 3-D simulator in the evaluation of the SLMCPR is twofold:

1. The SLMCPR methodology sums up contributions of the probability of boiling transition from every node in the core, and hence is affected by the relative power distribution. A flat power distribution results in a higher SLMCPR because more nodes are near the limiting node and contribute materially to the probability of boiling transition. The 3-D simulator is used to establish a limiting power distribution, usually one that is as flat as possible and then the power is increased until one of the bundles is on limits. This search for the limiting power distribution leads to possible operating distributions, and since the limiting bundle is arbitrarily placed on limits, has little to do with the uncertainty of the power in that bundle. The axial power is calculated to be consistent with the core state. For a given core state, the critical power is a very weak function of small changes in the axial power. Further, the RMS difference of all predictive to measured TIP response results in an average difference of less than [[ ]] for the reference BWR's, indicating that axial power distributions are also predicted adequately. Changes in TIP and LPRM signals affect the axial power and are modeled in the SLMCPR process.

2. The 3-D simulator is used to evaluate the impact of changes in operating conditions such as operating pressure, core flow, instrument response, pressure drop and a variety of other inputs. Since the simulator is used to evaluate changes relative to a base state, the absolute accuracy of the model does not contribute to the magnitude of the change.

Since the intended purpose of the 3-D simulator in the SLMCPR evaluation does not depend on the axial power accuracy and only to set the initial conditions and to evaluate changes, the predictive mode is used to set the initial power distribution.

RAI SRXB-A-46

Describe GE's standard approach for determining the cross-sections for operation at high void conditions (fit/interpolate/extrapolate). Discuss why the cross-sections are not directly generated for void fractions above 70%, using TGBLA. Describe any code-to-code comparisons that were performed in order to assess this method at high void fractions.

Response to RAI SRXB-A-46

The extrapolation accuracy beyond 70% voids has been discussed extensively in Section 2.1 of MFN-04-026. In these studies the extrapolated inputs to the 3-D simulator were compared to MCNP Monte Carlo values. Despite the fact that extended operation at 90% voids is rare, even for the highest power density plants, the extrapolation errors were shown to be small and acceptable for use in the 3-D simulator. Further, the application of the current 3-D simulator to high power density applications show little or no degradation in methods accuracy. Finally, the margin proposed in the Alternate Approach is designed to provide assurance that the accuracy and margin standards are upheld.

RAI SRXB-A-47

Provide plots of the isotopic concentrations and fission fractions of U-235, U-236, U-238, Pu-239, Pu-240, and Pu-241 as functions of burnup. Use lattices that are limiting in terms of enrichment and the number of hot pins and Gd concentration. Present the isotopic concentration vs. exposure plots for depletion at 40%, 70%, and 90% void conditions. The objectives are to baseline potential changes due to spectral hardening for operation at different void conditions and to determine both the accuracy of TGBLA and how TGBLA's accuracy changes with void fraction. Provide plots similar to the plots in Figures 3-8 and 3-9 of NEDE-20944-P for depletion at different void conditions and for different lattices.

Response to RAI SRXB-A-47

Plots of the atomic density of the identified isotope as a function of exposure and void fraction are provide in the attached figures. U-234 is included to identify that an initial fraction of U-234 is present in the initial isotopic inventory.

Five lattices designed for use in Vermont Yankee Cycle 25 are provided for use in observing the effect of void fraction on the transmutation of U-238 into higher atomic number actinide isotopes. The range of data provided is 0, 40, 70, and 90% void fractions and the exposure range from 0 to 65 GWd/st. The lattices provided are as follows:

Lattice Number Designation	Lattice Type	Lattice Nuclear Name
6996	Dominant	P10DNAL453-16G6.0-100T-T6-6996
6997	Dominant	P10DNAL453-12G6.0-100T-T6-6997
6999	Vanished	P10DNAL448-12G6.0-100T-V-T6-6999
7007	Dominant	P10DNAL413-14G6.0-100T-T6-7007
7009	Vanished	P10DNAL403-14G6.0-100T-V-T6-7009

For each lattice analyzed there are 7 plots that contain the isotopic inventory in the units of atom density for the required isotopes. The enrichment and gadolinium are given in Table 1 of each section. The plots of isotopic concentration in order of presentation are U-234, U-235, U-236, U-238, Pu-239, Pu-240, and Pu-241. A list figures and tables is provided below for plot and table identification.

47.1 Lattice P10DNAL453-16G6.0-100T-T6-6996 Isotopic Results

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47.2 Lattice P10DNAL453-12G6.0-100T-T6-6997 Isotopic Results

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47.3 Lattice P10DNAL448-12G6.0-100T-V-T6-6999 Isotopic Results

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47.4 Lattice P10DNAL413-14G6.0-100T-T6-7007 Isotopic Results

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47.5 Lattice P10DNAL403-14G6.0-100T-V-T6-7009 Isotopic Results

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RAI SRXB-A-48

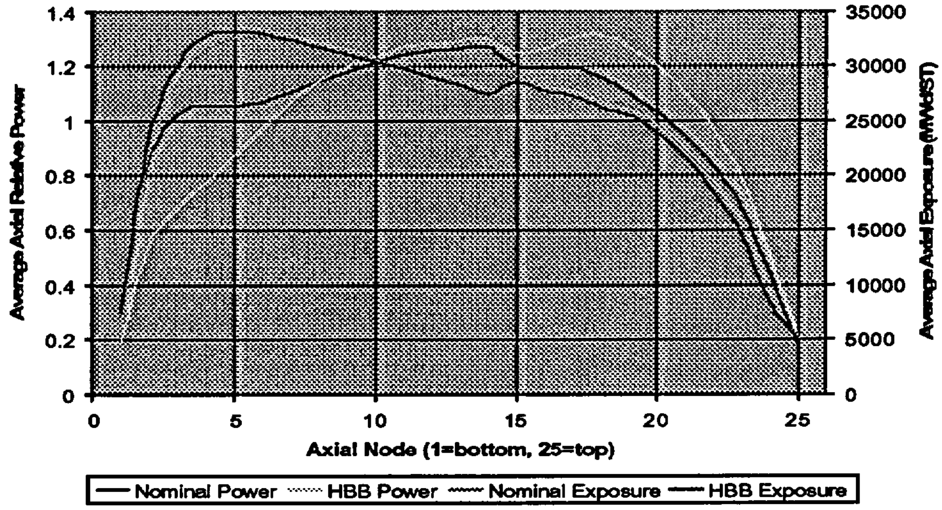
With respect to the accuracy of the steady state neutronic methods for current fuel design (GE14) and operating strategies, demonstrate that possible variations in isotopic content, under applicable EPU conditions, have been accounted for including consideration of void reactivity coefficient.

Response to RAI SRXB-A-48

In the response to RAI-25, RAI-27, and RAI-29 of MFN 05-029 TAC No. 5780, data have been presented which show that under power uprate conditions of 20%, the exit void fraction increases by about 3% on average. Figure 29-5 of RAI-29 shows that under high power uprate and long cycle conditions, exposure weighted void histories do not exceed 85% and those nodes with voids greater than 83% rarely exceed 20 GWD/st.

At this point in exposure, the lattice reactivity is changing and the void coefficient is dependent on the inventory of four major isotopes: Gd157, Gd155, U235, and PU239 (U238 is important, but the percentage change with exposure is small). The total change in void reactivity over this range from BOL to 20 GWD/st is about a factor of two based on a number of factors, including Gd depletion, Pu buildup and U235 depletion. Changes in depletion history can change the detail of the overall void coefficient behavior, but these effects are much smaller than the overall change. The major driver for transient response is not the effect of isotopics on void coefficient, but the influence of lattice depletion and operating strategy on the axial power distribution at the end of an operating cycle. For pressurization transients, which are the most severe transients for most BWR's, including VYNPS, the single most important parameter is the scram reactivity, which influences how effectively the transient is mitigated. Operating history can have a significant effect on scram reactivity, as shown in Figure SRXB-A-48-1.

Figure SRXB-A-48-1 Axial Power and Exposure



Note that the end of cycle HBB exposure shape is peaked to the bottom of the core relative to the nominal exposure and has a significant impact on the EOC power distribution. The HBB power shape is peaked to the top of the core. This power shape is conservatively used to determine the end of cycle transient response and is significantly different from the nominal power shape. As the control blades enter the bottom of the core, the scram strength is sensitive to the power level near the bottom. This difference in power shape is an important nuclear contributor to the transient response. It is influenced by the exposure and core isotopics, in that they play a role in the overall power shape. Their influence on void coefficient is secondary. For this reason, the HBB strategy has become the primary mechanism for ensuring conservative nuclear inputs to the transient evaluation process.

RAI SRXB-A-49

With respect to the GEXL correlation, for the VYNPS EPU conditions, state if any double-hump power shape is projected. Describe the methods and criteria used to determine that no additional SLMCPR penalty should apply. If any double hump power shape is predicted irrespective of the corresponding bundle power level, justify why it is acceptable to predict the MCPR performance of a bundle using a correlation that was developed with out the specific power shape, without adding a bias.

Response to RAI SRXB-A-49

The GEXL correlation for double humped axial power shape has been evaluated by comparison to data trends from GETAB (Ref. 1) and KTH (Ref. 2) and by benchmarking to detailed subchannel calculations. These comparisons all showed a consistent trend that the critical power was overpredicted by approximately [[ ]] for most GNF-A fuel products. These comparisons were then used to develop a bounding penalty at the [[ ]] confidence level to the bias and to the uncertainty of the GEXL correlation for the double humped axial power shape. When the SLMCPR is evaluated, the axial power shape is examined for all bundles in the core, and the above bias is applied to those bundles that have a double humped axial power shape. Note, the SLMCPR is determined on a cycle specific basis. This process was described in detail to the NRC at the 2002 technology update meeting (Ref. 3). There were no double humped axial power shapes for the VYNPS EPU cycle 25.

RAI SRXB-A-50

For the transient and LOCA events analyzed along the proposed operating domain statepoint, state if the power/flow ranges fall within the data base for which the GEXL correlation was developed. Give specific examples.

Response to RAI SRXB-A-50

The critical power correlation for GNF-A fuel is developed from critical power test data. The critical power data are obtained for bundle mass fluxes ranging from [[ ]], inlet subcoolings [[ ]] 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/MELLLA+ applications. As these data cover power levels up to the critical power for each flow, void fractions up to [[ ]] are included in the data. These parameter ranges also cover the expected ranges for LOCA and transient events. The process follows the NRC approved process as described in GESTAR II (Ref. 1). An example of this process is shown in Figure SRXB-A-50-1 that shows the GE14 application range together with the expected range for typical operational 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.

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RAI SRXB-A-51

Provide an evaluation that demonstrates that the void reactivity coefficients are applicable and were developed for the ranges of core thermal-hydraulic conditions expected for the transient and accident conditions, including ATWS and SBO.

Response to RAI SRXB-A-51

The Alternative Approach section summarizes the use of the 3-D simulator as the basis for all of the nuclear input to transient evaluations. The 3-D simulator has a bypass void model and the nuclear inputs are accounted for. The discussion in the answer to SRXB RAI-48 lists the reactivity response to axial power shapes changes and the physical reason that for all transients (including ATWS), the most important feedback from void reactivity comes from the power at the bottom of the core. The nuclear input is supplied on a nominal basis, whether it is supplied to the ODYN, TRACG, or stability evaluation tools. Conservatism is accounted for by operating assumptions, and in the case of the TRACG code, extensive statistical analysis base on variations of input parameters.

The VYNPS SBO analyses do not specifically model any particular transient or accident conditions with respect to core performance. Therefore, void reactivity coefficient changes do not influence the results.

RAI SRXB-A-52

Demonstrate that the pressure drop measurement database include GE14 fuel designs and the test were performed to cover the ranges expected for operation at EPU conditions. Include in your response, the maximum range of bundle power to flow ratios and bundle mass flux over which the data were taken. With respect to bundle delta pressure validation, the information provided to date indicates an underprediction in delta pressure for high bundle power to flow ratios. How are the delta pressure uncertainties applied?

Response to RAI SRXB-A-52

The range of mass flux and power to flow ratio for the GE14 pressure drop measurements are given in Section 2.2 Figures 2-41 and 2-42 of MFN 04-026. The pressure drop measurements cover from 30% of rated flow up to 120% of rated flow. The maximum mass flux in Figure 2-41 of 1.5 Mlb/hr/ft^2 corresponds to a bundle flow of 148,000 lb/hr and covers power to flow ratios up to 1.52(MW/(lb/hr))*10E-4, compared with a maximum of 0.9(MW/(lb/hr))*10E-4 observed in the compilation of bundle power to flow ratios (Figure 6-2 of BVY05-024). This maximum power to flow ratio is therefore 50% larger than expected in EPU applications. All predictions are within [[ ]] of the measured values, with a maximum of [[ ]] at high power to flow values. The standard deviation of all the errors is [[ ]]. An uncertainty of [ ]] is assumed in the SLMCPR analysis documented in NEDC-32601P-A. A change in channel pressure drop of [[ ]] has a negligibly small impact on the SLMCPR.

RAI SRXB-A-53

The document states that the ASEA-813 and ASEA-513 tests, which varied rod power distributions and side-to-side or in-out skew, were not included in the correlation development because of concerns over bias in the measurement. Enclosure 3 adds that there are many points of consistent overlap with ASEA-713 and the data serves well to validate the correlation developed using the most accurate data sources. Provide a tabulation or other means of justification that shows the data points supporting the validation of the correlation to the high void ranges. Identify the corresponding power profiles for the specific data set, the applicable test conditions /ranges (e.g., the bundle power/flow ratios.) and fuel designs and characteristics.

Response to RAI SRXB-A-53

Consideration of uncertainties in the void correlation and where they are actively considered in the GE licensing methodology are addressed in the response to RAI SRXB-A-54. Justification of the void-quality correlation in use by GE is contained in Attachment A (page A-39) of NEDC-32601P-A, "Methodology and Uncertainties for Safety Limit MCPR Evaluations". This attachment addresses sensitivity of the void fraction to bundle type, water rods, part length rods, spacers, and pin peaking.

RAI SRXB-A-54

An EPU or a high density plant can have an exit void fraction of [[

]] in the corresponding water density calculations?

Response to RAI SRXB-A-54

The void fraction prediction from the core simulator is best-estimate and further utilized for identification of appropriate cross sections for the state. The uncertainty in the void fraction impacts the flow and power distributions. As a result, any inaccuracy in the void fraction prediction is quantified as a part of the bundle & nodal TIP comparisons. (e.g. see the response to RAI SRXB-A-35). It is to be noted that such comparisons also include the direct effects of core instrumentation & measurement uncertainty. Therefore, the total uncertainties in these parameters and therefore also the effect of the void fraction uncertainty are accounted for in the SLMCPR.

VY EPU void fractions for realistic core designs (Cycle 24 and Cycle 25) are not expected to exceed 87% while the database for the void-quality correlation extends to 95% void fraction.

RAI SRXB-A-55

The VYNPS response to RAI SRXB-A-6 states that the review of the steady state calculations at natural circulation indicates that the [[

]]. The RAI response then justifies applying the [[

]] would not have significant impact on the thermal-hydraulic conditions and, therefore, the coupled reactivity feedback would not be affected.

As an Option 1D plant, VYNPS relies on APRM scram to provide SLMCPR protection for operation in the high powered/low flow zone of the power/flow map. For void fractions greater than [[ ]], issues that had not been addressed in the response are whether, (1) the SLMCPR would be protected before an ARPM scram occurs in order to meet GDC-12, and (2) would the reliability and effectiveness of the APRMs for stability protection be impacted by the high void conditions. Specifically, with 2RPT event conditions, [[

]], the instrument noise, and potentially the temperature, could increase, affecting the reliability of the APRMs relied upon for instability protection. Right after a 2RPT event, the core thermal-hydraulic conditions may result in an SLMCPR value that is higher than for the cycle-specific SLMCPR value. In addition, the coupled neutronic and thermal-hydraulic response at the cited void ranges may change depending on the response of the void reactivity coefficient with void fraction. Therefore, the RAI response is inadequate.

Provide an evaluation that considers all impacts discussed above, including the any increases in the uncertainties of the [[ ]]. For any conclusions or assessment made, please provide the supporting bases.

Response to RAI SRXB-A-55

The impact of high in-channel voids (such as those that could occur under 2 RPT conditions) on the methods and on SLMCPR determination, is described in the alternate approach. This response will focus on the impact of the high bypass voids on the protection provided by the APRM scram in mitigating stability events under 2 RPT conditions.

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 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 minor impact on steady state operation. These impacts are greatest for the highest elevation LPRM (D level) where the highest bypass voiding occurs, and are discussed quantitatively below.

1. Impact of Bypass Voids on LPRM and TIP

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The impact of bypass voids on the APRM can be estimated by determining the bypass voids [[ ]] at each of the [[ ]] 4 LPRM elevations (A, B, C and D) and each of [ ]] LPRM strings in that APRM, reducing the LPRM signals [[ ]] and averaging them, and then comparing this average to average of the LPRM signals R_{ij} without the reduction due to bypass voids.

2 Impact of High Bypass Voids on VY Option 1D Stability Solution

For VY Option 1D 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 scram occurs before stability oscillations get 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 APRM scram.

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3. Bases of the VY Option 1D Stability Setpoint Analytical Limit Validity

The detect and suppress evaluation for the VYNPS EPU is performed under the approved LTR basis (NEDO-32465-A, General Electric Company, "Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.). 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 Operating Limit MCPR and the Safety Limit MCPR not being exceeded during the instability oscillation.

The detect and suppress evaluation for Cycle 24 under EPU conditions has been re-performed to demonstrate the adequacy of safety margins should a thermal-hydraulic instability event develop at VYNPS.

The detect and suppress calculation assumes a flow runback along the rated licensing rodline to natural circulation flow. The APRM flow-biased flux scram setpoint analytical limit has been rescaled for EPU operation. The flow-biased APRM trip analytical limit at natural circulation is 53.7% of rated power. The best-estimate power level on the rated rodline at natural circulation is 49.4% of rated power. [[

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The discussion in item 2 of this response shows that [[

]] because of the conservatism in the stability evaluation, the margin to current setpoint analytical limit (AL) is adequate to include the effect of bypass voids, and that no downward adjustment of the setpoint is required.

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]] Hence, the SLMCPR is fully protected for VY EPU 2RPT runback conditions.

4. Impact on Noise due to Bypass Voiding

The increased voiding in the bypass region could potentially affect (increase) the LPRM noise because of the steam bubbles going by the LPRM instrument assembly in the water gap. The increase in noise depends upon the bubble dynamics as described below. Note that the discussion in this section refers only to the extra noise caused by the bubbles in the bypass region, and not the normal noise (~2% for APRM) that is present because of the flow induced vibration of the LPRM assembly in the water gap and because of other thermal-hydraulic phenomena inside the fuel channels. Note that there is an additional LPRM detector noise component due to the random nature of the process that produces neutron flux, which is proportional to the square-root of the neutron flux and is the source of the signal used for the IRM detectors. However the magnitude of that noise for LPRMs is small in the low frequency range of interest in this measurement, and does not need to be considered for this evaluation.

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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 since the decrease in normal noise at off-rated conditions is more than the increase due to bypass voiding.

5. Impact of LPRM Temperature Increase due to Bypass Voiding

There is no significant increase in detector temperature due to an increase in bypass flow voids. The basis for this conclusion is that bypass flow temperature external to the LPRM tubes is limited to a range from approximately 40 degrees Fahrenheit subcooled to saturation and that the heat transfer coefficient on the LPRM tube surface is not sensitive to the void content. Additionally, LPRM tubes for BWR/2 to BWR/5 plants have internal cooling flow that maintains even lower temperatures at the LPRM detectors and in internal tube component surfaces. Note that LPRM and TIP detectors are ionization chambers made of high temperature materials and are capable of operating reliably at higher temperatures. For example, the BWR-6 LPRM detectors operate in a dry tube, where the temperatures are higher.

References:

1. NEDO-32465-A, Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications, August 1996

2. OG01-0228-001, Additional information – Determination of Figure of Merit for Stability DIVOM Curve Applicability, July 16, 2001 (updated at NC and 45% flow, 1/16/2004)

RAI SRXB-A-56

Provide an evaluation of the impact on instrument random noise and reliability for operation at high power/low flow conditions during plant maneuvers and SLO operation.

Response to RAI SRXB-A-56

This discussion addresses the treatment of noise in the instrument readings for both high and low power and flow. The EPU includes a new operating region in the high power high flow operating conditions only. The operation below ~ 75% flow in terms of absolute power and flow, which includes single loop operation (SLO), is the same as pre-EPU MELLLA, so there are no new low flow noise effects with EPU for VYNPS.

The effect of process noise on both the power and flow instrumentation is accounted for as uncertainties in the calculation of the SLMCPR, and in the calculation of APRM setpoints and these are different for two and single loop operation. Also, for SLO there is an additional bias that needs to be considered in the drive flow to core flow relationship because of the effects of the reverse flow through the inactive jet pumps. However, this is not a noise source and is accounted for properly in the SLO APRM setpoint calculation. There is some additional uncertainty in power and flow measurements in SLO and that is accounted for in the licensing analysis through the use of approved uncertainties for the calculation of the SLMCPR (Reference 1) for both dual and single loop operation. These uncertainties are applicable to EPU operation because they reflect normal BWR noise and are given in terms of % of power.

Also, reactor power and flow noise during steady state operation is significantly reduced by BWR instrumentation because of the multiple measurements used both in power (multiple LPRM detectors) and flow (multiple jet pump DP transmitters). Additionally, noise filters are used in the instrumentation system to improve the measurement of average power and flow during steady state operation. The noise in the power signal is approximately proportional to the absolute power, but the noise at the APRM output at rated EPU power is approximately the same as at rated pre-EPU power because the APRM instruments read normalized power. The change in noise for EPU has no impact on the ability to perform normal plant maneuvers and surveillance tests. The effect of noise on instrumentation reliability during transient conditions is addressed in response to RAI SRXB-A- 55.

Reference:

1. Letter F. Akstulewicz (NRC) to G. A. Watford (GE), Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, "Methodology and Uncertainties for Safety Limit MCPR Evaluations"; NEDC-32694P, "Power Distribution Uncertainties for Safety Limit MCPR Evaluation"; and Amendment 25 to NEDE-24011-P-A on Cycle-Specific Safety Limit MCPR (TAC NOS. M97490, M99069 and M97491), March 11, 1999

RAI SRXB-A-57

In the response to RAI SRXB-A-6, the licensee states "[t]he reactivity events are analyzed with the steady state tools and the results presented regarding steady-state methods in this response are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events." This RAI response does not provide sufficient detail.

a) State the specific reactivity event being referred to (e.g., control rod drop accident, rod withdrawal error).

b) State what steady state methods evaluation were described in the response to the RAI SRXB-A-6 response. The [[ ]] would not serve to demonstrate that impact of local reactivity event on the fuel enthalpy and performance. Revise the RAI response and provide an explicit discussion of the event.

Response to RAI SRXB-A-57

The two transient reactivity events are the control rod withdrawal error (RWE) and the mislocated bundle error (MLE).

RWE

The RWE analysis is performed by first finding a limiting condition during the plant cycle with high rod worth conditions. The event represents the normal withdrawal of a control rod by the operator, however an error is made in the rod selection such that the operating fuel limits may be exceeded. The core simulator is then used to determine the change on MCPR for the condition with the maximum worth rod from a fully withdrawn condition. This MCPR, from the change in conditions at the limiting bundle plus a statistical adder to ensure it is bounding, is added to the safety limit MCPR (SLMCPR) to compare with the plant operating limit MCPR (OLMCPR) from other events to determine the most limiting value. In addition, the RWE analysis confirms compliance to the 1% plastic strain criteria. The type of hardware installed at the plant determines the type of rod withdrawal analysis that has to be performed. The RBM is not credited at VYNPS, so the highest worth rod is chosen from a fully withdrawn condition. The

BWR steady-state simulator is used for the analysis throughout the core cycle evaluation, determination of the highest worth rod and the MCPR analysis since this is a slow event where analyzing the neutronic and thermal-hydraulic conditions in equilibrium is appropriate. The conditions for this analysis lie within the range of the methods applications defined in the RAI SRXB-A-6 response. The following are the bases for the analysis:

1. The core is operating in a high worth nominal control rod pattern. The operating state is characterized by a control rod pattern, which is at high power and in an equilibrium xenon condition.

2. The RWE transient occurs over sufficient time to allow void re-distribution and heat transfer to reach equilibrium while leaving xenon concentrations and core inlet conditions unchanged.

3. The highest worth control rod when withdrawn will insert the maximum reactivity into the core and cause the greatest local bundle power response.

MLE

This is not specifically a transient event in that it is due to a bundle being mislocated in the core. The analysis is to determine the ?CPR for a bundle which is mislocated to a location, which would increase the power of such a bundle to a value greater than it would be in its planned location. Since core instrumentation does not monitor some fuel bundles, no analysis credit is taken for higher local power which may be detected by the instrumentation. The BWR simulator is applied for the fuel cycle and the specific bundle analysis. The selection process involves high reactivity worth bundles being placed in high power regions. The design limit calculated is the MCPR. The change in MCPR is added to the SLMCPR to determine the limiting value to compare with other plant transient events to provide the core operating limit MCPR.

RAI SRXB-A-58

For the transients, LOCA and ATWS, during the initial condition, the axial power peaking and distributions and the accuracy of the steady state neutronic methods affect the plants response. State whether uncertainties are applied to the axial nodal powers, and the calculated void fraction. If uncertainties are not applied, please justify.

Response to RAI SRXB-A-58

All power distributions input to the transient process are based on nominal 3D simulator calculations. As stated in the Alternate Approach section, the operating strategy and burn inputs are structured to obtain power distributions giving the maximum transient, or conservative response. The hard bottom burn assumptions ensure that the control blade scram reactivity response is minimized, yielding conservative results.

In Section 56 of NEDC-24154P-A, (Supplement 1-Volume 4), it is stated that the ODYN comparison to TRACG and TRAC-BF1 demonstrates that it qualifies as a best estimate code for ATWS analysis, which is conservative in most cases. It is also stated in this section that for ATWS applications, prior regulatory approval has been granted for best-estimate code application based on the low probability of the event, conservatisms in key inputs and the acceptance criteria.

The axial power profile [[

]] The average and hot bundle void profile is determined by SAFER at the initial and post-LOCA conditions. Uncertainties in predictions of void reactivity have no impact in the SAFER/GESTR methodology.

Entergy

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September 10, 2005

Docket No. 50-271
BVY 05-083
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 32
Extended Power Uprate – Additional Information

References:
1) 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

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

3) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The attachments to this letter provide supplemental information in response to requests for additional information from the NRC staff (Reference 2) and other supplemental information to update the application for a license amendment. As a result of a recent audit of certain analytical methodologies of General Electric (GE) that are used for the design and evaluation of VYNPS' fuel, the NRC staff identified the need for additional information reflected in several of the requests for additional information (RAIs) contained in Reference 2. Because of the recency of the requests, the attached is only a partial response to the Reference 2 RAIs; the remaining RAIs will be addressed in a submittal that will be made by September 16, 2005.

Reference 1 discussed the plant modifications necessary to support the extended power uprate (EPU) of VYNPS and a planned two-step power increase to 120% of currently licensed thermal power (CLTP). The two step process was necessary because EPU-enabling plant modifications were scheduled to occur during two refueling outages—Spring 2004 (RFO-24) and Fall 2005 (RFO-25). Modifications completed during RFO-24 support an approximate 15% increase in reactor thermal power, and modifications planned for RFO-25 support the achievement of the full power uprate to 1912 MWt. Because the modifications necessary to support full EPU will be completed during RFO-25, VYNPS will be able to implement the ascension to 120% CLTP in one step (subject to the limitations that may be imposed as part of power ascension testing). Upon startup from RFO-25 the plant modifications necessary to achieving a full power uprate to 1912 MWt will be complete.

In addition, to update the application, it should be noted that VYNPS will complete its transition to the GE14 fuel design during the upcoming RFO-25.

Attachments 1-3 concern regulatory commitments that have either been fulfilled, or will be during future RFOs. Attachment 4 provides an updated response to RAI SRXB-A-17 that was posed in Reference 3. Attachments 5-8 provide responses to RAIs in Reference 2.

Certain Reactor Systems Branch RAIs and responses thereto in Attachment 5 contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with the provisions of that regulation. Attachment 5 is considered to be Proprietary Information in its entirety. Attachment 6 is a non-proprietary version of Attachment 5. An affidavit provided by General Electric Company, supporting the proprietary nature of the document, is provided as Attachment 9.

There are two new regulatory commitments contained in this submittal associated with modifications to sampling probes in the condensate and feedwater systems, and future steam dryer inspections. The commitments are summarized in Attachment 10.

The following attachments are included in this submittal:

Attachment	Title
1	Steam Dryer Inspections
2	Feedwater Sample Probes
3	Motor-Operated Valve Program Commitment
4	Revised Response to RAI SRXB-A-17 Rod Withdrawal Error Transient
5	Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70 (Proprietary Information)
6	Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69, and 70 (Non-Proprietary Version)
7	Responses to RAIs EEIB-A-6 through EEIB-A-8
8	Responses to RAIs SPLB-A-30 and 31
9	GE Affidavit
10	New Regulatory Commitments

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings at your earliest convenience to resolve any remaining issues. If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 10, 2005.

Sincerely,

Norman L. Rademacher
Director, Nuclear Safety Assurance
Vermont Yankee Nuclear Power Station

Attachments (10)

cc: Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 B1
Washington, DC 20555

Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
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USNRC Resident Inspector (w/o attachments)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner (w/o Attachment 5)
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

BVY 05-083
Docket No. 50-271

Attachment 4

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Revised Response to RAI SRXB-A-17

Total number of pages in Attachment 4 (excluding this cover sheet) is 2.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT
VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides a revised response to the NRC Reactor Systems Branch's (SRXB) request for additional information (RAIs) in NRC's letter dated July 27, 2005.[1] Subsequent to making the response to RAI SRXB-A-17 in Entergy's letter of August 1, 2005,[2] discussions were held with the NRC staff that resulted in this revision

The individual RAI is re-stated as provided in NRC's letter of July 27, 2005.

RAI SRXB-A-17

In Supplement 4, Attachment 5, Matrix 8, page 13, note for SE Section 2.8.5.4.1, there is an explanation for uncontrolled control rod withdrawal from a subcritical or low power startup condition. In this explanatory section, this event is considered as an accident and a fuel enthalpy of 170 calories/gram is given as the acceptance criterion. However, in SRP Section 15.4.1, this event is considered as a transient, not as an accident, and hence specified acceptable fuel design limit criteria is applied. Why is this event considered as an accident rather than a transient?

Response to RAI SRXB-A-17

(The following response supersedes the response to RAI SRXB-A-17 that was provided in license amendment request, Supplement 30, Entergy's letter of August 1, 2005, BVY 05-072)

Consistent with the SRP, this event is indeed considered a transient event, not an accident.

The transient thermal limits are established such that no fuel damage is to occur during the most severe abnormal operating transient. Fuel damage is defined as perforation of the cladding that permits release of fission products. Fuel damage can occur due to two primary mechanisms: (1) severe overheating of the fuel cladding caused by inadequate cooling, and (2) fracture of the fuel cladding due to stresses which may be induced by the relative expansion of the fuel pellet inside the cladding.

To achieve severe overheating of the cladding due to inadequate cooling, it would be necessary to generate more thermal power (heat) in the fuel than can be adequately transferred through the cladding to the coolant. Transients that can cause this type of behavior, typically occur

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[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," July 27, 2005

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 30, Extended Power Uprate – Response to Request for Additional Information," BVY 05-072, August 1, 2005

during higher power operation. Operation within the Operating Limit Maximum Critical Power Ratio (OLMCPR) protects against this.

At lower power, rapid fission gas generation and pellet expansion induced cladding stresses are a concern. In order to protect against events of this type, including the Continuous Rod Withdrawal during Startup transient, a criterion was developed that limited peak fuel enthalpy below the cladding stress failure limit.

For the Continuous Rod Withdrawal during Reactor Startup transient, NEDO-23842[3] establishes a peak fuel enthalpy licensing basis criterion of 170 cal/gm that shall not be exceeded. This criterion was adopted from NEDO 10527,[4] which states that this value is the fuel cladding failure threshold. This criterion is widely used by operating BWRs, and its use has been accepted by NRC. In fact, NUREG 1433[5] Section B3.3.1.1 states "to demonstrate the capability of the IRM System to mitigate control rod withdrawal events, generic analyses have been performed (Ref. 4) to evaluate the consequences of control rod withdrawal events during startup that are mitigated only by the IRM." The "(Ref. 4)" from this section of NUREG 1433 is NEDO-23842.

VYNPS Updated Final Safety Analysis Report[6] (UFSAR) Section 14.5.3.2, "Continuous Rod Withdrawal during Reactor Startup," states that the peak fuel enthalpies resulting from this event are less than 60 cal/gm, which is significantly less that the licensing basis limit of 170 cal/gm. As such, this is VYNPS' current licensing basis for this event, and it is not being changed for EPU. Because this event is considered a non-limiting transient, it is not required to be analyzed for EPU per NEDO-33004-A,[7] as approved by the NRC in a safety evaluation dated March 31, 2003. However, VYNPS did perform an evaluation of the Continuous Rod Withdrawal during Reactor Startup transient for EPU.

For EPU by itself, peak fuel enthalpy is not expected to increase. However, indirectly, EPU fuel and core designs may lead to higher rod worth and, therefore, higher peak fuel enthalpy at low power. It was conservatively assumed that a 20% increase in rated power would increase peak fuel enthalpy at low power by 20%, resulting in a peak fuel enthalpy for the Continuous Rod Withdrawal during Reactor Startup of 72 cal/gm, still far below the peak fuel enthalpy limit of 170 cal/gm.

[3] NEDO-23842, R.C. Stirn & J.F. Klapproth, "Continuous Control Rod Withdrawal Transient in the Startup Range," April, 18, 1978

[4] NEDO-10527, C.J. Paone, R.C. Stirn, & J.A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972

[5] NUREG-1433, Revision 3.0, "Standard Technical Specifications General Electric Plants, BWR/4," June 2004

[6] Updated Final Safety Analysis Report (UFSAR), Vermont Yankee Nuclear Power Station, Revision 19

[7] NEDO-33004-A, Revision 4, "Licensing Topical Report, Constant Pressure Power Uprate," July 2003

BVY 05-083
Docket No. 50-271

Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 32

Extended Power Uprate – Additional Information

Responses to RAIs SRXB-A-59, 60, 61, 62, 63, 64, 66, 69 and 70

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Total number of pages in Attachment 6 (excluding this cover sheet) is 54.

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides responses to the NRC Reactor Systems Branch's (SRXB) individual requests for additional information (RAIs) in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. In certain instances the intent of individual RAIs may have been modified based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The individual RAIs are re-stated as provided in NRC's letter of September 7, 2005.

RAI SRXB-A-59

The response to RAI SRXB-A-8 in Supplement 30, Attachment 9, is not clear regarding whether single loop operation of shutdown cooling (SDC) is assumed as part of the VYNPS Appendix R analysis. If single loop operation is assumed, has an evaluation been performed at the proposed EPU conditions to demonstrate that VYNPS can achieve cold shutdown, within the required time, with only a single SDC loop during an Appendix R fire event?

Response to RAI SRXB-A-59

Single loop operation of RHR shutdown cooling (SDC) is assumed for decay heat removal as part of the VYNPS Appendix R analysis in order to achieve cold shutdown within the time required by Appendix R (i.e., 72 hours). An underlying assumption in the Appendix R analysis is that one loop of RHR is unavailable due to the postulated event.

Section 3.10.1 of the VYNPS Power Uprate Safety Analysis Report (PUSAR) discusses the SDC analysis for constant pressure power uprate (CPPU).

It should be emphasized that the design criterion cited was based on using both RHR heat exchangers, and the requirement to cool the reactor vessel from approximately 327° F (saturation temperature at 100 psig) to 125° F, which takes approximately 11 hours. This analysis is based on 85°F cooling water which provides only a 40° F ΔT thermal driving force with the reactor at 125°F.

The VYNPS Technical Specifications define cold shutdown as having a reactor coolant temperature of less than or equal to 212° F. When the reactor coolant temperature is at 212° F,

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

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there is a 127° F ΔT, which is approximately three times the thermal driving force as the two heat exchangers case. For the Appendix R scenario, the increased thermal driving force more than compensates for the assumed loss of one heat exchanger. Thus, the time required to achieve cold shutdown (i.e., 212°F) under the Appendix R scenario conditions is less than 24 hours.

Because of the much larger temperature difference between the assumed service water temperature (i.e., 85°F) and reactor coolant during hot shutdown conditions, heat exchanger performance is more effective; thus, the rate of cooldown is increased, and cold shutdown is achieved well within the 72-hour requirement assuming the operation of a single loop of RHR SDC. Thus, significant margin exits to achieve cold shutdown within 72 hours.

RAI SRXB-A-60

Clarify the distinction between the terms "equilibrium core," in the response to RAI SRXB-A-10, "representative cycle core" in Section 2.2 of the VYNPS Power Uprate Safety Analysis Report (PUSAR) (i.e., Attachment 4 of the application dated September 10, 2003), and "power uprate representative equilibrium cycle core design" in the response to RAI SRXB-A-9.

Response to RAI SRXB-A-60

The three terms "equilibrium core," "representative cycle core," and "power uprate representative equilibrium cycle (PUREC) core design" are synonymous.

RAI SRXB-A-61

The response to RAI SRXB-A-11 in Supplement 30, Attachment 9, states that the current licensing basis requirements for new or spent fuel storage are not being changed by the proposed EPU. However, the response does not address whether any analysis was performed regarding the affect of the proposed EPU on new and spent fuel storage. Please address whether this analysis was done and, if so, the results of the analysis. The response should address the affects of enrichments levels in new fuel, and potential increase of some elements/isotopes (such as plutonium) in spent fuels, etc.

Response to RAI SRXB-A-61

VYNPS has Technical Specification requirements that limit the effective multiplication factor, Keff, of the spent fuel pool (SFP) to less than or equal to 0.95 and to ensure that the infinite multiplication factor, Kinf, of any segment of fuel assembly stored in the SFP is less than 1.31 at 20°C.

Analysis has been performed that shows that ensuring the Kinf of any fuel segment is less than 1.31 will ensure that the SFP Keff remains below 0.95. For each reload, the fuel vendor, currently Global Nuclear Fuel (GNF), calculates the Kinf at 20°C for each different fuel lattice

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type to be utilized, as a function of void history and lattice exposure. These calculations address the change in elements/isotopes including plutonium. Results indicate that the peak Kinf occurs at zero void fraction due to quicker gadolinia burn out. Near the end of bundle life, Kinf is higher for bundles burned at higher void fractions than those bundles burned at lower void fractions. However, this Kinf is significantly less than the peak Kinf for bundles burned at zero void fraction. VYNPS ensures that the peak Kinf is less than 1.31 for all fuel lattice types used in the reload.

VYNPS has a Technical Specification requirement to limit the effective multiplication factor, Keff, of the new fuel storage facility to less than 0.90 when dry and 0.95 when flooded. The new fuel storage vault will not be used until a criticality analysis is completed that considers fire fighting foam entering the vault.

RAI SRXB-A-62

The proposed changes to Technical Specification (TS) 3.4.C.3 are shown on page 8 of Attachment 1 to the application dated September 10, 2003. This TS includes a mathematical expression showing the relationship between standby liquid control (SLC) system pump flow rate, boron concentration, and boron enrichment that is required to demonstrate SLC system operability consistent with the requirements in 10 CFR 50.62(c)(4). Additional information is required to demonstrate that the proposed value of 1.29 in this mathematical expression is acceptable at EPU conditions.

Response to RAI SRXB-A-62

The equivalency equation in TS 3.4.C.3 conforms to the SLC system requirements of 10CFR50.62(c)(4) for anticipated transients without scram (ATWS). The EPU ATWS analysis also provides assurance that various VYNPS reactor and containment parameter acceptance criteria are met. The EPU analysis was performed using the following SLC system nominal values:

- flow rate of 40.5 gpm,
- boron concentration of 10.42 wt%, and
- boron-10 enrichment of 43%

When these values are combined with the mass ratio (628,300 lbs./401,247 lbs.), the result is slightly less than 1.29. To ensure that the EPU ATWS analysis remained bounding, the equivalency equation was modified to require meeting the more stringent value of 1.29 rather than the value of 1.

A review of the proposed change to TS 3.4.A.3 indicates that use of symbols for the subject four factor expression could be clarified. The combined use of an equal (=) sign and a greater than or equal (≥) sign for "Q" should be changed to a single greater than or equal (≥) sign. The two TS replacement pages provided at the end of this Attachment are a revised markup of the

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TS and a re-typed page. These pages should be substituted for those provided in the original application of September 10, 2003.

RAI SRXB-A-63

Section 2.8.5 of the safety evaluation template in Review Standard RS-001 directs the NRC staff to evaluate the licensee's accident and transient analyses to determine if the analyses adequately account for operation of the plant at the proposed EPU power level. Please describe the transients that are analyzed at the current licensed power level for determination of the operating limit minimum critical power ratio and discuss which transient is most limiting. In addition, please confirm that the seven transients listed in Section 9.1 of the NRC staffs safety evaluation dated March 31, 2003, for General Electric (GE) licensing topical report NEDC-33004P, "Constant Pressure Power Uprate," will be analyzed for the first EPU core.

Response to RAI SRXB-A-63

The transients that are analyzed at the current licensed power level for determination of the Vermont Yankee Nuclear Power Station (VYNPS) operating limit minimum critical power ratio (OLMCPR) are as follows:

[[

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In addition, the [[

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The above transient selection is consistent with *General Electric Standard Application for Reactor Fuel (GESTAR)*, NEDE-24011-P-A-14, June 2000; and the U.S. Supplement, NEDE-24011-P-A-14-US, June 2000. The above transients are analyzed for each VYNPS reload.

For VYNPS current operating cycle at Current Licensed Thermal Power, the limiting transient for determination of [[ ]]

Section 9.1 of the NRC safety evaluation dated March 31, 2003, for GE licensing topical report NEDC-33004P, "Constant Pressure Power Uprate," lists the following transients that will be re-analyzed at Extended Power Uprate [[

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The above transients listed in Section 9.1 of the NRC safety evaluation dated March 31, 2003, for GE licensing topical report NEDC-33004P, "Constant Pressure Power Uprate," will be analyzed for the first VYNPS EPU core. [[

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RAI SRXB-A-64

Provide the values for maximum bundle power and average power densities at VYNPS before and after the EPU.

Response to RAI SRXB-A-64

The maximum allowable bundle power is determined by the VYNPS thermal limits that may vary from cycle to cycle. The values for maximum bundle power before and after EPU are 7.02 MWt and 7.37 MWt, respectively. This represents a 5% increase in maximum bundle power for a 20% increase in rated thermal power. The values for average power density before and after EPU are 48.9 kW/liter and 58.7 kW/liter, respectively.

RAI SRXB-A-66

CASMO/TGBLA04 Code-to-Code Comparisons

In the June 30, 2005, meeting with Entergy, the NRC staff discussed with the licensee the need for code-to-code comparisons to confirm GENE's lattice physics code capability with depletion. Currently, GE uses MCNP to perform the code-to-code comparisons without coupling MCNP calculations with an independent depletion code. Therefore, the uncertainties and the biases of TGBLA are established using MCNP with isotopic concentration from TGBLA to account for depletion effects. This approach provides the inherent bias and uncertainties of the TGBLA methods and data assuming the isotopics concentrations and excluding the effects of errors in the depletion calculations. Therefore, the uncertainties are developed using TGBLA/MNCP comparisons. Considering the lack of measurement data for the current fuel design as operated, Entergy is in a position to perform lattice physics code-to-code benchmarking using CASMO4. From the July 12, 2005, telephone call, the NRC staff understood that Entergy was going to perform code-to-code lattice physics data comparisons. However, the licensee failed to provide the CASMO4/TGBLA lattice physics data code-to-code comparison. Core follow thermal limits comparisons of TGBLA/PANACEA and CASMO4/SIMULATE-3/JAFCPR 2.1 were

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provided. The staff finds the thermal limits comparisons useful; however, the main task at hand should have been to provide the independent code-to-code benchmarking of the standard GE TGBLA lattice physics method. Specifically, a code-to-code method would provide a means to evaluate the errors associated with the standard GE fit/extrapolation method.

a) As originally stated, provide code-to-code comparisons for some of the limiting lattices in terms of bundle powers, enrichment and gadolinium loading. Provide plots of the lattice code-to-code cross-section and pin power peaking and isotopic inventory comparisons. Provide plots comparing the same neutronic parameters as those included in MFN 04-026, Enclosure 3. Perform these comparisons on a lattice basis. Alternatively, state why CASMO/TGBLA code-to-code comparisons were not, or cannot, be provided. Note that errors in the cross-sections affect the predicted bundles powers, the nodal (bundle wise) axial power peaking and profiles and the changes in the core reactivity with change in the voids during anticipated operational occurrences (AOOs) and accident conditions. While it is difficult to reconcile differences in the cross-sections (e.g., flux ratios) between two independent depletion codes, the differences and trending are useful in evaluating the capability of the code being assessed. In particular, if the independent code predictions are supported by comparisons to measured data (bundle and pin gamma scans) based on current fuel designs and operated at the current conditions, then such comparisons are valuable as an interim process. The reason for seeking the CASMO-4/TGBLA comparisons is that MCNP is not a depletion code.

b) Provide additional information on the uncertainties applied in the CASMO4/SIMULATE-3/JAFCPR2.1 calculations. State if the Simulate-3 uncertainties are based on LPRMs or TIP-based uncertainties.

Response to RAI SRXB-A-66

Response to Part (a)

As discussed during the NRC staff's audit of GE Methods on September 7, 2005, the two codes (CASMO-4 and TGBLA-6) use fundamentally different methodologies to calculate core parameters, including cross sections. One fundamental difference between the two codes is that each performs calculations using different neutron energy groups. Consequently, it is difficult to generate lattice cross sections that provide for meaningful comparisons. Therefore, those comparisons are not provided. However, comparisons of other parameters for five lattices designed for use in VYNPS Cycle 25, and identical to those presented in Supplement 30,[2] are provided for code comparison purposes. These lattice calculations were performed with what are understood to be identical inputs (temperatures, dimensions, etc.) within the known allowances of the methods. Because some of the comparisons include high void (90%) cases, the TGBLA-6 results are from the non-production version used to address the Pu-240

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 30, Extended Power Uprate – Response to Request for Additional Information," BVY 05-072, August 1, 2005

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resonance treatment (to be discussed in the future response to RAI SRXB-A-67, part (e)). The range of data provided is 0, 40, 70, and 90% void fractions and the exposure range from 0 to 65 GWd/st. The lattices provided are as follows:

Lattice Number Designation	Lattice Type	Lattice Nuclear Name
6996	Dominant	P10DNAL453-16G6.0-100T-T6-6996
6997	Dominant	P10DNAL453-12G6.0-100T-T6-6997
6999	Vanished	P10DNAL448-12G6.0-100T-V-T6-6999
7007	Dominant	P10DNAL413-14G6.0-100T-T6-7007
7009	Vanished	P10DNAL403-14G6.0-100T-V-T6-7009

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For each lattice analyzed there are plots showing lattice K-infinity, local peaking and Pu-239 and 240 atom densities as calculated by the two codes. In addition, per audit request, Pu-241 atom density comparisons are also included. A list of these figures is provided below.

As shown in these figures, K-infinity performance is generally as expected with slight differences at low exposure due to gadolinium (Gd) burnout modeling differences. After Gd burnout, agreement between the two methods is good over the range of exposures with the exception of the 90% void cases which will be further discussed in the future response to RAI SRXB-A-67, part (e). While the lattice K-infinity differences are larger at this higher void, this lattice reactivity difference would have little core-wide effect due to both the small fraction of the core at those conditions and the limited power produced in those regions. This is demonstrated by the good

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comparisons of predicted (PANAC11 and SIMULATE) and measured axial powers in the core follow data provided previously.

For the lower void cases, local peaking, which is a comparison of the peak pin in the lattice, agrees well over the lower exposure range where lattices are generally limiting (high power). The two methods tend to deviate at higher exposure, non-limiting conditions. As in the case of K-infinity, the high void (90%) cases exhibit peaking differences earlier in exposure but, due to the little power produced by high void nodes, these differences are not considered significant to safety.

Isotopic comparisons between the two methods agree well and are considered to be within the ranges seen in similar methods comparisons (ref. ORNL-6901[3]). As is the case for the other parameter comparisons, the 90% void cases exhibit the greatest differences and, due to the little power and exposure occurring under those conditions, the differences are not considered significant to safety.

Additional Audit Question Responses:

Based upon the NRC staff's questions during the September 7, 2005, audit of GE Methods, the following additional information is provided:

Figure SRXB-A-66.6-1	Lattice 6696 K-infinity Fit Comparisons
Figure SRXB-A-66.6-2	Lattice 6697 K-infinity Fit Comparisons
Figure SRXB-A-66.6-3	Lattice 6999 K-infinity Fit Comparisons
Figure SRXB-A-66.6-4	Lattice 7007 K-infinity Fit Comparisons
Figure SRXB-A-66.6-5	Lattice 7009 K-infinity Fit Comparisons

These figures demonstrate the agreement between K-infinity as a function of exposure when calculated at 90% void and when extrapolated to 90% from a fit of 0%, 40%, and 70% cases. Both CASMO-4 and TGBLA-6 results are shown. These data indicate that the two methods are self consistent, i.e., data fitting and extrapolation in void is a reasonably accurate substitution for specific depletion calculations.

Figure SRXB-A-66.7-1 Lattice 7009 Void Coefficient Comparisons

This figure depicts the results of lattice void coefficient calculations (CASMO-4 and TGBLA-6) for a lattice located near the core exit which is a region of higher void. As noted in discussion with the NRC staff, the standard practice by GNF is to perform all instantaneous void cases from a 40% void history case. Therefore, CASMO-4 and TGBLA-6 comparisons of that practice

[3] ORNL6901, "OECD/NEA Burnup Credit Calculational Criticality Benchmark Phase I-B Results", June 1996

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are shown and indicate very good agreement between the two methods. In addition, CASMO-4 and TGBLA-6 results for equivalent calculations from the 70% void history case are shown for information. While the results show a difference in the void coefficient obtained from the two methods, the impact of those differences has been discussed with the staff and should not be taken out of the context of ultimate application; i.e., reactivity feedback is a function of both void coefficient, change in void, and local power (flux) such that most of the reactivity void feedback in transients occurs in lower void initial condition regions (nearer 40%).

Figure SRXB-A-66.8-1 RMS of Lattice 7009 Pin Power Differences

This figure depicts the results of a statistical evaluation of the differences in relative pin powers calculated by CASMO-4 and TGBLA-6 for a representative lattice for the 0, 40, 70, and 90% void history depletions. Due to differences in depletion steps (metric tons and short tons), only approximate exposure comparisons can be made and the data reflect the limited number of points. However, the general trends are evident and examination of the underlying data indicates that lower powered pins drive the differences while higher powered pins generally agree well. The peak or leading pin comparisons are most relevant in assessing fuel performance and safety and those are provided elsewhere in this response.

Tabulated data (EXCEL spreadsheets) of all this information will be transmitted under separate cover in a future submittal.

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Figure SRXB-A-66.1-1
Lattice 6996 K-infinity Comparison

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Figure SRXB-A-66.1-2
Lattice 6996 Local Peaking Comparison

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Figure SRXB-A-66.1-3
Lattice 6996 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.1-4
Lattice 6996 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.1-5
Lattice 6996 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-1
Lattice 6997 K-infinity Comparison

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Figure SRXB-A-66.2-2
Lattice 6997 Local Peaking Comparison

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Figure SRXB-A-66.2-3
Lattice 6997 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-4
Lattice 6997 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.2-5
Lattice 6997 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-1
Lattice 6999 K-Infinity Comparison

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Figure SRXB-A-66.3-2
Lattice 6999 Local Peaking Comparison

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Figure SRXB-A-66.3-3
Lattice 6999 Pu239 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-4
Lattice 6999 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.3-5
Lattice 6999 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.4-1
Lattice 7007 K-Infinity Comparison

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Figure SRXB-A-66.4-2
Lattice 7007 Local Peaking Comparison

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Figure SRXB-A-66.4-3
Lattice 7007 Pu239 Isotopic Concentration Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.4-4
Lattice 7007 Pu240 Isotopic Concentration Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.4-5
Lattice 7007 Pu241 Isotopic Concentration Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.5-1
Lattice 7009 K-infinity Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.5-2
Lattice 7009 Local Peaking Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.5-3
Lattice 7009 Pu239 Isotopic Concentration Comparison

NON-PROPRIETARY VERSION

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Figure SRXB-A-66.5-4
Lattice 7009 Pu240 Isotopic Concentration Comparison

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Figure SRXB-A-66.5-5
Lattice 7009 Pu241 Isotopic Concentration Comparison

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Figure SRXB-A-66.6-1
Lattice 6696 K-Infinity Fit Comparisons

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Figure SRXB-A-66.6-2
Lattice 6697 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-3
Lattice 6999 K-infinity Fit Comparisons

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Figure SRXB-A-66.6-4
Lattice 7007 K-Infinity Fit Comparisons

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Figure SRXB-A-66.6-5
Lattice 7009 K-infinity Fit Comparisons

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Figure SRXB-A-66.7-1
Lattice 7009 Void Coefficient Comparisons

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Figure SRXB-A-66.8-1
RMS of Lattice 7009 Pin Power Differences

NON-PROPRIETARY VERSION

Response to Part (b)

The CASMO-4/SIMULATE-3 data presented in the alternate approach response to RAI SRXB-A-6 included MFLCPR values calculated using Entergy's own in-house code, JAFCPR2.1. These data were intended to demonstrate the type of agreement seen between GNF and the independent methods used by Entergy. JAFCPR2.1 is a utility code that accurately reproduces the GEXL correlation results and, in the case shown, uses power distribution and core parameters fed to it from SIMULATE-3. Since this information is used for independent monitoring and verification, it is applied in a best estimate manner, without applying any uncertainties.

RAI SRXB-A-69

Void Fraction Uncertainties

RAI SRXB-A-54 asked the following, "An EPU or a high density plant can have an exit void fraction of [[
]] Do these void fraction predictions include the [[ ]] uncertainties in the corresponding water density calculations?"

The RAI response stated that the uncertainty in the void fraction impacts the flow and power distributions. The response states that an uncertainty is not added to the void fraction because the core follow TIP comparisons would have indicated any inaccuracies in the void fraction calculations. This RAI response did not provide sufficient justification. As discussed in response to RAI SRXB-A-36, the TIP response has many contributors and the core follow data does not provide the level of accuracy required to account for under-prediction in the nodal void fractions. In addition, the predicted void fraction is used in the offline safety analyses. The following requests address the basis for assuming no uncertainty in the void fraction calculation.

a) State if the void fraction calculations were benchmarked against measured data for all of codes that predict the void fractions and are used in the safety analyses, supporting the VYNPS EPU (e.g., PANACEA/ODYN/ISCOR/TASC). Demonstrate that the void fraction errors are insignificant or discuss the void fraction uncertainties assumed in the applicable codes. Justify why the current uncertainty is acceptable and applicable for the ranges to which it is being applied.

b) The core monitoring system was never reviewed and approved by the NRC. However, many of the RAI responses seem to qualify the impact of the higher void conditions on VYNPS by stating that the void fraction would be limited to specific value. However, no uncertainties were assumed in the predicted void fraction. If no void fraction measurement validation is available, then apply the [[ ]] uncertainty until such data can be used to demonstrate the accuracy of the prediction of the void fraction.

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Response to RAI SRXB-A-69

Response to Part (a)

The GE design correlation (Ref. 69-1) for void fraction was derived in the seventies as an extension of the drift-flux model (Ref 69-2), based on void fraction measurements in simple geometries as well as full scale bundle data covering a wide range of conditions (See Table SRXB-A-69-1).

The measurement uncertainty in the multi-rod data is [[ ]]. The void correlation fits these data with an average error of [[ ]] and a standard deviation of [[ ]]. No trend is observed with bundle size or geometry (See Table SRXB-A-69-2). In addition to the multi-rod data, the void correlation has been qualified to simple geometry data covering a wide range of conditions (Ref. 69-1).

The void correlation is correlated as a function of Reynolds number, quality and fluid properties. Since the Reynolds number is a function of mass flux, hydraulic diameter and fluid properties, and the fluid properties are a function of pressure, the void correlation can also be correlated as a function of hydraulic diameter, mass flux, quality and pressure. The range in hydraulic diameters in the data is [[ ]], which is much larger than the range of hydraulic diameters in the fuel designs. The hydraulic diameter in recent GE fuel products ranges from [[ ]] for 8X8 fuel to [[ ]] in the fully rodded region of 10X10 fuel. In the region above the part length rods, the hydraulic diameters range from [[ ]] for 10X10 fuel to [[ ]] for 9X9 fuel. The pressure range covers atmospheric pressure to twice normal operating pressure for a BWR. The mass flux in a BWR ranges from approximately 400 kg/m^2-sec at natural circulation to approximately 1350 kg/m^2-sec at rated core flow, and it is seen that the mass flux range in the data far exceeds this range. The void fraction range in the data is from [[ ]], while a typical exit void fraction in BWR fuel ranges from [[ ]], for the average bundle, to approximately [[ ]] for a high power 10X10 fuel bundle such as GE14 under EPU conditions. In summary, the database for the void correlation covers all fuel products including 10X10 fuel and all operating ranges including EPU conditions.

The GE void fraction correlation is described in detail in the approved Reference 69-3. The qualification documented in the approved Reference 69-4, where the void correlation was compared to [[ ]] data points from the most representative full-scale bundles, yielded a standard deviation of [[ ]] in the void fraction, while the qualification against the wider set of [[ ]] data points as documented in References 69-1, 69-5 and the approved reference 69-7 yielded a standard deviation or [[ ]] in the void fraction (See Table SRXB-A-69-2).

The part length rod (PLR) is the major new feature in current fuel products. The impact of PLRs has been investigated for a 4X4 bundle for a pressure of 1 MPa and more recently for an 8X8 bundle at rated BWR pressure of 7.2 MPa (Ref. 69-7). A small increase, approximately [[ ]], was observed in void fraction downstream of the PLRs compared to the case with no PLR (See Figure SRXB-A-69-1) for the low-pressure 4X4 data. The recent more representative 8X8 data taken at normal operating pressure shows a small increase, on the order of [[

NON-PROPRIETARY VERSION

]].

The void correlation has been implemented into the GE design codes such as PANACEA/ODYN/ISCOR/TASC and the correct implementation of the void correlation has been demonstrated by functional testing. Therefore, the qualification of the void correlation applies for all design codes except TRACG. TRACG (Ref. 69-6) has been separately compared to a set of the same data discussed above and yielded a standard deviation of [[ ]] in the void fraction.

Finally, comparisons have been made to pressure drop data taken in the ATLAS test facility using full-scale test assemblies for all fuel products including the current 10X10 GE14 fuel. This testing covers a wide range of conditions including EPU conditions. For GE14 the bundle pressure drop was predicted with a mean error of [[ ]] and a standard deviation of [[ ]]. Since the pressure drop cannot be matched unless the void fraction is accurately predicted, these tests serve as an independent confirmation of the void correlation.

In the current licensing methodology with ODYN/TASC the modeling uncertainty is derived from the comparisons to the Peach Bottom 2 turbine trip tests (Ref. 69-4). Reference 69-4 also contained an alternate analysis where the void fraction was perturbed and the impact on the OLMCPR determined. In this alternate analysis the void fraction was perturbed by [[ ]], which bounds the uncertainty in the void correlation at the 95% confidence level. This comparison demonstrated that the uncertainty in the void correlation is covered in the current design process. This process has been repeated with the introduction of new fuel types such as 10X10 fuel. A similar approach is used for TRACG (Ref. 69-6) where the impacts of all model uncertainties including the uncertainty in the void fraction are combined in a statistical process to determine the OLMCPR at the 95% confidence level.

Response to Part (b)

The monitoring system is based on a best estimate calculation with PANACEA and is used to monitor that the design limits, such as the OLMCPR, are not exceeded. These design limits are determined, as discussed above, considering the model uncertainties, which include the void fraction uncertainty, e.g., the power distribution uncertainties include the effect of the void fraction uncertainty. In the application methodology these model uncertainties are explicitly considered such that bounding values for the design limits, such as the OLMCPR, are determined. In other words, an adder to cover the void fraction uncertainty is already included in the OLMCPR. Therefore, including an uncertainty in the monitoring system to account for the void fraction uncertainty would be equivalent to accounting for this uncertainty more than once and would be inappropriate. In summary, the core monitoring system is based on best estimate methods, where no uncertainties are considered, and the impacts of the uncertainties, such as void fraction uncertainty, are considered in the thermal limits to which the bundles are monitored. GE's 3D-MONICORE core monitoring system and the process by which the uncertainties are included in the limits were reviewed and approved by NRC as documented in References 69-7 and 69-8.

NON-PROPRIETARY VERSION

References:

69-1 J. A. Findlay and G. E. Dix, BWR Void Fraction and Data, NEDE-21565, January 1977. General Electric Proprietary Information.

69-2 N. Zuber and J. A. Findlay, Average Volumetric Concentration In Two-Phase Flow Systems, ASME J. Heat Transfer, November 1965.

69-3 TASC-03A, A Computer Program for Transient Analysis of a Single Channel, NEDC-32084P-A, Revision 2, July 2002.

69-4 Letter, J. S. Charnley (GE) to H. N. Berkow (NRC), Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A, MFN-003-086, January 16, 1986.

69-5 Letter, G. Stramback (GE) to NRC, Completion of Responses to MELLLA Plus AOO RAIs (TAC No. MB6157), MFN 04-026, March 4, 2004.

69-6 TRACG Application for Anticipated Operational Occurrences (AOO) transient Analyses, NEDE-32906P-A, Revision 1, April 2003.

69-7 Methodology and Uncertainties for Safety Limit MCPR Evaluations, NEDC-32601P-A, August 1999.

69-8 Power Distribution Uncertainties for Safety Limit MCPR Evaluation, NEDC-32694P-A, August 1999.

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Table SRXB-A-69-1
Void Fraction Correlation Database

Data Source	Geometry	Hydraulic Diameter (m)	Pressure (MPa)	Mass Flux (kg/m^2-sec)	Inlet subcooling (K)	Exit quality (Max.)
[[						
						]]

Table SRXB-A-69-2
Comparison Between Void Correlation and Database
(Taken from References 69-5 and 69-7)

Data Source	Data Points (N)	Average Error $\overline{Da} = \overline{a_m - a_c}$	Standard Deviation S_{Da}
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Figure SRXB-A-69-1
4X4 Void fraction Data – Sensitivity to PLR

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Figure SRXB-A-69-2
8X8 Void fraction Data – Sensitivity to PLR for Low Flow

NON-PROPRIETARY VERSION

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Figure SRXB-A-69-3
8X8 Void fraction Data – Sensitivity to PLR for High Flow

RAI SRXB-A-70

The response to RAI SRXB-A-55 did not fully answer the question. Explain why it is acceptable to exceed the void-quality correlation ranges. Provide the plot that shows the void fractions behavior at the high void conditions or quality behavior.

Response to RAI SRXB-A-70

As explained in the response to RAI SRXB-A-69, part (a), the void correlation is based on void fraction data up to [[ ]], 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). A void fraction of [[ ]] is actually 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., momentarily during a transient) by any significant amount.

NON-PROPRIETARY VERSION

For illustrative purposes, consider a one-dimensional, steady state energy balance for a BWR fuel channel. It can be shown that

$$X(z) = \frac{h_{in} - h_f}{h_{fg}} + \frac{1}{\dot{m} h_{fg}} \int_0^z \dot{q}'(\xi) d\xi, \qquad (70\text{-}1)$$

where the definition of flow quality is given by

$$X = \frac{\dot{m}_g}{\dot{m}_f + \dot{m}_g} \qquad (70\text{-}2)$$

The flow quality given by Equation 70-1 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.

Figure SRXB-A-70-1 shows a typical plot of the void-quality relationship for a flow typical of a high power/flow ratio fuel bundle. This Figure shows the void-quality relationship for the entire range from zero to one. It should, however, 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. 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 [[ ]]. A high void fraction of 1.0 is only possible for a severe accident scenario such as a loss of coolant accident. It is seen that the void-quality relationship is very flat in the high quality range and even a substantial increase in quality (substantial increase in power) would have negligible impact on the void fraction (exit void fraction). Therefore, even if the [[ ]] upper range of the void correlation were to be exceeded, no significant error will be introduced relative to the uncertainty in the void correlation, which is already included in the licensing methodology.

Another point can be inferred from Equation 70-1, together with Figure SRXB-A-70-1. The highest void fraction is at the top of the fuel bundle and is a result of the total integrated power in the bundle. The highest nodal power, however, is located well below the top of the bundle. Therefore, the nodes with the highest power will have a lesser void fraction than the maximum bundle void fraction. Similarly, in a transient event, the quality response in the fuel bundle is given by the mass and energy balance. It is evident from Figure SRXB-A-70-1 that the void response and the corresponding void reactivity feedback from a given quality response is much less at high void fractions that at low void fractions.

In summary, the GE void correlation is based on test data and covers a broad range of conditions (See the response to RAI SRXB-A-69). The correlation supports the full range of conditions expected during BWR operation, even at up-rated conditions. The correlation uncertainty is well defined, relatively small, and appropriately accounted for in the SLMCPR. It is not necessary to incorporate any additional penalties. Extrapolation beyond the test database ([[ ]] voids) is considered unusual and rare; and if required for a particular situation, the need to extrapolate would not be expected to introduce any appreciable error.

NON-PROPRIETARY VERSION

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Figure SRXB-A-70-1
Typical Void-Quality Relation at High Power/Flow Ratio

3.4 LIMITING CONDITIONS FOR OPERATION

2. The solution temperature, including that in the pump suction piping, shall be maintained above the curve shown in Figure 3.4.2.

3. The combination of Standby Liquid Control System pump flow rate, boron concentration, and boron enrichment shall satisfy the following relationship for the Standby Liquid Control System to be considered operable:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq \cancel{1}\ 1.29$$

12

where:

C = the concentration of sodium pentaborate solution (weight percent) in the Standby Liquid Control System tank

E = the boron-10 enrichment (atom percent) of the sodium pentaborate solution

Q $\cancel{=}$ $\geq$ 35 gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

D. If Specification 3.4.A or B is not met, an orderly shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

E. If Specification 3.4.C is not met, action shall be immediately initiated to correct the deficiency. If at the end of 12 hours the system has not been restored to full operability, then a shutdown shall be initiated with the reactor in cold shutdown within 24 hours of initial discovery.

4.4 SURVEILLANCE REQUIREMENTS

2. Sodium pentaborate concentration shall be determined at least once a month and within 24 hours following the addition of water or boron, or if the solution temperature drops below the limits specified by Figure 3.4.2.

3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

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where:

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$Q \geq 35$ gpm

$\frac{M251}{M}$ = a constant (the ratio of mass of water in the reference plant compared to VY)

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4.4 SURVEILLANCE REQUIREMENTS

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3. The boron-10 enrichment of the borated solution required by Specification 3.4.C.3 shall be tested and verified once per operating cycle.

Entergy

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September 18, 2005

Docket No. 50-271
BVY 05-086
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 34
Extended Power Uprate – Additional Information

References:

1) 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

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

3) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 32, Extended Power Uprate – Additional Information," BVY 05-083, September 10, 2005

4) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 33, Extended Power Uprate – Response to Request for Additional Information," BVY 05-084, September 14, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The attachments to this letter provide supplemental information in response to requests for additional information from the NRC staff (Reference 2) and other supplemental information to update the application for a license amendment. As a result of recent discussions with the NRC staff and its recent audit of analytical methodologies of General Electric (GE) that are used for the design and evaluation of VYNPS' fuel, the NRC staff identified the need for additional information reflected in several of the requests for additional information (RAIs) contained in Reference 2. Because of the recency of the requests, one (Reference 2) RAI remains to be addressed (i.e., NRC RAI SRXB-A-68); the remaining RAI will be addressed in a submittal that will be made by September 23, 2005.

Attachment 1 to this letter is a revision to Exhibit EMEB-B-18-1, Rev. 1, Attachment 4 (regarding the steam dryer acoustic load uncertainty evaluation) that was provided to the NRC staff in Reference 4. Inadvertently, several figures were not included in the original submittal. The omitted figures include comparisons of power spectral densities for certain transmitter locations. Attachment 1 consists of thirty figures (EMEB-B-18-1-4-1 through EMEB-B-18-1-4-30) and supersedes, in its entirety, Exhibit EMEB-B-18-1, Rev. 1, Attachment 4 provided in Reference 4, Attachment 1 (Proprietary Information) and Attachment 8 (Non-Proprietary Version). Attachment 1 to this letter does not contain proprietary information.

In the response to RAI SRXB-A-66 (Reference 3), Entergy stated that certain tabulated data supporting the response to the RAI would be submitted to the NRC staff as Microsoft Excel spreadsheets. That information is included herein as Attachment 2 on a compact disk. The data contained on the compact disk is considered Proprietary Information to General Electric and is covered by the affidavit accompanying the response to SRXB-A-66 in Reference 3. An explanatory "Read Me" file (non-proprietary) contained on the CD is included in hardcopy as part of Attachment 2.

As a result of discussions with the NRC staff, Entergy is providing in Attachment 3 a more extensive response to RAI SRXB-A-64. This response supplements the response that was originally provided in Reference 3.

Attachment 4 contains responses to NRC Reactor Systems Branch RAIs SRXB-A-65 and SRXB-A-67 that were posed in Reference 2. These RAIs and the responses thereto contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with the provisions of that regulation. Attachment 4 is considered to be Proprietary Information in its entirety. Attachment 5 is a non-proprietary version of Attachment 4. An affidavit provided by General Electric Company, supporting the proprietary nature of the document, is provided as Attachment 7.

Attachment 6 provides a response to RAI SRXB-A-71 that was asked in Reference 2.

Attachment 8 of this letter provides a copy of the demonstrated shutdown margin (SDM) calculation for the current operating cycle (i.e., cycle 24). This SDM calculation is referenced in the response to RAI SRXB-A-67, part (b).

There are no new regulatory commitments contained in this submittal.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

The following attachments are included in this submittal:

Attachment	Title
1	Revised Exhibit EMEB-B-18-1, Rev. 1, Attachment 4
2	RAI SRXB-A-66 Data (Compact Disk) (PROPRIETARY INFORMATION)
3	Supplemental Response to SRXB-A-64
4	Responses to RAIs SRXB-A-65 and SRXB-A-67 (Proprietary Information)
5	Responses to RAIs SRXB-A-65 and SRXB-A-67 (Non-Proprietary Version)
6	Response to RAI SRXB-A-71
7	General Electric Affidavit
8	Demonstrated Shutdown Margin

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings at your earliest convenience to resolve any remaining issues. If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 18, 2005.

Sincerely,

Norman L. Rademacher
Director, Nuclear Safety Assurance
Vermont Yankee Nuclear Power Station

Attachments (8)

cc: (see next page)

cc: Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
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U.S. Nuclear Regulatory Commission
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Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
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USNRC Resident Inspector (w/o attachments)
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P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner (w/o proprietary information)
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

Attachment 3

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Supplemental Response to SRXB-A-64

Total number of pages in Attachment 3 (excluding this cover sheet) is 2.

RAI SRXB-A-64

Provide the values for maximum bundle power and average power densities at VYNPS before and after the EPU.

Supplemental Response to RAI SRXB-A-64

Core thermal power information for VYNPS is provided in Table SRXB-A-64-1. The table provides the average power densities before and after EPU. The table also provides channel (bundle) power information requested by the RAI.

Table SRXB-A-64-1
Vermont Yankee Nuclear Power Station
Power Information

Parameter	Pre-EPU	Post-EPU	% Change
Total Core Thermal Power (MWt)	1593	1912	20
Power Density (kW/liter)	48.9	58.7	20
Channel Average Power (MWt)	4.33	5.20	20
Maximum Channel Power (MWt)	~7	~7	N/A

The channel average power is the total core thermal power divided by the number of fuel channels (368). The maximum channel powers shown in Table SRXB-A-64-1 are essentially unchanged by EPU operation. The values are presented as approximately 7 MWt in order to emphasize this point. The reason for this is that high power channels are limited by thermal limits. In other words, the peak LHGR and/or OLMCPR limits effectively put a ceiling on the maximum allowable bundle power. These limits are associated with the fuel and core designs, and are not a direct function of EPU. The actual pre- and post-EPU maximum bundle powers are 7.02 and 7.37 MWt, respectively. Again, the maximum values will likely change in the future depending on the particular reload core and bundle design. The maximum bundle power could also (potentially) be impacted by other design constraints, for example, the margin to the OLMCPR limit (i.e., how the peak bundles are projected to operate relative to the limit).

The NRC safety evaluation (SE) for constant pressure EPU documented in NEDC-33004P-A summarizes key elements related to the power uprate, including a discussion of power density. Section 1.3.3 of the SE contains the statements: "The CPPU approach achieves the power uprate by increasing the core average power density proportional to the core thermal power increase. This affects the reload core design and operating flexibility, the reactivity characteristics and the cycle energy requirements. No changes in fuel mechanical designs or fuel design limits are required to implement the CPPU process." From a core designer's point of view, the power uprate is effectively achieved by flattening the core radial power shape. More channels operate at or above the pre-uprate average bundle power level.

The next VYNPS operating cycle (i.e., cycle 25) core was designed to support operation under constant pressure power uprate (CPPU) conditions. The additional reactivity necessary to achieve the target power and cycle length is provided through the reload core design (i.e., the selection of bundle enrichments and the reload batch fraction).

Attachment 5

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Responses to RAIs SRXB-A-65 and SRXB-A-67

NON-PROPRIETARY VERSION

Total number of pages in Attachment 5 (excluding this cover sheet) is 28.

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RAI SRXB-A-65

Linear Heat Generation Rate (LHGR)

The NRC staff had previously asked whether any uncertainties were applied to the LHGR limit (curve) and the actual operating nodal steady state kilowatt/foot (kw/ft). The response to RAI SRXB-A-41 took credit for a reduced value in the gradient uncertainty. However, the power allocation and the pin power uncertainty values were increased to accommodate the lack of gamma scans of the current GE14 fuel designs as operated. The RAI response states that a local uncertainty of [[ ]] in LHGR is assumed in the development of the LHGR, implying that the [[ ]] kw/ft uncertainty addressed in the response to the staff RAI 5, associated with the NRC-approved safety limit minimum critical power ratio (SLMCPR) topical report NEDC-32694P-A, was intended for the generation of the LHGR limit. However, it is the staff's understanding that the uncertainty analyses provided in the RAI 5 response was addressing the uncertainty to be applied to the kw/ft calculated by the core monitoring system (e.g., 3D MONICORE) as opposed to a [[ ]] uncertainty assumed during the development of the LHGR curve.

The RAI 5 to NEDC-32694P-A stated that the process computer monitors peak kw/ft and maximum average planar linear heat generation rate (MAPLHGR). The peak kw/ft and the MAPLHGR depend on the bundle axial power distribution and, consequently, are significantly more sensitive to the 3-D MONICORE replacement of the traversing incore probe (TIP)/local power range montior (LPRM) axial power distribution. The RAI asked for uncertainty analysis for the 3-D MONICORE prediction of peak kw/ft and MAPLHGR. In the response, GE provided the following uncertainty analyses, which specified the uncertainty that would be applied to the peak kw/ft calculations:

<u>Nodal Power Uncertainty:</u> The nodal power uncertainty for 3D MONICORE is a combination of: 1) the uncertainty in the four bundle power at axial node k; 2) the uncertainty in the power allocation factor at node k; 3) the LPRM update uncertainty; and 4) the uncertainty in the TIP axial power distribution at node k. [[

]] The total nodal power uncertainty is, therefore, equal to:

[[
]]

<u>Pin Power Peaking Uncertainty:</u> The pin power peaking uncertainty can be determined from the factors outlined for the R-factor uncertainty summarized in Section 3 of NEDC-32601. Specifically, the pin power peaking uncertainty is a combination of 1) the model uncertainty, 2) the manufacturing uncertainty, and 3) the channel bow uncertainty. As in Section 3 of NEDC-32601P, the model uncertainty is a combination of the pin

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peaking uncertainty determined from Monte Carlo comparisons (1.44%) and an uncertainty due to flux gradients from neighboring bundles. [[

]] All of these pin power uncertainties have been combined in NEDC-32601P as:

[[
]]

The total LHGR uncertainty is the combination of nodal and pin power uncertainties:

[[ ]]

Staff Position

As shown in the NRC-approved SLMCPR methodology specified in NEDC 2694P-A, σ_{LHGR} changes with σ_{pal} and σ_{MC}. Accepting the reduction in the gradient uncertainty, a σ_{LHGR} of [[ ]] should be applied to the calculated kw/ft as discussed and specified in the NRC-approved licensing topical report. Because a [[ ]] uncertainty is assumed in the generation of the LHGR limit, this does not mean that the uncertainties due to the impact of modeling uncertainties on the operating kw/ft can be traded off with the [[ ]] uncertainty assumed in the development of the limit. The limit is developed based on the accuracy of the thermal-mechanical analytical models, methods and code systems. Therefore, any uncertainty currently applied in the development of the LHGR limit, can only be taken credit for or changed if it is demonstrated that for the current fuel designs and operating conditions additional nonconservatisims would not offset the "no cause" [[ ]] uncertainty.

The increase in the power allocation and pin power uncertainty applied to the SLMCPR does not directly lead to a proactive increase in the predicted steady state kw/ft. Therefore, potential underestimation in the nodal powers (bundle and peak pin) need to be accounted for. As evident in the RAI responses, the core-wide axial and nodal uncertainties determined through the TIP comparisons are not applied to the transient or accident analyses. The core-wide radial (e.g., bundle uncertainty σP4B) uncertainty is limited to the SLMCPR calculations. Therefore, there are no nodal or pin uncertainties that are applied to the predicted kw/ft. It is the staff's position that a [[ ]] kw/ft uncertainty be applied to the operating kw/ft calculated in the core simulator code, because of the following reasons:

1. Since there are no measurement data to validate the bundle and pin axial power, the uncertainties in the cross-sections and the pin powers are based on the TIP four bundle readings and the MCNP/TGBLA code-to-code comparisons. The four radial bundle uncertainty $\sigma_{P4B\ nodal}$ is derived from TIP comparisons and is applied to the SLMCPR. The power allocation between the four bundles $\sigma_{PAL\ nodal}$ derived from measurement data is also applied to the SLMCPR. The predicted operating kw/ft

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relies on the predicted axial bundle power and the pin powers. Although the 3D MONICORE adjusts the four bundle axial power peaking to the TIP reading, the adjusted axial power peaking is based on at least four bundle TIP response. Therefore, the power allocation in each bundle must be incorporated in the predicted kw/ft. Similarly, the uncertainty in the pin power needs to be included in the calculation of the peak kw/ft. Therefore, the calculated [[ ]] uncertainty needs to be applied to the predicted kw/ft, to account for the uncertainties in the cross-sections and the pin powers.

2. The [[ ]] power uncertainty bias, applied in the fuel rod internal pressure cited in the Alternative Approach (Supplement 30, Attachment 1), accounts for the differences between the design conditions the rod internal pressure calculations are based on and the rod internal pressures that would be obtained if actual operating history conditions were simulated. In other words, the [[ ]] uncertainty accounts for the difference between the as-designed and as-operated conditions.

3. The Alternative Approach cites an additional power uncertainty of [[ ]] power that is not specifically assigned to any cause. The Alternative Approach also states that separate experimental benchmarking information confirms that the model uncertainties remain valid. However, it is the NRC staff's understanding that, for the current fuel designs (GE14) as operated, no benchmarking of the fission gas inventory was performed. It is also the understanding that the [[ ]] "no cause" uncertainty is based on the original NRC-approval of the thermal-mechanical methodology and models. Therefore, it is not evident if a conservatism of [[ ]] would actually be available, if the operating and core design changes implemented since the initial development of the fuel thermal-mechanical models are evaluated. Neither the RAI response nor the Alternative Approach demonstrated this. The RAI response also did not discuss what uncertainties are assumed in the transient overpower kw/ft and if there is sufficient margin available.

4. The application of [[ ]] margin to the calculated kw/ft values would ensure that there are sufficient margins to the pellet exposure limits. The [[ ]] additional margin in the peak kw/ft would require a decrease in the nodal (bundle-wise) operating kw/ft, which would provide additional margin in bundle averaged accumulated exposure.

Response to RAI SRXB-A-65

The 3D Monicore surveillance system discussed in the RAI is intended to be [[

]] The following points are provided in response to items 1 – 4 under the *staff position* heading in the RAI.

1. As stated above, the GE objective is for the core monitoring methods to provide the most accurate [[ ]] quantification of the actual operating state. The

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uncertainty in that operating state calculation is addressed [[

]], even when uprated conditions are considered, as discussed further below.

2. The [[ ]] bias applied to fuel rod internal pressure calculations is an allowance [[

]]

Variations between the analyzed power history and actual power histories are addressed through the analysis assumption [[

]] Figure SRXB-A-65-1 presents the [[

]] (LHGR Operating Limit), as compared to an actual projected operating history for Bundle JLC505 Rod K4 Node 5 both under power uprate conditions and without power uprate. JLC505 experiences the highest bundle nodal exposure (Node 5) for any bundle in the VYNPS Cycle 25 core both with and without power uprate conditions. Rod K4 of JLC505 experiences the highest local exposure within that peak exposure bundle node. It is noted from Figure SRXB-A-65-1 that (1) the difference between the non-uprated and uprated nodal operating histories is relatively small, and (2) both operating histories are well bounded by [[

]] the LHGR Operating Limit. It should be noted that at any point in time the local fuel rod power level could potentially momentarily approach or even be at the LHGR Operating Limit[[

]]. The presented power history for JLC505 Rod K4 Node 5 provides a characterization of a typical operating history for a fuel rod node that operated at highest power, on the average over lifetime, of all fuel rods in the third cycle reload batch present in VYNPS Cycle 25. In this case, JLC505 Rod K4 Node 5 did not approach the LHGR Operating Limit prior to Cycle 25 and is not projected to approach the LHGR Operating Limit during VYNPS Cycle 25, although, again, it is recognized that any individual fuel rod, either JLC505 Rod K4 during actual Cycle 25 operation or any other fuel rod, could briefly operate at the LHGR Operating Limit. [[

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3. The basic fuel rod thermal-mechanical design analysis methodology currently used by GNF was implemented with GESTAR Amendment 7 with corresponding NRC approval as documented in Reference 65-1. Subsequent to the initial methodology approval, the NRC, in conjunction with NRC consultant and fuel rod thermal-mechanical analysis expert Dr. Carl Beyer (PNL), again reviewed the fuel rod thermal-mechanical design analysis methodology as documented in Reference 65-2. At the time of the original NRC review and approval of the fuel rod thermal-mechanical design and analysis methodology, the uncertainty in the fuel rod operating power level was addressed (1) directly through explicit consideration of the local power level variations that could develop [[

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]].

In NEDC-32694-P-A, the response to RAI-5 (page A-10) identified an uncertainty in local LHGR of [[ ]]. Later revisions to the uncertainty treatment described in RAI-5 resulted in a slight increase to [[ ]] (page B-3 in the same topical report). Applying the adjusted uncertainty driven by lack of gamma scan data from RAI SRXB-A-41 for VYNPS would result in an uncertainty of [[

]]

For the fuel rod thermal-mechanical transient overpower analyses, again, the fuel rod is assumed [[

]]. This approach introduces considerable conservatism relative to the conditions that would be calculated for an actual operating history with a randomly placed transient event.

4. [[ ]] exposure limits are established for each product line. These limits are conservatively established with approved methods, including appropriate provisions for uncertainties. The limit established for GE14 fuel is applicable under the proposed CPPU conditions for VYNPS. The fuel rod thermal-mechanical performance consideration of greatest interest at exposures near the peak pellet exposure limit is the fuel rod internal pressure. As discussed above, a significant conservatism, most especially for the fuel rod internal pressure calculation, is [[

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

The discussion below supports items 1 – 4 above and contains additional information regarding the first paragraph of RAI SRXB-A-65.

As a point of clarification to the first paragraph of the RAI, the response to RAI II.5 in NEDC-32694P-A applies to uncertainties and core monitoring considerations. The LTR covers these topics, as well as their relevance to the SLMCPR methodology. The original RAI response provided a derivation of the uncertainty in the predicted peak LHGR. As discussed in the topical report, the same component uncertainties are incorporated into the SLMCPR. However, the LTR did not directly address how the uncertainties were incorporated [[ ]]. The responses documented in

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NEDC-32694P-A accurately describe the uncertainties, but only in terms of their application in the SLMCPR.

The response to SRXB-A-41 indicated a slight increase in the predicted peak LHGR uncertainty. The response also indicated that power uncertainty is considered [[ ]]. The response included the statement "A local uncertainty of [[ ]]." This statement is accurate. This [[ ]] local power uncertainty is utilized with the application of the GESTR thermal-mechanical model [[ ]] for each fuel product line. Additional discussion concerning determination of the exposure-dependent LHGR Operating Limit is given below.

For each GNF fuel design, including GE14 as applied to VYNPS, 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, including those which address response to anticipated operational occurrences, are explicitly satisfied.

The exposure-dependent LHGR operating limits are determined through the performance of a number of fuel rod thermal-mechanical analyses. As shown to the NRC staff during the GE Methods audit, 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. [[

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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.

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Figure SRXB-A-65-2 is a chart presented to the USNRC in recent discussions to describe the results of the GE14 fuel rod thermal-mechanical design and licensing analyses, and is included here for documentation purposes. The primary result of the fuel rod thermal-mechanical design and licensing analyses is development of the LHGR Operating Limit. The analyses that contribute directly to the development of that limit are the analyses for [[

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In summary, with this methodology, the exposure-dependent LHGR Operating Limit is determined to ensure that the fuel rod thermal-mechanical design and licensing limits, such as the fuel rod internal pressure limit, will not be exceeded [[

]].

References

65-1. Letter from C. O. Thomas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A Amendment 7 to Revision 6, GE Standard Application for Reactor Fuel," March 1, 1985

65-2. Letter from Robert M. Gallo (NRC) to C. P. Kipp (GE), "NRC Inspection Report No. 99900003/96-01," September 10, 1996

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Figure SRXB-A-65-1
VYNPS Cycle 25 Projected Actual Operating History for JLC505 Rod K4 Node 5 - Comparison Between Uprated and Non-Uprated Conditions

(JLC505 Node 5 is the highest projected bundle nodal exposure in VYNPS Cycle 25; rod K4 is the highest exposure rod node in bundle JLC505 Node 5. See further description in Item 2 text above on page 4 of 28.)

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RAI SRXB-A-67

Shutdown Margin (SDM)

In the Alternative Approach and in the RAI responses, VYNPS SDM data was not provided as discussed in the July 12, 2005, telephone conference. As the NRC staff pointed out in the June 30, 2005 meeting, Figure 25-18, "Cold Critical Eigenvalues-All Cycles Studies," of the MFN-05-029 shows that the actual cold eigenvalue tracking of different plants show a scatter of the bias of each plant. However, the uncertainty applied to each plant is obtained by RMS averaging of bias from all plants. Thus, it seems that a bias of 0.38% Δk/k is applied to the calculated core-wide critical keff (in-sequence cold eigenvalue) although the bias from critical (keff = 1.0) may be larger for a given plant. Also, presenting the calculated cold critical eigenvalue alone does not indicate if the critical control rod positions were predicted.

a) Provide the VYNPS cold critical eigenvalues for at least two cycles. Include the recent mid-cycle startup cold critical eigenvalue. Include tables of the predicted keff with the CR withdrawals and indicate predicted critical eigenvalue and the calculated cold critical eigenvalue corresponding to when the core became critical. Evaluate the bias in the VYNPS cold critical eigenvalue data.

b) Provide the actual calculated SDM, with the correction for the period, temperature and peak reactivity.

c) The alternative approach states that for VYNPS "the standard design SDM is 1.1% Δk/k to provide additional flexibility in cycle length and operations." Clarify this statement. Is this an additional margin included to meet the cycle energy needs or is this additional conservatism that ensures SDM for any point in the cycle?

d) The Alternative Approach did not include impact of potential underprediction in reactivity and bundle and pin powers on the SLC system cold shutdown capability. Provide an evaluation of the SLC system shutdown capability and rod withdrawal error analysis.

e) Demonstrate that the [[ ]] would not have an important impact when the [[ ]] void fraction and extrapolation to higher voids are used. Also, provide a discussion on what such an under-prediction would have on the accuracy of the local reactivity predictions and what impact, if any, it would have on the SDM, SLC system cold shutdown and rod withdrawal error calculations.

f) The RAI responses stated that the objective is for the eigenvalue trendline to remain constant and consistent from cycle to cycle for a given plant, unless significant change in core loading design results in some change in the trendline. However, the trendline is not a licensing parameter and can be adjusted according to a new trendline fitting a change in the data. The licensing parameter is the SDM.

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Therefore, from a licensing and safety perspective, the difference between the calculated keff for a critical reactor and the deviation from 1.0 is the most important parameter. Explain why it is not desirable for the keff bias and uncertainty to be derived on plant-specific bases. Thus, ensuring a better adjustment applied to the keff bias assumed in the SDM calculations would be based on individual plant's characteristic response and the accuracy of the neutronic methods.

Response to RAI SRXB-A-67

Response to Part (a)

The cold critical eigenvalues for the Vermont Yankee Nuclear Power Station (VYNPS) Cycles 23 and 24 are presented below. The results shown below are for the TGBLA06/PANAC11 set of methods. Because Cycle 24 was the first cycle at VYNPS to be designed and licensed with PANAC11, no predicted eigenvalues had been established for earlier cycles. The previous cycle cold criticals were analyzed with PANAC11 however in order to establish a data base from which the Cycle 24 predicted eigenvalues were developed. The mid-cycle Cycle 23 predicted eigenvalue was established by taking the actual beginning of cycle (BOC) eigenvalue and adjusting it by the standard reduction in cold eigenvalue with cycle exposure (used when sufficient mid-cycle information is not available for a plant). The process for determining predicted cold critical eigenvalues is discussed in the response to part (f) of this request.

The cold eigenvalues shown are very typical for other BWRs operating with GE fuel and analyzed with PANAC11 methods (Reference 67a-1). [[

]]

Cycle	Cycle Exposure (MWd/ST)	Predicted Eigenvalue	Critical Eigenvalue	Difference (Δk)
23	BOC	[[		
	7417			
24	BOC			
	961			]]

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Response to Part (b)

The VYNPS Technical Specification (TS) Shutdown Margin (SDM) is determined following a core reload, at the beginning of each cycle during plant startup. A copy of the demonstrated SDM calculation for the current operating cycle (cycle 24) is attached (see Attachment 8). The calculation involves correcting the SDM for the effects of temperature and period present at the critical measurement. As calculated, the SDM also includes a correction for any difference in peak reactivity at any point in the cycle, R. The period and temperature correction is obtained from the Cycle Management Report, as is the correction for the difference in peak reactivity, R. It should be noted that the temperature correction is a translation to the most reactive condition.

As shown in the accompanying worksheet, Cycle 24 SDM was demonstrated by test to be 1.291. The cycle was designed for Extended Power Uprate (115% CLTP) and a SDM of 1.1, which indicates that the SDM design criterion was easily met for this cycle.

During discussions related to this subject, some other issues were identified by the reviewers, and are addressed below:

The VYNPS TS require that the SDM, at any time there is fuel in the core, shall be greater than or equal to 0.38% ΔK/K with the analytically determined highest worth rod fully withdrawn. The 0.38% ΔK/K was determined based upon a statistical combination of allowed manufacturing tolerances and calculational uncertainties. The calculational uncertainties were determined from a statistical analysis of measured and calculated criticals performed at an operating BWR.

Procedurally, if the demonstrated SDM is less than 0.38% ΔK/K, then the shift manager is immediately notified and SDM must be restored within 6 hours or the reactor must be in Hot Shutdown within the next 12 hours. If the corrected critical eigenvalue is different from the expected critical eigenvalue by more than 1% ΔK/K, then the shift manager is immediately notified and the reactor must be shut down until the cause is determined. Additionally, if the corrected critical eigenvalue is different from the expected critical eigenvalue by greater than 0.75% ΔK/K, then the reactor engineering superintendent is notified and a Condition Report is initiated.

Typically, the SDM demonstration is performed during the beginning-of-cycle (BOC) startup. Within the calculation of the demonstrated in-sequence SDM, there is a factor, R, that accounts for a decrease in SDM during the most reactive point in the cycle. This factor is zero when SDM is determined at the most reactive point in the cycle. For those situations when the SDM is not determined at the most reactive point in the cycle, the R factor is subtracted from the demonstrated SDM.

With regard to the effect of the assumed critical eigenvalue and its uncertainty on the demonstrated SDM, the following discussion is offered:

Per the Cycle Management Report (CMR), the equation for SDM is as follows:

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$$SDM = K_{Crit} - K_{SRO} + K_{Temp} - K_{Per} - R$$

where,

K_{Crit} = Eigenvalue when critical is achieved,
K_{SRO} = Eigenvalue with the strongest rod out (SRO),
K_{Temp} = ΔK temperature correction,
K_{Per} = ΔK period correction, and
R = Maximum decrease in SDM throughout the cycle.

But since,

K_{Crit} = K_{eff} with all rods in (ARI) + ΔK of the critical rod pattern (CRP)
= $K_{ARI} + \Delta K_{CRP}$, and

K_{SRO} = K_{eff} with all rods in + ΔK of the strongest rod out
= $K_{ARI} + \Delta K_{SRO}$, then

$$SDM = (K_{ARI} + \Delta K_{CRP}) - (K_{ARI} + \Delta K_{SRO}) + K_{Temp} - K_{Per} - R$$

which simplifies to:

$$SDM = \Delta K_{CRP} - \Delta K_{SRO} + K_{Temp} - K_{Per} - R$$

K_{ARI} is subject to the influence of the assumed critical eigenvalue and its uncertainty. It can be seen from the final equation that K_{ARI} cancels out and the demonstrated SDM is not influenced by the assumed critical eigenvalue or its uncertainty. However, it should be noted that ΔK_{SRO} includes a 0.003 ΔK/K adjustment to account for the methods bias which occurs when normalizing shutdown margin calculations to a cold eigenvalue derived from in-sequence critical benchmarking data.

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Response to Part (c)

The VYNPS TS 3.3.A.1 requires that any time fuel is the in the core, the core loading shall be limited to that which may be made subcritical in the most reactive condition during the operating cycle with the highest worth, operable control blade fully withdrawn and all other operable rods inserted.

The shutdown margin shall be:

(a) Greater than or equal to 0.38% Δk/k with the highest worth rod analytically determined; or

(b) Greater than or equal to 0.28% Δk/k with the highest worth rod determined by test.

Entergy confirms sufficient SDM for VYNPS at the BOC based upon greater than or equal to 0.38% Δk/k.

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. To ensure that ≥ 0.38% Δk/k is always satisfied, a design margin of 1% SDM has been used by GE for many years. 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

In all of these factors, the most significant factor is allowance for operation different from that projected. VYNPS must maintain sufficient operational flexibility to protect the core and fuel while maintaining acceptable economic objectives. Factors affecting the GE application methodology are quantified through the uncertainty in cold critical eigenvalue and deviation from expectations. These data are provided in the responses to RAIs SRXB-A-67 part (a) and SRXB-A-67 part (b).

The additional 0.1% Δk/k that VYNPS requires results from consideration of inverted B_4C tubes in the core. Based upon a total of 82 inverted B_4C tubes in 44 control rods in 1975, a 0.07% Δk/k SDM adder was required to compensate for the inverted B_4C tubes. [Reference 67c-1] While there are only 30 inverted B4C tubes in 13 peripherally located control blades, the 0.07% Δk/k SDM adder is still being applied until all affected control rods are discharged.

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If the SDM demonstration at VYNPS results in a SDM less than Technical Specification requirement, the plant will take actions as specified in the Technical Specifications.

Response to Part (d)

The standby liquid control system (SLCS) calculation is performed on a cycle specific basis to assure that the plant will remain subcritical in the most reactive condition when the Technical Specification (Tech Spec) minimum requirement for soluble boron is introduced into the core. The calculation is performed as a function of exposure throughout the cycle to determine the minimum SLCS shutdown margin during the cycle. This is an analytical determination, and no actual demonstration of this shutdown capability is performed as is done in the one-rod-out shutdown margin. In order to provide a high degree of assurance that the analytically determined shutdown margin will indeed result in a subcritical condition, a SLCS shutdown margin criteria is established, requiring that the analytically determined shutdown margin be greater than or equal to this shutdown margin criteria. The criteria accounts for all of the biases and uncertainties inherent in the various components of the SLCS methodology.

It should be noted that unlike the one-rod-out shutdown margin, which must be demonstrated subsequent to any reconfiguration of the core, and which is highly sensitive to the local conditions in the four bundles surrounding the withdrawn blade, the SLCS shutdown margin is driven more by core-wide reactivity effects. This makes the calculation less sensitive to nodal uncertainties in exposure and isotopic content, and more dependent on the average exposure and reactivity behavior of the various fuel batches loaded in each cycle.

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The severity of the RWE transient is largely dependent on the worth of the rod being withdrawn. The limiting bundle for the RWE for the VYNPS Cycle 25 analysis shows a controlled to uncontrolled ΔK∞ of approximately [[ ]] Of the four bundles face-adjacent to the error rod, two bundles are approximately [[ ]] including the limiting bundle. The other two bundles are approximately [[ ]]. The higher exposure bundles show a smaller ΔK∞, [[ ]], and a corresponding lower change in power and CPR during the RWE. The trend of reduction in ΔK∞, and corresponding lower change in power and CPR during the RWE, continues at exposures greater than [[

]]

Response to Part (e)

[[

]] as the impact on the 0, 40, and 70% void data is minimal. Consequently, this effect does not significantly impact the extrapolation using the 0, 40 and 70% void data to voids higher than 70%.

The above discussion indicates that there is potential for a change in the lattice reactivity of [[

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To demonstrate the reactivity impacts of this modification to the [[ ]] evaluation, a cycle of plant performance tracking using GE14 fuel in a high power density core was performed using both the current TGBLA production engineering computer program (ECP) and a version of TGBLA that was modified to correct this issue.

The hot core reactivity impact on the core tracking is shown in Figure SRXB-A-67-1 and the impact to Shutdown Margin (SDM) as a function of cycle exposure is shown in Figure SRXB-A-67-2. Table SRXB-A-67-1 and Table SRXB-A-67-2 provide the core reactivity and SDM detailed results comparisons, respectively.

As shown in the figures and tables, [[

]] These levels of impact are not significant compared to the historical uncertainty of these calculated parameters.

[[

]]

The response to NRC RAI SRXB-A-67 part (d) provides a discussion of the impact of this potential reactivity uncertainty on the SLCS SDM and Rod Withdrawal Error (RWE) analyses.

The [[ ]] (see response to NRC RAI SRXB-67d).

Response to Part (f)

The current process is consistent with the expressed concern ("Explain why it is not desirable for the k-eff bias and uncertainty be derived on plant-specific bases."). [[

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Cycle	Cycle Exposure (MWd/ST)	Critical Eigenvalue
21	BOC	[[
22	BOC	
23	BOC	
	7417	
24	BOC	
	961	]]

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References:

67a-1 MFN 05-029, TAC No. MC5780

67c-1 Letter, Dennis L. Ziemann (NRC) to G. Carl Andognini (YAEC), Docket No. 50-271, June 6, 1975

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Attachment 6

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

Response to RAI SRXB-A-71

Total number of pages in Attachment 6 (excludina this cover sheet) is 1.

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides a response to the NRC Reactor Systems Branch's (SRXB) request for additional information (RAI) SRXB-A-71 in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. The intent of individual RAI is addressed based on clarifications reached during these discussions. The information provided herein is consistent with those clarifications.

The RAI is re-stated as provided in NRC's letter of September 7, 2005.

RAI SRXB-A-71

In the response to RAI SRXB-A-6, the licensee stated "the reactivity events are analyzed with the steady state tools and the results presented regarding steady-state methods in this response are directly applicable. There are some increases in power, which are significant but remain within the comparisons between the above plants for corresponding events." This RAI response does not provide sufficient detail. The response to RAI SRXB-A-57 requested clarification to the above quoted statement. The generic event sequence was described, rather than explaining the statement in the initial RAI response. Please explain the intent of the statement in the initial submittal.

Response to RAI SRXB-A-71

The intent of the statement in quotations was that the VYNPS events analyzed with the 3D core thermal-hydraulic PANACEA model, such as the Rod Withdrawal Error and Fuel Loading Error, started from conditions within the range in other analyses as shown in Figures 6-1 through 6-6 of the response to RAI SRXB-A-6[2]. No comparison was made against the events analyzed with the steady state methods for the other plants of Figures 6-1 through 6-6 because of differences in the plant size, core design and loading, rod block monitor setup, power distribution and control rod patterns, which result in inconsistent comparisons.

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

[2] Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, 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 7

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 34

Extended Power Uprate – Additional Information

General Electric Affidavit

Total number of pages in Attachment 7 (excluding this cover sheet) is 3.

General Electric Company

AFFIDAVIT

I, George B. Stramback, state as follows:

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

(2) The information sought to be withheld is contained in Enclosure 2 of GE letter, GE-VYNPS-AEP-403, *Responses to NRC RAIs SRXB-64, 65, 67, and 71*, dated September 16, 2005. The proprietary information in Enclosure 2, *Responses to NRC RAIs SRXB-64, 65, 67, and 71*, is delineated by a double underline inside double square brackets. Figures and large equation objects are identified with double square brackets before and after the object. In each case, the superscript notation[3] refers to Paragraph (3) of this affidavit, which provides the basis for the proprietary determination.

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

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

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

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

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

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

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

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

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

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

(8) The information identified in paragraph (2), above, is classified as proprietary because it contains detailed results and conclusions from analyses supporting the extended power uprate of the Vermont Yankee Power Station utilizing analytical models and methods including computer codes and methods of applying these for safety analyses, which GE has developed. The development of these models and computer codes and methods was achieved at a significant cost to GE, on the order of several million dollars.

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

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

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

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

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

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

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

Executed on this 16 th of September 2005.

George B. Stramback
George B. Stramback
General Electric

Entergy

Entergy Nuclear Vermont Yankee, LLC
Entergy Nuclear Operations, Inc.
185 Old Ferry Road
Brattleboro, VT 05302-0500

September 28, 2005

Docket No. 50-271
BVY 05-088
TAC No. MC0761

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

Subject: **Vermont Yankee Nuclear Power Station**
Technical Specification Proposed Change No. 263 – Supplement No. 35
Extended Power Uprate – Response to Request for Additional Information

References:

1) 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

2) U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

3) Entergy letter to U.S. Nuclear Regulatory Commission, "Vermont Yankee Nuclear Power Station, Technical Specification Proposed Change No. 263 – Supplement No. 30, Extended Power Uprate – Response to Request for Additional Information," BVY 05-071, August 1, 2005

This letter provides additional information regarding the application by Entergy Nuclear Vermont Yankee, LLC and Entergy Nuclear Operations, Inc. (Entergy) for a license amendment (Reference 1) to increase the maximum authorized power level of the Vermont Yankee Nuclear Power Station (VYNPS) from 1593 megawatts thermal (MWt) to 1912 MWt.

The attachments to this letter provide supplemental information in response to a request for additional information (RAI) from the NRC staff (Reference 2). Entergy previously responded to all but one of the individual RAIs in Reference 2 (i.e., NRC Reactor Systems Branch RAI SRXB-A-68). Attachment 1 provides Entergy's response to the remaining RAI. This RAI and the response thereto contain Proprietary Information as defined by 10CFR2.390 and should be handled in accordance with the provisions of that regulation. Attachment 1 is considered to be Proprietary Information in its entirety. Attachment 2 is a non-proprietary version of Attachment 1 and is suitable for public disclosure. An affidavit provided by General Electric Company (GE), supporting the proprietary nature of the document, is provided as Attachment 3.

In response to NRC staff requests for information regarding GE's analytical methodologies for establishing fuel thermal limits, Entergy provided a VYNPS-specific approach to address postulated uncertainties in GE's methodologies in Reference 3. To provide additional conservatism and margin, Entergy also put forward the concept of an interim license condition that would impose an increase in the safety limit minimum critical power ratio for extended power uprate. Attachment 4 provides a re-statement of the proposed license condition.

There are no new regulatory commitments contained in this submittal.

This supplement to the license amendment request provides additional information to clarify Entergy's application for a license amendment and does not change the scope or conclusions in the original application, nor does it change Entergy's determination of no significant hazards consideration.

Entergy stands ready to support the NRC staff's review of this submittal and suggests meetings at your earliest convenience to resolve any remaining issues. If you have any questions or require additional information, please contact Mr. James DeVincentis at (802) 258-4236.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on September 28, 2005.

Sincerely,

Jay K. Thayer

Jay K. Thayer
Site Vice President
Vermont Yankee Nuclear Power Station

Attachments (4)

cc: (see next page)

cc: Mr. Richard B. Ennis, Project Manager
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Mail Stop O 8 B1
Washington, DC 20555

Mr. Samuel J. Collins (w/o attachments)
Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
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USNRC Resident Inspector (w/o attachments)
Entergy Nuclear Vermont Yankee, LLC
P.O. Box 157
Vernon, Vermont 05354

Mr. David O'Brien, Commissioner (w/o proprietary information)
VT Department of Public Service
112 State Street – Drawer 20
Montpelier, Vermont 05620-2601

BVY 05-088
Docket No. 50-271

Attachment 2

Vermont Yankee Nuclear Power Station

Proposed Technical Specification Change No. 263 – Supplement No. 35

Extended Power Uprate – Additional Information

Response to RAI SRXB-A-68

NON-PROPRIETARY VERSION

Total number of pages in Attachment 1 (excluding this cover sheet) is 17.

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION REGARDING APPLICATION FOR EXTENDED POWER UPRATE LICENSE AMENDMENT VERMONT YANKEE NUCLEAR POWER STATION

PREFACE

This attachment provides a response to the NRC Reactor Systems Branch's (SRXB) request for additional information (RAI) SRXB-A-68 in NRC's letter dated September 7, 2005.[1] Upon receipt of the RAI, discussions were held with the NRC staff to further clarify the RAI. The intent of the RAI was clarified during these discussions, and the information provided herein is consistent with those clarifications.

The RAI is re-stated as provided in NRC's letter of September 7, 2005.

RAI SRXB-A-68

RAI SRXB-A-51 asked that Entergy provide an evaluation that demonstrates that the void reactivity coefficients are applicable and are developed for the range of core thermal-hydraulic conditions expected for the transient and accident conditions, including anticipated transients without scram (ATWS). The RAI response did not explicitly address the NRC staff question. The response instead discussed the conservative axial power distribution that is assumed (HBB and UB) that minimizes the scram reactivity worth. However, the staff RAI was focused on assessing ODYN's capability to simulate the change in core reactivity with the change in voids for the current EPU fuel and core designs. In addition, the objective of the RAI is also to determine if the void reactivity coefficient bias and uncertainty derived in the original ODYN licensing topical report remains valid and applicable for the EPU core and fuel designs.

The RAI response also referred to a sensitivity study performed during the initial ODYN licensing (NEDO-24154P-A, Volume III, page Q12) based on the Peach Bottom turbine trip transient simulation. The void coefficient was changed by [[ ]]. The sensitivity studies determined the impact changes in the void coefficient would have on the ΔCPR/ICPR response. The document concludes that a model uncertainty due to void reactivity response of [[ ]] is assumed. This sensitivity study [[ ]] It is also not clear that the void reactivity coefficients for the current fuel and core design are [[

[1] U.S. Nuclear Regulatory Commission (Richard B. Ennis) letter to Entergy Nuclear Operations, Inc. (Michael Kansler), "Request for Additional Information – Extended Power Uprate, Vermont Yankee Nuclear Power Station (TAC No. MC0761)," September 7, 2005

]]

As stated in the RAI response, it is true that ODYN AOO response is [[ ]] than TRACG. While TRACG applies a [[

]] Therefore, for the current EPU high energy core designs and the associated core thermal-hydraulic conditions, an uncertainty analyses is necessary in order to access the code's capability to model the changes in the core reactivity changes with changes in the void fractions.

The response to the staff's RAI 38 of the initial ODYN licensing topical report (NEDO-241154P-A, Volume 1) provides a void reactivity coefficient uncertainty analysis. The lattice k_{∞} values at the three void fractions of [[

]] The following questions relate to the appropriateness of [[ ]] used in deriving the uncertainties and biases associated with the void reactivity coefficients.

a) Provide an uncertainty analyses of the changes in the core reactivity with changes in the void fractions. Include in the uncertainty analyses how the adequacy of ODYN's predications of the reactivity coefficients can be assessed for the current EPU fuel/ core designs and operating strategy.

b) The lattice void reactivity coefficient is [[ ]] Justify the use of [[ ]] for the derivation of the uncertainties for high void conditions.

c) Provide plots showing the linear void reactivity coefficient function extended to the higher void conditions for limiting lattices in your uncertainty analysis. Include plots providing the void coefficient changes with depletion at different void conditions for the full range of instantaneous void fractions. Evaluate the changes seen in the void coefficient values with the historical void fractions for the range of the instantaneous void fractions, using limiting GE14 lattices. Based on these plots, explain the void coefficient uncertainties that would be associated with the higher void conditions for the different historical void fraction cases.

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d) The response to the staff's RAI 12 of NEDE-24154P-A (page Q12-4) Volume II states that a void coefficient uncertainty of [[ ]] is applied as presented in RAI 38 (Volume I). However, the response to question 38 (page Q38-4) states that, [[

]] Explain this statement and state if any uncertainty is applied to the void coefficient in ODYN. If so, justify why the void coefficient calculational method currently employed in ODYN, if any, is [[ ]] for the core thermal-hydraulic conditions EPU boiling water reactors (BWRs) would experience and justify the uncertainties currently used in ODYN.

e) Provide a discussion of how the changes in the void coefficient uncertainties as seen from the lattice data would affect the different transient events, instability and ATWS response.

Response to RAI SRXB-A-68

Part (a)

See the response to part (d) below.

Part (b)

A void reactivity coefficient is not input as a linear function into ODYN. ODYN uses cross sections that are fit as a quadratic function of moderator density for each control state at each axial height as described in NEDO-24154-A, Vol. I, p. 5-11. The kinetics model diffusion parameters (Σ) are provided as quadratic functions of relative water density (u) as shown below for each control state at each axial node.

$$\Sigma = \Sigma_0\left(1 + a(u-u_0) + b(u-u_0)^2\right)$$

Σ_0 = basestate diffusion cross section

a = linear coefficient

b = quadratic coefficient

u = average relative water density = ρ/ρ_{ref}

u_0 = basestate relative water density

The fitting process utilizes the cross section parameter at the basestate (steady-state) relative water density and the parameter at several other relative water densities chosen to cover the

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expected range of *u* variation. The raw cross section data is defined by the TGBLA/PANACEA database. The nodal diffusion parameters will change as the relative water density varies during the transient. The resultant change in reactivity versus the water density (or void fraction) change can be interpreted as the ODYN void coefficient.

To further define high void fraction, the VYNPS (Vermont Yankee Nuclear Power Station) Cycle 25 transient results were reviewed to determine the time=0 ODYN predicted 1D axial void fraction. The maximum exit void fraction considering the variation in axial power shapes and core flow was [[ ]]. An exit void fraction of [[ ]] occurs for both CLTP/minimum core flow and EPU/minimum core flow since the void fraction is relatively constant along a rod line. The ODYN predicted axial void fraction remains essentially unchanged at EPU conditions.

Part (c)

VYNPS is applying TGBLA06 methodology in core design, transient analysis, stability analysis, and monitoring. Figure SRXB-A-68-1 provides TGBLA06 void coefficient data and Figure SRXB-A-68-2 provides the corresponding MCNP data for 5 representative 10x10 lattices for the full range of instantaneous void (IV) conditions. The calculations are based on a 40% void history (VH) depletion followed by branch calculations at 0, 40, and 70% IV. The results are extrapolated above 70% IV. In Table SRXB-A-68-1, 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 [[ ]]. Table SRXB-A-68-1 shows the TGBLA06 vs. MCNP data at 70% IV. Table SRXB-A-68-2 provides the 5 lattice details for selected exposures (selection discussed later). The average uncertainty at 70% IV in Table SRXB-A-68-1 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 68-1) is [[ ]].

The bias and uncertainty above 70% IV has two potential issues:

3. The void coefficient data in Figure SRXB-A-68-1 and SRXB-68-2 is [[

]]

4. The data that is utilized to develop the cross section parameters is based on instantaneous void branch cases from a [[ ]]. Upper axial nodes are operating at [[

]]

The following additional analyses have been performed for Vermont Yankee lattice 7009. MCNP calculations have been performed from 40% void history, 70% void history, and 90%

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

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]]

In summary, for transient applications that utilize TGBLA06 based modeling (PANAC11, ODYN, and ODYSY) the evaluation discussed above for [[ ]] void fraction (Table SRXB-A-68-1) 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 and is applied in the response to part (d) below.

Part (d)

As documented in Reference 68-1, the uncertainty in the ΔCPR/ICPR calculated by ODYN is determined by comparison of predictions with reactor data. The basis is the same as that used in NEDO-24154-A. The reactor data used for determining the uncertainty are the [[ ]]. To verify this model uncertainty is reasonable, a [[ ]] was performed on key parameters at a bounding value judged to be at the 2σ level including the void coefficient. The results from this study documented in Reference 68-1 showed that the model uncertainty based on the model perturbation analysis supports the model uncertainty determined from the comparison to plant data. It was concluded that the approved model uncertainty process is sufficient to account for void coefficient uncertainty along with the uncertainty in other nuclear and model parameters.

The model uncertainty [[ ]] was also updated with the latest TGBLA06 / PANAC11 methods following the approved process. With the updated model uncertainty the statistical adders were also updated. These were provided to the NRC in Reference 68-2.

An analysis was performed for VYNPS Cycle 25 to quantify the sensitivity of this core to void coefficient. The ΔCPR/ICPR uncertainty based on perturbations with a 2σ uncertainty of [[ ]] is approximately [[ ]]. This sensitivity is consistent with the sensitivity provided in Reference 68-1 [[

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]] (see Reference 68-1 Table 3-2). When combined with the other uncertainties in Reference 68-1 Table 3-2, the total uncertainty from the analytical perturbation analysis is negligibly impacted [[ ]].

Part (e)

These results indicate that the void coefficient uncertainty is not substantially different at the void fractions expected for EPU conditions. The data shows no evidence of new uncertainties that would invalidate the qualification basis for models applied to transient, ATWS, or stability analysis. The void reactivity coefficient bias and uncertainty derived in the original ODYN licensing topical report remains valid and applicable for the EPU core and fuel designs. The void reactivity coefficients are applicable for the range of core thermal-hydraulic conditions expected for the transient and accident conditions, including ATWS.

References:

68-1 "Revised Supplementary Information Regarding Amendment 11 to GE Licensing Topical Report NEDE-24011-P-A," MFN-003-86, January 1986

68-4 "ODYN Statistical Adders Update," FLN-2000-014, September 22, 2000

68-5 "TRACG Application for Anticipated Operational Occurrence Transient Analysis," NEDE-32906P-A, Revision 1, April 2003

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[[

[3]]]

Figure SRXB-A-68-1
Void Coefficient Averaged for 5 10x10 Lattices at Exposures of 0,5,10,15,20,25,30,50,70 GWd/ST – TGBLA06

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[[

[3]]]

Figure SRXB-A-68-2
Void Coefficient Averaged for 5 10x10 Lattices at Exposures of 0,5,10,15,20,25,30,50,70 GWd/ST – MCNP

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Table SRXB-A-68-1
Void Coefficient Comparison between TGBLA06 and MCNP for 5 10x10 Lattices at 70% IV

[[				
				]]

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Table SRXB-A-68-2
Void Coefficient Comparison between TGBLA06 and MCNP for 5 10x10 Lattices Details at 70% IV (10, 15, & 25 GWd/ST)

[[			
			]]

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Table SRSB-A-68-3
Void Coefficient Comparison between TGBLA06 and MCNP for Lattice 7009 at ≥ 70% IV

[[			
			]]

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Table SRSB-A-68-3
Void Coefficient Comparison between TGBLA06 and MCNP for Lattice 7009 at ≥ 70% IV

[[			
			]]

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[[

]]

Figure SRXB-A-68-3

[[ ]]

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[[

]]

Figure SRXB-A-68-4

[[ ]]

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Table SRXB-A-68-4
TRACG Impact of High Exposure Void Coefficient Bias

Parameter (*)	Base	High Exposure Biased	** % Difference
[[			
			]]

* LRNBP is Generator Load Rejection without Bypass
MSIVF is MSIV Closure with Flux Scram

** % Difference is defined as ((High Exp Biased – Base) / (High Exp Biased)) X 100

[[

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