ENCLOSURE 2

M180100

Proposed Amendment 47 to GESTAR II Main Sections 1 and 4 with Bases Summary

Non-Proprietary Information

IMPORTANT NOTICE

This is a non-proprietary version of Enclosure 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by white space with an open and closed bracket as shown here [[ ]]

GESTAR II Modifications to Support TSTF-564 Implementation – Bases Discussion

The methodology in NEDC-32601P-A and NEDC-32694P-A (References 1 and 2) defines the statistical Safety Limit Minimum Critical Power Ratio (SLMCPR) as ensuring that 99.9% of the fuel rods in the core would not be susceptible to transition boiling, which is the current approved SLMCPR methodology in GESTAR II Rev. 27. However, there are elements of the analysis process and scope that are defined in GESTAR II. In addition, there are process details and variations that have been documented via various plant-specific License Amendment Requests (LARs) to change the Technical Specification (TS) SLMCPR.

Amendment 47 will bring all necessary items from these LARs into GESTAR II. This is appropriate and necessary because plants implementing TSTF-564 (Reference 3) that defines an alternate TS SLMCPR as ensuring that there is a 95% probability at a 95% confidence level that no fuel rods in the core would be susceptible to transition boiling will need to have the applied methodology fully defined and approved via a TS reference. This is because plants implementing TSTF-564 will place the statistical SLMCPR using Reference 1 and 2 methodology as supplemented by the Amendment 47 GESTAR II changes in their Core Operating Limits Report (COLR). The NRC approval of TSTF-564 will be referenced in GESTAR II.

Discussion

The changes and additions to GESTAR II fall into three categories:

1) Process elements that are added to GESTAR II, Section 1.1, Fuel Licensing Acceptance Criteria, and
2) Methodology restrictions that are addressed for each product line. These will be added to GESTAR II, Section 4.3.1.1, Fuel Cladding Integrity Safety Limit, and
3) Deviations from Reference 1 uncertainties which are addressed as differences from the approved methodology. These uncertainties are added to GESTAR II, Section 4.3.1.1, Fuel Cladding Integrity Safety Limit.

In addition, specific notes have been added to Section 1.1.5 and duplicated in Section 4.3.1 to explain the changes in terminology and process when a plant implements TSTF-564.

Section 1.1 Fuel Licensing Acceptance Criteria

Sections 1.1.5 and 1.2.5

Notes have been added to Section 1.1.5 to explain the changes in terminology and process when a plant implements TSTF-564.

Sections 1.1.5 and 1.2.5 describe the requirements for performing the cycle specific SLMCPR calculation. The power/flow (P/F) statepoints to be analyzed were expanded following the Reference 4 conclusions. Further, the MELLLA+ Topical Report (TR) (Reference 5) is specific about the P/F statepoints to be analyzed. Specific P/F statepoint guidance is added to Section 1.2.5 Subsection viii. The P/F statepoint guidance is organized by the non-MELLLA+ and MELLLA+ domain requirements.

The Double-Hump (DH) bias and uncertainty is applied in the determination of the cycle specific SLMCPR when a DH shape is identified. The requirement to apply the DH biases is included in Section 1.1.5 Subsection vix. The methodology used to identify a DH power shape is defined in Section 1.2.5 Subsection vix.

Sections 1.1.7 and 1.2.7

Sections 1.1.7 and 1.2.7 pertain to the GEXL correlation development for each fuel product line. Because the DH bias and uncertainty are product line specific, the requirement to consider the potential for a DH power shape and document the bias and uncertainty is added to Section 1.1.7 Subsection D. The methodology for determining the DH bias and uncertainty is added to Section 1.2.7 Subsection D. The methodology is similar to that described in Reference 6 but is defined more completely and using a generic process that can be applied to each fuel product line. The DH bias and uncertainty values will be included in the GESTAR II compliance report for each product line.

Section 1.5

References 1-16 to 1-20 have been added to support the new content.

Section 4.3.1

The notes from Section 1.1.5 were replicated to explain the changes in terminology and process when a plant implements TSTF-564. This is added to Section 4.3.1 to ensure the information regarding the differences is not missed.

Section 4.3.1.1

Unrelated to TSTF-564, the statements about the uncertainties from Section 4.3.1.1.1, Statistical Model, were moved to 4.3.1.1.2, BWR Statistical Analysis, because it fit the subject better. The second sentence of this moved statement was modified to require confirmation of the uncertainties or use larger values if elected by the plant.

Section 4.3.1.1.3, Methodology Restrictions, was added to include the contents of the section of the same name in the standard plant-specific Safety Limit MCPR LARs. Within this section, the evaluation and response to the restrictions for GE14, GNF2, and GNF3 are included by reference.

Section 4.3.1.1.4, Deviations from Reference 1 Uncertainties, was added to include the contents of the section of a similar name in the plant-specific Safety Limit MCPR LARs. Within this section the following subjects are covered: R-Factor, Core Flow Rate and Random Effective TIP Reading, and Flow Area Uncertainty. The Flow Area Uncertainty value will be included in the GESTAR II compliance report for each product line.

Section 4.4

References 4-43 to 4-51 have been added to support the new content.

References

1 "Methodology and Uncertainties for Safety Limit MCPR Evaluation," NEDC–32601P–A, August 1999.

2 "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC–32694P–A, August 1999.

3 Letter from Technical Specifications Task Force to Document Control Desk (NRC), Subject: Transmittal of TSTF-564, "Safety Limit MCPR," TSTF-17-12, PROJ0753, August 28, 2017.

Letter from Technical Specifications Task Force to Document Control Desk (NRC), Subject: TSTF Response to NRC Questions on TSTF-564, "Safety Limit [Minimum Critical Power Ratio] MCPR", TSTF-18-05, PROJ0753, May 29, 2018.

The final approval of the TSTF will be referenced in GESTAR II.

4 Letter from Jason Post (GE Nuclear) to Document Control Desk (NRC), Subject: Part 21 Final Report: Non-Conservative SLMCPR, MFN 04-108, September 29, 2004.

5 "General Electric Boiling Water Reactor, Maximum Extended Load Line Limit Analysis Plus," NEDC-33006P-A, Revision 3, June 2009.

6 Letter, Glen A. Watford (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Alan Wang (NRC), "NRC Technology Update – Proprietary Slides – July 31 – August 1, 2002", FLN-2002-015, October 31, 2002.

1. Introduction

This report presents generic information relative to the fuel design and analyses of General Electric Boiling Water Reactor plants for which General Electric provides fuel. The report consists of a description of the fuel licensing criteria and fuel thermal–mechanical, nuclear, and thermal–hydraulic analyses bases. This report provides information and methods used to determine reactor limits that are independent of a plant–specific application. Plant–specific information and the transient and accident methods used are given in the country–specific supplement accompanying this base document.

The generic information contained in this report is supplemented by plant cycle–unique information and analytical results. This cycle–unique information includes a listing of the fuel to be loaded in the core and safety analysis results. This information is documented in the plant FSAR for initial core loadings and in a separate plant–unique cycle–dependent report for each reload. The format for this *Supplemental Reload Licensing Report* is given in Appendix A of the country–specific supplement to this document. Fuel bundle design information for the specific fuel bundles used for each cycle is given in the *Fuel Bundle Information Report* (FBIR). The format for the FBIR is given in Appendix A of the country–specific supplement to this document.

Proposed changes to this document are submitted to the appropriate regulatory body for review and approval. A listing of NRC approved amendments is given in the GESTAR II Revision Status Sheet located in the front of this document. The latest approved changes are incorporated as a revision into the text and indicated by change bars in the margin.

1.1 Fuel Licensing Acceptance Criteria

A set of fuel licensing acceptance criteria have been established for evaluating new fuel designs and for determining the applicability of generic analyses to these new designs. Fuel design compliance with the fuel licensing acceptance criteria constitutes USNRC acceptance and approval of the fuel design without specific USNRC review. The fuel licensing acceptance criteria are presented in the subsections that follow.

Fuel designs that have received specific USNRC review and approval or that have been shown to meet the fuel licensing acceptance criteria are documented in References 1–1 and 1–2. A detailed description of the 8x8 and 8x8R fuel designs is given in Reference 1–1 while the newer designs are described in Reference 1–2. Since the approval of GESTAR II Amendment 22 in 1990, a compliance report, sometimes called Compliance with Amendment 22 of GESTAR II, has been produced for each fuel product line. Section 1.4 provides the compliance reports for each fuel product line. Fuel bundle design information for bundles more recent than those included in Reference 1-2 is found in the plant-cycle specific FBIR.

The fuel licensing acceptance criteria are as follows.

1.1.1 General Criteria

A. NRC–approved analytical models and analysis procedures will be applied.

B. New design features will be included in lead use assemblies.

C. The generic post–irradiation fuel examination program approved by the NRC will be maintained (References 1–3 and 1–4).

D. New fuel related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval.

E. If any of the criteria in Subsection 1.1 are not met for a new fuel design, that aspect will be submitted for review by the NRC separately.

1.1.2 Thermal–Mechanical

A. The fuel design thermal–mechanical analyses are performed for the following conditions:

i. Either worst tolerance assumptions are applied or probabilistic analyses are performed to determine statistically bounding results (i.e. upper 95% confidence).

ii Operating conditions are taken to bound the conditions anticipated during normal steady–state operation and anticipated operational occurrences.

B. The fuel design evaluations are performed against the following criteria.

i. The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability.

ii. Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material.

iii. The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components.

iv. The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards C776–83 and C934–85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding.

v. The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition.

vi. Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.

vii. The fuel assembly (including channel box), control rod and control rod drive are evaluated to assure control rods can be inserted when required. These evaluations are performed in accordance with NUREG–0800 (Appendix A to SRP Section 4.2) where the effect of combined Safe Shutdown Earthquake (SSE) and Loss–of–Coolant Accident (LOCA) loads (which conservatively bound the worst case hydraulic loads possible during normal conditions) are evaluated to assure component deformation is not severe enough to prevent control rod insertion and vertical liftoff forces will not unseat the lower tie–plate such that the resulting loss of lateral fuel bundle positioning would prevent control rod insertion.

viii. Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap.

ix. Loss of fuel rod mechanical integrity will not occur due to fuel melting.

x. Loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction.

A detailed description of the thermal–mechanical bases currently in use in the US is given in Section 2. There were significant changes to the thermal-mechanical design bases in GESTAR II Revision 17. Therefore, the thermal-mechanical design bases for older fuel products are as defined in versions of GESTAR II prior to Revision 17. The bases for older fuel products are applicable to the bundle designs described in Reference 1–2. Reference 1–1 provides a description of the thermal–mechanical bases used for the 8x8 and 8x8R fuel designs. The compliance reports included in Section 1.4 reference the relevant GESTAR II revision for each respective product line.

1.1.3 Nuclear

A. A negative Doppler reactivity coefficient shall be maintained for any operating conditions.

B. A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels shall be maintained for any operating conditions.

C. A negative moderator temperature coefficient shall be maintained for temperatures equal to or greater than hot standby.

D. For a super prompt critical reactivity insertion accident (e.g., control rod drop accident) originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative.

E. A negative power coefficient, as determined by calculating the reactivity change due to an incremental power change from a steady–state base power level, shall be maintained for all operating power levels above hot standby.

F. The plant shall be calculated to meet the cold shutdown margin requirement for each plant cycle specific analysis.

G. The effective multiplication factor for new fuel designs stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k–infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 for GE designed regular or high density storage racks.

Nuclear analyses that are performed for each individual fuel project are documented in Section 3.

1.1.4 Hydraulic

A. Flow pressure drop characteristics shall be included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.

Thermal–hydraulic analyses that are performed for each individual fuel project are documented in Section 4.

1.1.5 Safety Limit MCPR

Notes:

Plants Adopting TSTF-564

For plants that have adopted TSTF-564, the Technical Specification Safety Limit MCPR is cycle-independent as described in Reference 1-18. TSTF-564 uses the term $SLMCPR_{95/95}$ to define the cycle-independent safety limit that will be applied in Technical Specification (TS) 2.1.1.2. (This TS Section reference is used in the TSTF and is based on the Standard Technical Specifications. Specific plants may have a different TS section for the Safety Limit MCPR.) The cycle specific SLMCPR is termed $MCPR_{99.9\%}$ in TSTF-564 and will be included in the cycle-specific Core Operating Limits Report (COLR). The following table summarizes the CPR terminology.

$MCPR_{95/95}$	Cycle independent value determined based on the GEXL correlation statistics using the expression defined in TSTF-564
$SLMCPR_{95/95}$	Cycle-independent Technical Specification Safety Limit
$MCPR_{99.9\%}$	Cycle-specific COLR SLMCPR

There is no change in the methodology used to calculate the $MCPR_{99.9\%}$. The cycle-specific SLMCPR methodologies remain as described in this section, in Section 1.2.5, and in Section 4.3.1.1.

The proposed $MCPR_{95/95}$ values for fuel product lines GE14, GNF2, and GNF3, which may be used to define the $SLMCPR_{95/95}$, are included in Table 1 of TSTF-564. Section 3.1 of TSTF-564 describes the methodology to be used in the development of the $MCPR_{95/95}$. For new fuel products, GNF will provide the NRC a letter like Reference 1 of TSTF-564, which may be referenced by a licensee requesting a change to $SLMCPR_{95/95}$ in their Technical Specifications.

Historically, the term SLMCPR has been used for the statistical limit defined by the approved methodology (References 1-19 and 1-20). This term is used in GESTAR II, the SLMCPR methodology documents, and in numerous reports. GNF does not intend to change any previous usage.

Plants Not Adopting TSTF-564

For plants that have "not" adopted TSTF-564, the Technical Specification Safety Limit MCPR will remain as the cycle-specific SLMCPR described in this section, in Section 1.2.5, and in Section 4.3.1.1.

A. A cycle–specific Safety Limit MCPR will be calculated on a cycle–specific basis following the steps in 1.1.5.B.

B. Cycle–specific Safety Limit MCPR calculations will be performed under the following conditions.

   i. Analysis shall be performed for the specific plant.

   ii. Analysis shall be performed for the specific core loading and the specific bundle design.

   iii Core radial power distributions shall be selected to reasonably bound the number of bundles at or near thermal limits.

   iv. Local fuel pin power distribution shall be based on specific bundle design.

   v. Ninety–nine point nine percent (99.9%) of the rods in the core must be expected to avoid boiling transition.

   vi. Uncertainties used in the analysis shall be the same as documented in Section 4 including the uncertainty associated with the appropriate critical power correlation. The critical power correlation uncertainty used in the Safety Limit

MCPR determination shall be that uncertainty associated with the operating regions that can be obtained during normal operation or during Anticipated Operational Occurrences (AOO).

vii. Analyses are performed for multiple exposure points throughout the cycle. Typically the most limiting value is applied over the entire cycle, but exposure–dependent values may be applied.

viii. Analyses are performed at selected core power/flow points consistent with the licensed domain boundary.

vix. Increased bias and uncertainty is applied when a Double-Hump (DH) power shape is identified during the determination of the cycle–specific Safety Limit MCPR.

A discussion of the statistical analyses used to derive the cycle–specific Safety Limit MCPR is presented in Section 4.

1.1.6 Operating Limit MCPR

A. Plant Operating Limit MCPR is established by considering the limiting anticipated operational occurrences for each operating cycle. This may be calculated as a function of exposure.

B. For each new fuel design the applicability of generic MCPR analyses described in Section 4 or in the country–specific supplement to this base document shall be confirmed for each operating cycle or a plant specific analysis will be performed.

AOO descriptions and evaluation methodologies and procedures used to derive the Operating Limit MCPR are presented in Section 4 and in the country–specific supplement to the base document.

1.1.7 Critical Power Correlation

A. The currently approved critical power correlations will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design.

B. A new correlation may be established if significant new data exists for a fuel design(s).

C. The criteria for establishing the new correlation are as follows.

i. The new correlation shall be based on full–scale prototypical test assemblies.

ii. Tests shall be performed on assemblies with typical rod–to–rod peaking factors.

iii. The functional form of the currently approved correlations shall be maintained.

iv. Correlation fit to data shall be best fit.

v. One or more additional assemblies will be tested to verify correlation accuracy (i.e., test data not used to determine the new correlation coefficients).

vi. Coefficients in the correlation shall be determined as described in References 1–5 or 1–6.

vii. The uncertainty of the resulting correlation shall be determined by:

$$\sigma^2 = \frac{1}{N-1}\sum_{i=1}^{N}(\mu - ECPR_i)^2$$

where:

σ = standard deviation.

μ = $\frac{1}{N}\sum_{i=1}^{N} ECPR_i$

N = Total number of data in both the data set used to determine the coefficients and the set used for verification.

$ECPR$ = Calculated bundle critical power divided by experimentally determined bundle critical power.

D. DH axial power shapes may exist in cycle core designs. The product line critical power correlations developed using the process defined in Section 1.1.7 and 1.2.7, Subsection C, have historically been known to be non-conservative for DH shapes, therefore specific analyses are used to estimate a bounding effect on the bias and uncertainty.

1.1.8 Stability

New fuel designs must satisfy either criterion A or B below:

A. The stability behavior, as indicated by core and limiting channel decay ratios, must be equal to or better than a previously approved GE BWR fuel design.

B. If the core and limiting channel decay ratios are not equal to or better than a previously approved GE fuel design, it must be demonstrated that there is no change to the exclusion zone.

1.1.9 Overpressure Protection Analysis

A. Adherence to the ASME overpressure protection criteria shall be demonstrated on plant cycle specific analysis.

A discussion of evaluations performed to demonstrate compliance with overpressure limits is presented in the country–specific supplement to this document.

1.1.10 Loss–of–Coolant Accident Analysis Methods

A. The criteria in 10CFR50.46 shall be met on plant specific or bounding analyses.

B. Plant MAPLHGR adjustment factors must be confirmed when a new fuel design is introduced.

Specific LOCA evaluation methodologies are discussed in the country–specific supplement to this base document.

1.1.11 Rod Drop Accident Analysis

A. Plant cycle specific analysis results shall not exceed the licensing limit described in the country specific supplement to this base document.

B. Applicability of the bounding BPWS analysis must be confirmed.

Discussions of plant specific and generic rod drop accident evaluation methodologies are presented in the country–specific supplement to this base document.

1.1.12 Refueling Accident Analysis

A. The consequences of a refueling accident as presented in the country–specific supplement to this base document or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed (using the methods and assumptions described in the country supplement) and documented when a new fuel design is introduced.

1.1.13 Anticipated Transient Without Scram

The fuel must meet either criteria A or B below:

A. A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in References 1–7 and 1–8, shall be maintained for any operating conditions above the startup critical condition.

B. If criterion 1.1.13.A is not satisfied, the limiting events (as described in References 1–7 and 1–8) will be evaluated to demonstrate that the plant response is within the ATWS criteria specified in References 1–7 and 1–8.

1.1.14 Fuel Loading Error (FLE) Event Analysis

Section S.5.3 of the country–specific supplement presents the requirements for analyzing the FLE (misloaded or misoriented fuel bundle) as an Infrequent Incident. Should a plant not meet the requirements in Section S.5.3, the event will be analyzed as an AOO.

A. As an Infrequent Incident, the FLE events are subject to the radiological limits of 10% of 10CFR100, or 10% of 10CFR50.67 for Alternate Source Term plants. Bounding radiological analysis of these events is referenced in the country–specific supplement to this base document.

B. As an AOO, the FLE events are subject to the MCPR criteria. (See Section 1.1.5 and 1.1.6)

1.2 Basis for Fuel Licensing Criteria

The following provides the basis for the criteria documented in Subsection 1.1.

1.2.1 General Criteria

A. *NRC–approved analytical models and analysis procedures will be applied.*

Consistent with current practice, NRC–approved procedures and methods are used to evaluate new fuel designs.

B. *New design features will be included in lead use assemblies.*

GE's "test before use" fuel design philosophy includes irradiation experience with new fuel design features in full–scale fuel assemblies (Lead Use Assemblies) in operating reactors prior to standard reload application. A method for licensing LUAs and the NRC acceptance of this method are documented in References 1–9 and 1–10, respectively.

GNF proposed in Reference 1-14 an enhanced lead use program for the use of channels made of the niobium-tin-iron (NSF) zirconium alloy. The US NRC has reviewed and approved the program by Reference 1-15. This program allows NSF Lead Use Channels (LUC) to be used in quantities up to 8% of the total number of channels in the core. The NSF LUC limit of 8% is exclusive of other lead assembly programs. In other words, other lead use programs are not affected and continue to be allowed up to the ~2% limit of GESTAR II.

C. *The generic post–irradiation fuel examination program approved by the NRC will be maintained.*

Section 4.2.II.D.3 of the SRP requires each plant to implement a post–irradiation fuel surveillance program to detect anomalies or to confirm expected fuel performance. The NRC has found (Reference 1–3) that the GE fuel surveillance program (Reference 1–4) is an acceptable means for licensees to satisfy the post–irradiation surveillance requirement of the SRP. The GE program includes examination of LUAs and selected discharge bundles with the results reported to the NRC in a yearly operating experience report.

D. *New fuel related licensing issues identified by the NRC will be evaluated to determine if the current criteria properly address the concern; if necessary, new criteria will be proposed to the NRC for approval*.

New licensing concerns related to fuel design and performance may arise after the establishment of approved fuel licensing acceptance criteria. Upon identification of a new issue by the NRC, GE will evaluate the concern against the established criteria to determine if this issue can be resolved through the application of approved criteria. If the current criteria does not adequately address the identified concern, GE will propose a new criterion (criteria) to the NRC for review and approval.

E. *If any of the criteria in Subsection 1.1 are not met for a new fuel design, that aspect will be submitted for review by the NRC separately*.

If a new fuel design does not meet one of the criteria in Subsection 1.1, it does not mean this design is unacceptable. It simply means the design has gone beyond the generic approval and must be reviewed.

1.2.2 Thermal–Mechanical

A. *The fuel design thermal–mechanical analyses are preformed for the following conditions*:

i. *Either worst tolerance assumptions are applied or probabilistic analyses are performed to determine statistically bounding results (i.e. upper 95% confidence)*.

ii. *Operating conditions are taken to bound the conditions anticipated during normal steady–state operation and anticipated operational occurrences*.

These analyses are performed generically for each new fuel design or previous analyses are determined to be applicable.

B. *The fuel design evaluations are performed against the following criteria:*

i. *The fuel rod and fuel assembly component stresses, strains, and fatigue life usage shall not exceed the material ultimate stress or strain and the material fatigue capability.*

The fuel rod and assembly components are evaluated to ensure that the fuel will not fail due to stresses or strains exceeding the fuel assembly component mechanical capability. The limit is patterned after ANSI/ANS–57.5–1981. The figure of merit employed is the Design Ratio where:

$$Design\ Ratio = \frac{Effective\ Stress}{Stress\ Limit} \quad or \quad \frac{Effective\ Strain}{Strain\ Limit}$$

The material capability limit is taken as the material ultimate stress or strain. The limit used is that the Design Ratio must be less than or equal to one (Design Ratio ≤ 1.0). Fatigue is addressed in a similar manner where the calculated fatigue duty must be less than the material fatigue capability (Fatigue Life Usage ≤ 1.0). A more detailed discussion of the stress/strain and fatigue bases, limits, and evaluations is presented in Subsections 2.2.1.1 and 2.2.1.2.

ii. *Mechanical testing will be performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear when operating in an environment free of foreign material*.

Evaluations of the fuel assembly for fretting wear are based on mechanical testing and extensive reactor operating experience. A more detailed discussion of the fretting wear evaluation methodology is presented in Subsection 2.2.1.3.

iii. *The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these effects influence the material properties and structural strength of the components*.

The effects of cladding oxidation and corrosion product buildup on the fuel rod surface (i.e., increased calculated temperatures, material property changes and cladding thinning) are explicitly included in the evaluations performed relative to criteria 1.1.2.B.i, 1.1.2.B.vi, 1.1.2.B.vii, 1.1.2.B.viii, 1.1.2.B.ix and 1.1.2.B.x.

iv. *The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards C776–83 and C934–85 to assure that loss of fuel rod mechanical integrity will not occur due to internal cladding hydriding*.

Internal cladding hydriding is controlled during fuel manufacture by restricting the level of moisture and other hydrogenous impurities within limits consistent with SRP 4.2. Extensive operating experience with fuel designs manufactured to the hydrogen content limits specified in the SRP demonstrate that hydriding is not an active failure mechanism for normal operation or AOOs.

v. *The fuel rod is evaluated to ensure that fuel rod or channel bowing does not result in loss of fuel rod mechanical integrity due to boiling transition*.

As part of the GE Fuel Surveillance Program and other inspections, the peripheral row of fuel rods is visually inspected to determine the extent of fuel rod–to–fuel rod gap closure due to rod bowing caused by fuel rod growth. Observations of gap closure greater than 50% are reported to the NRC. Any changes to the 50% closure requirement will be based on thermal–hydraulic testing to assure that the criterion is satisfied.

The effect of potential channel bow on fuel rod/bundle performance and critical power margins is accounted for by adjusting R–factor values in the plant process computer databank.

vi. *Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading*.

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]] A more detailed discussion of the fuel rod internal pressure evaluation is presented in Subsection 2.2.1.6.

vii. *The fuel assembly (including channel box), control rod and control rod drive are evaluated to assure control rods can be inserted when required. These evaluations are performed in accordance with NUREG–0800 (Appendix A to SRP Section 4.2) where the effect of combined Safe Shutdown Earthquake (SSE) and Loss–of–Coolant Accident (LOCA) loads (which conservatively bound the worst case hydraulic loads possible during normal conditions) are evaluated to assure component deformation is not severe enough to prevent control rod insertion and vertical liftoff forces will not unseat the lower tie–plate such that the resulting loss of lateral fuel bundle positioning would prevent control rod insertion*.

A more detailed description of this evaluation is provided in Subsection 2.2.2.9.

viii. *Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap*.

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]] Subsection 2.2.2.2 provides further discussion of the cladding collapse analysis.

ix. *Loss of fuel rod mechanical integrity will not occur due to fuel melting*.

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x. *Loss of fuel rod mechanical integrity will not occur due to pellet–cladding mechanical interaction.*

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1.2.3 Nuclear

Generic analyses are performed to assure that the following criteria A through E are satisfied. These analyses are performed as follows:

1. A large BWR/4 or BWR/5 plant shall be used to perform the generic analyses.

2. The analyses shall be performed for an equilibrium core loading of the new fuel design.

3. The analyses shall be performed at the limiting points of the cycle and will cover all expected modes of operation.

Criterion F is demonstrated on a cycle specific basis for each plant. Criterion G is calculated generically for each bundle nuclear design.

A. *A negative Doppler reactivity coefficient shall be maintained for any operating conditions.*

   The Doppler reactivity coefficient is of high importance in reactor safety. The Doppler coefficient of the core is a measure of the reactivity change associated with an increase in the absorption of resonance–energy neutrons caused by a change in the temperature of the material and is a function of the average of the bundle Doppler coefficients. A negative Doppler coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on a gross or local basis and thus assures the tendency of self–control for the BWR.

B. *A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels shall be maintained for any operating conditions.*

   The core moderator void coefficient resulting from boiling in the active flow channels is maintained negative over the complete range of BWR operation. This flattens the radial power distribution and provides ease of reactor control due to the negative void feedback mechanism.

C. *A negative moderator temperature coefficient shall be maintained for temperatures equal to or greater than hot standby.*

   The moderator temperature coefficient is associated with a change in the moderating capability of the water. Once the reactor reaches the power producing range, boiling begins and the moderator temperature remains essentially constant. The moderator temperature coefficient is negative during power operation.

D. *For a super prompt critical reactivity insertion accident (e.g., control rod drop accident) originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel shall be negative.*

The mechanical and nuclear design of the fuel shall be such that the prompt reactivity feedback (requiring no conductive or convective heat transfer and no operator action) provides an automatic shutdown mechanism in the event of a super prompt reactivity incident such as a control rod drop accident. This characteristic will assure rapid termination of super prompt critical accidents with additional long–term shutdown capability provided by Criterion 1.1.3.B for those cases where conductive heat transfer from the fuel to the water results in boiling in the active channel region.

E. *A negative power coefficient, as determined by calculating the reactivity change, due to an incremental power change from a steady–state base power level, shall be maintained for all operating power levels above hot standby.*

A negative power coefficient provides an inherent negative feedback mechanism to provide more reliable control of the plant as the operator performs power maneuvers. It is particularly effective in preventing xenon initiated power oscillations in the core. The power coefficient is effectively the combination of Doppler, void and moderator temperature reactivity coefficients. For fast system transients, these three individual reactivity components are explicitly considered to determine the core transient response.

F. *The plant shall be calculated to meet the cold shutdown margin requirement for each plant cycle specific analysis.*

The core must be capable for being made subcritical with margin in the most reactive condition throughout an operating cycle with the most reactive control rod in its full out position and all other rods fully inserted. This parameter is dependent upon the core loading and is calculated for each plant cycle prior to plant operation of that cycle.

G. *The effective multiplication factor for new fuel stored under normal and abnormal conditions shall be shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k–infinity calculated in a normal reactor core configuration meets the limits provided in Section 3 for GE designed regular or high density storage racks.*

For GE designed fuel storage racks, the storage criteria are satisfied if the uncontrolled lattice k–infinity calculated in the normal reactor core configuration meets the conditions documented in Subsection 3.5.

1.2.4 Hydraulic

A. *Flow pressure drop characteristics shall be included in plant cycle specific analyses for the calculation of the Operating Limit MCPR.*

Because of the channeled configuration of BWR fuel assemblies, there is no bundle to bundle cross flow inside the core and the only issue of hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its impact on margin to thermal limits (i.e., MCPR and MAPLHGR and/or LHGR). The coupled thermal–hydraulic–nuclear analyses performed each cycle for each plant to determine fuel bundle flow and power distribution uses the various bundle pressure loss coefficients to determine the flow distribution required to maintain total core pressure drop boundary conditions to be applied to all fuel bundles. The margin to the thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

1.2.5 Safety Limit MCPR

See the Note in Section 1.1.5

A. *A cycle–specific Safety Limit MCPR will be calculated on a cycle–specific basis following the steps in 1.1.5.B.*

The Safety Limit MCPR is sensitive to bundle design parameters and associated GEXL or GEXL–PLUS critical power correlations. Bundle design parameters of particular importance are the rod diameter, thermal time constant, spacer design and bundle R–factor. Therefore, any change in the bundle design or thermal analysis correlation requires that the Safety Limit MCPR be reassessed and revised as required. The Safety Limit MCPR is recalculated or is reconfirmed each operating cycle for each plant following the steps in Subsection 1.1.5.B and is documented in the cycle–specific supplemental reload licensing report.

B. *Cycle–specific Safety Limit MCPR calculations will be performed under the following conditions.*

i. *Analysis shall be performed for the specific plant.*

ii. *Analysis shall be performed for the specific core loading and the specific bundle design.*

iii. *Core radial power distributions shall be selected to reasonably bound the number of bundles at or near thermal limits.*

iv. *Local fuel pin power distributions shall be based on specific bundle design.*

v. *Ninety–nine point nine percent (99.9%) of the rods in the core must be expected to avoid boiling transition.*

vi. *Uncertainties used in the analysis shall be the same as documented in Section 4 including the uncertainty associated with a new critical power correlation. The critical power correlation uncertainty used in the Safety Limit MCPR determination, shall be that uncertainty associated with the operating regions that can be obtained during normal operation or during anticipated operational occurrences (AOO).*

vii. *Analyses are preformed for multiple exposure points throughout the cycle. Typically the most limiting value is applied over the entire cycle, but exposure–dependent values can be applied.*

viii. Analyses are performed at selected core power/flow points consistent with the licensed domain boundary.

The analyzed power/flow points are based on the following:

1. **Non-MELLLA+ Plants with Minimum Core Flow ≥99%: Rated Core Power / Rated Core Flow**
2. **Non-MELLLA+ Plants with Minimum Core Flow <99%: Rated Core Power / Rated Core Flow, and Rated Core Power / Minimum Core Flow**
3. **MELLLA+ Plants: Rated Core Power / Rated Core Flow, Rated Core Power / Increased Core Flow, Rated Core Power / Minimum Core Flow, and Off-Rated Core Power at Minimum Core Flow on MELLLA+ boundary.**

The acronym MELLLA+ represents the Maximum Extended Load Line Limit Analysis Plus expanded operating domain (Reference 1-16). A generic power-flow operating domain illustration is shown in Figure S-5 in the US Supplement to GESTAR II. The licensed operating domain is specific to each plant.

vix. Increased bias and uncertainty is applied when a Double-Hump (DH) power shape is identified during the determination of the cycle–specific Safety Limit MCPR.

The DH power shape is identified by the following expression:

[[

]]

Higher uncertainties and non-conservative biases for the DH axial power shape could exist relative to the values based on the product line critical power correlations developed using the process defined in Section 1.1.7 and 1.2.7 Subsection C. Section 1.2.7 Paragraph D presents the methodology for determining the DH bias and uncertainty.

The cycle–specific Safety Limit MCPR is performed for each plant in accordance with commitments made to the NRC (Reference 1–11). Because the Safety Limit MCPR is highly dependent upon the core loading pattern and the actual fuel bundle design parameters, this limit is cycle dependent for each plant and may vary through the cycle. Typically, the most limiting value is applied over the entire cycle, but exposure–dependent Safety Limit MCPR values are technically correct and may be applied if necessary. The criterion that 99.9% of the rods in the core must be expected to avoid boiling transition and the uncertainties used in the analysis (except the critical power correlation uncertainty) have been approved by the NRC and are documented in Subsection 4.3.1.1. The uncertainty associated with the critical power correlation shall be determined as documented in Subsection 1.1.7.

1.2.6 Operating Limit MCPR

A. *Plant Operating Limit MCPR is established by considering the limiting anticipated operational occurrences for each operating cycle. This may be calculated as a function of exposure.*

The operating limit MCPR is determined by adding the change in the CPR for the limiting analyzed anticipated operational occurrence to the Safety Limit MCPR. The MCPR operating limit calculational procedure and descriptions of the limiting AOO events are documented, respectively, in Subsection 4.3.1.2 and in the country–specific supplement. These limiting events were established based on sensitivity studies of bundle and plant parameters. Because the operating limit MCPR is dependent upon the core loading pattern, this limit is cycle dependent for each plant and is calculated just prior to operation of the cycle.

B. *For each new fuel design the applicability of generic MCPR analyses described in Section 4 or in the country–specific supplement to this base document shall be confirmed for each operating cycle or a plant–specific analysis will be performed.*

Generic event analysis results have been calculated for the Rod Withdrawal Error. These analyses are dependent upon the fuel design for BWR 3–5 plants without ARTS and the analytical methods, and must be reconfirmed whenever there is a change in either. Currently the generic analysis for these plants is approved for fuel designs through P8x8R and BP8x8R with both GENESIS and GEMINI methods and the GEXL and GEXL–PLUS critical power correlation. Analysis for these plants with GE8x8E/EB and GE8x8NB fuel must be performed on a cycle–specific basis. The generic analyses for plants with ARTS and BWR/6 plants with enrichments less than 3.25 weight percent enrichment are applicable to fuel designs through GE8x8E/EB with GENESIS and GEMINI methods and GEXL critical power correlation. A plant cycle specific evaluation must be performed for the GE8x8E/EB fuel design with GEXL–PLUS and the GE8x8NB fuel designs until a sufficient database exists to determine the applicability of the generic analyses. Similar cycle specific analyses will be performed for new fuel designs until an adequate database exists to perform generic analyses using methods previously approved by the NRC.

For plants analyzing FLE events as an AOO, the event is performed for initial cores and reload cores where the resultant CPR response may establish the operating limit MCPR (OLMCPR).

1.2.7 Critical Power Correlation

A. *The currently approved critical power correlations will be confirmed or a new correlation will be established when there is a change in wetted parameters of the flow geometry; this specifically includes fuel and water rod diameter, channel sizing and spacer design.*

The coefficients for the critical power correlation of a fuel design will be determined generically based on the criteria documented in Subsection 1.1.7. The fuel design parameters given in these criteria are those that have the primary effect on determining the need for a new critical power correlation when there is a change in the fuel design. New coefficients for the critical power correlation will be provided in the critical power correlation report for each fuel product line.

B. *A new correlation may be established if significant new data exists for a fuel design(s).*

When significant new data have been taken for a fuel design, a better fit to the data may be achieved by adjusting the coefficients in the critical power correlation. The resulting new critical power correlation would be a more accurate representation of actual plant operation. These coefficients will be determined generically and documented in the critical power correlation report for each fuel product line.

C. *The criteria for establishing the new correlation are as follows:*

i. *The new correlation shall be based on full–scale prototypical test assemblies.*

ii. *Tests shall be performed on assemblies with typical rod–to–rod peaking factors.*

iii. *The functional form of the currently approved correlations shall be maintained.*

iv. *Correlation fit to data shall be best fit.*

v. *One or more additional assemblies must be tested to verify correlation accuracy (i.e. test data not used to determine the new correlation coefficients).*

vi. *Coefficients in the correlation shall be determined as described in References 1–5 or 1–6.*

vii. *The uncertainty of the resulting correlation shall be determined by:*

$$\sigma^2 = \frac{1}{N-1}\sum_{i=1}^{N}(\mu - ECPR_i)^2$$

where:

σ = *standard deviation.*

μ = $\frac{1}{N}\sum_{i=1}^{N} ECPR_i$

N = *Total number of data in both the data set used to determine the coefficients and the set used for verification.*

$ECPR$ = *Calculated bundle critical power divided by experimentally determined bundle critical power.*

The criteria for establishing a new correlation are those which were used in establishing the current GEXL and GEXL–PLUS correlations approved by the NRC. The basis of the correlation is a best fit of data taken of prototypical test assemblies with typical rod–to–rod peaking factors. To assure that no unreviewed safety question exists, the functional form of the current correlations must be maintained. A correlation with a different form must be approved by the NRC prior to use. The correlation coefficients and uncertainties will be determined as approved by the NRC for the current correlations.

D. DH axial power shapes may exist in cycle core designs. The product line critical power correlations developed using the process defined in Section 1.1.7 and 1.2.7, Subsection C, have historically been known to be non-conservative for DH shapes, therefore specific analyses are used to estimate a bounding effect on the bias and uncertainty.

A methodology for the determination of the increased GEXL critical power correlation bias and uncertainty is documented in Reference 1-17 and has been referenced in SLMCPR Technical Specification license amendment requests since 2002. As noted in Reference 1-17, the GNF COBRAG subchannel model is used to estimate the difference in the mean and standard deviation between COBRAG and GEXL for the DH power shape. The COBRAG model has been qualified to the data sets used to develop the GEXL correlations. The inherent assumption in the approach is that COBRAG, which calculates Boiling Transition (BT) using a mechanistic modeling of the film mass balance and dryout, reasonably predicts the sensitivity to the power shape.

An improved statistical approach, which is similar to Reference 1-17, has been developed and is documented as follows:

Because there is no data for the DH shape, the DH ECPR ($ECPR_{DH}$) is estimated using the ECPR for a reference cosine shape ($ECPR_C$) and the difference between the DH and cosine estimated by COBRAG ($dCCPR_{DH-C}$).

$$ECPR_{DH} \approx ECPR_C + dCCPR_{DH-C}$$

The reference ECPR is

$$ECPR_{C,i} = \frac{CP_{GEXL,C,i}}{CP_{Data,C,i}}$$

Where the subscript C indicates the cosine shape.

The COBRAG DH critical power is calculated for the same conditions as the cosine data point (i.e., same flow, pressure, inlet subcooling and R-factor). Only the axial power shape is changed. The difference in Calculated CPR (CCPR) due to power shape is estimated by:

$$dCCPR_{DH-C,i} = \frac{CP_{GEXL,DH,i}}{CP_{COBRG,DH,i}} - \frac{CP_{GEXL,C,i}}{CP_{COBRG,C,i}}$$

Where

$CP_{GEXL,DH,i}$ **= GEXL prediction of the CP for a DH shape at point i**

$CP_{COBRAG,DH,i}$ **= COBRAG prediction of the CP for a DH shape at point i**

$CP_{GEXL,C,i}$ **= GEXL prediction of the CP for a cosine shape at point i**

$CP_{COBRAG,C,i}$ **= COBRAG prediction of the CP for a cosine shape at point i**

The mean ECPR of the DH shape predictions is

$$< ECPR_{DH} > = \frac{1}{n}\sum_{i=1}^{i=n}(ECPR_{C,i} + dCCPR_{DH-C,i})$$

The modification of the bias for the DH power shape is based on the 95% confidence limit for $< ECPR_{DH} >$**.**

$$< ECPR_{DH} >^{95\%} = < ECPR_{c} + dCCPR_{DH-C} >^{95\%}$$

The standard deviation of $ECPR_{DH}$ **is**

$$\sigma_{DH}^{2} = \frac{1}{n-1}\sum_{i=1}^{i=n}\left(ECPR_{C,i} + dCCPR_{DH-C,i} - < ECPR_{DH} >\right)^{2}$$

Similar to the treatment of the mean, the modification of the standard deviation for the DH power shape, $\sigma_{DH}^{95\%}$**, is the upper 95% confidence limit for the standard deviation for** σ_{DH} **.**

The resulting numerical values of the DH mean and uncertainty at the upper 95% confidence limit for each fuel product line will be documented in the GESTAR II Compliance Report for that product.

1.2.8 Stability

New fuel designs must meet either criterion A or B as specified below:

These evaluations will be performed generically as specified below:

A. *The stability behavior, as indicated by core and limiting channel decay ratios, must be equal to or better than a previously approved GE BWR fuel design.*

Previous fuel designs have demonstrated acceptable stability performance thereby assuring that the new fuel design also has acceptable performance. The fuel design comparative evaluation will be performed as follows:

1. A BWR 4 or BWR 5 shall be used as the plant in which the generic comparison is to be performed.

2. The comparison shall assume that the core is first fueled with an equilibrium loading of a previous fuel design approved by the NRC or which meets criterion 1.1.8.A and then with an equilibrium loading of the new fuel design.

3. Both core and limiting channel decay ratios will be calculated at the beginning, middle, and end of the equilibrium cycle.

4. The core and channel decay ratios for both fuel designs shall be calculated using identical operating state conditions for power, flow, inlet subcooling, and core pressure. The axial and radial core power shapes will correspond to the actual operating conditions at these state points, in accordance with the ODYSY procedure outlined in Reference 1–12 or Reference 1-13.

5. The power–flow condition selected shall be on the rated power control rod line and near the point of minimum recirculation pump speed.

6. The methods and procedures used to analyze both fuel designs shall be identical.

B. *If the core and limiting channel decay ratios are not equal to or better than a previously approved GE fuel design, it must be demonstrated that there is no change to the exclusion zone.*

Maintaining the current exclusion zone is an alternate method of demonstrating acceptable fuel stability performance. The evaluations performed to demonstrate compliance with this criterion shall use the same plant and operating conditions as those used to demonstrate compliance with criterion 1.1.8.A.

1.2.9 Overpressure Protection Analysis

A. *Adherence to the ASME overpressure protection criteria shall be demonstrated on plant cycle specific analysis.*

The demonstration of the adequacy of the plant overpressure protection system is dependent upon the plant core loading pattern and must be demonstrated each plant cycle. This cycle specific analysis is performed prior to operation of that core.

1.2.10 Loss–of–Coolant Accident Analysis Methods

A. *The criteria in 10CFR50.46 shall be met on plant–specific or bounding analyses.*

The criteria are currently met by plant exposure dependent, bundle/lattice specific MAPLHGR values that must be met during plant operation. In the future, other criteria or bounding analyses may be approved by the NRC.

B. *Plant MAPLHGR adjustment factors must be confirmed when a new fuel design is introduced.*

Plant MAPLHGR adjustment factors for operation in a configuration or region requiring revised MAPLHGR values such as single recirculation loop operation must be confirmed for each new fuel design. This will be done for each plant prior to the cycle of operation of the new fuel design in that plant.

1.2.11 Rod Drop Accident Analysis

A. *Plant cycle specific analysis results shall not exceed the licensing limit in GESTAR–II.*

The current licensing limit of the control rod drop accident analysis is 280 cal/gm. This limit is based on a large amount of margin to reactivity–induced dispersal of the core and the demonstrated conservatism of current models. New models may result in a revision of the licensing limit. The results of this analysis are dependent upon the plant control rod pattern and the fuel loaded in the core. Plants with BPWS rod sequence control currently are covered by a generic analysis for all fuel types up to GE8x8NB. Plants with group notch rod sequence control must be analyzed each cycle to assure compliance with the licensing criteria. This analysis is performed prior to plant startup each cycle.

B. *Applicability of the bounding BPWS analysis must be confirmed.*

The bounding rod drop accident analysis for plants with BPWS control rod withdrawal sequences is dependent upon the fuel design and must be confirmed generically for each new design. The applicability of the bounding analysis for a new fuel design is determined by comparing the local peaking, Doppler coefficient, and rod worths of the new fuel design with those used for the bounding analyses. The values of the local peaking and Doppler coefficient are obtained from the generic nuclear analyses documented in Subsection 1.2.3. This confirmation will be documented in the fuel design information report for older fuel products (Reference 1-2) and in the compliance reports for GE14 and newer fuel products (See Section 1.4).

1.2.12 Refueling Accident Analysis

A. *The consequences of a refueling accident as presented in the country–specific supplement or the plant FSAR shall be confirmed as bounding or a new analysis shall be performed (using the methods and assumptions described in the country supplement) and documented when a new fuel design is introduced.*

The consequences of the refueling accident are primarily dependent upon the number of fuel rods in a bundle. When the number of fuel rods changes, the effect on the refueling accident must be generically determined based on approved NRC methods. The results of this analysis will be documented in the fuel design information report for older fuel products (Reference 1-2) and in the compliance reports for GE14 and newer fuel products (See Section 1.4).

1.2.13 Anticipated Transient Without Scram

The fuel must meet either criteria A or B below.

This evaluation will assure compliance to the generic ATWS approval. Nuclear inputs used in the evaluation will be obtained from the generic nuclear analyses documented in Subsection 1.2.3.

A. *A negative core moderator void reactivity coefficient, consistent with the analyzed range of void coefficients provided in References 1–7 and 1–8 shall be maintained for any operating conditions above the startup critical condition.*

In response to the requirements of Alternate 3, set forth in NUREG–0460, References 1–7 and 1–8 present assessments of the capabilities of representative BWR plants to mitigate the consequences of a postulated ATWS event. Sensitivity studies are provided for the key parameters affecting plant response during the most limiting events requiring ATWS consideration. Values of parameters that fall within the range of characteristics studied have been shown to satisfy the ATWS acceptance criteria.

In terms of core response to an ATWS event, the core moderator void reactivity coefficient is the key parameter. Maintaining this coefficient within the range of point model void coefficients (or equivalent one–dimensional void coefficients) assumed in the sensitivity studies presented in References 1–7 and 1–8 when loading new fuel designs, assures that the conclusions reached regarding BWR mitigation of an ATWS event are still valid.

B. *If criterion 1.1.13 is not satisfied, the limiting events (as described in References 1–7 and 1–8) will be evaluated to demonstrate that the plant response is within the ATWS criteria specified in References 1–7 and 1–8.*

For new fuel designs that have core moderator void reactivity coefficients outside the range of void coefficients assumed in the sensitivity studies presented in References 1–7 and 1–8, a specific evaluation will be performed. The most limiting events identified in References 1–7 and 1–8 will be evaluated to assure that core and plant response is within the documented ATWS acceptance criteria.

1.2.14 Fuel Loading Error (FLE) Event Analysis

Section S.5.3 of the country–specific supplement presents the requirements for analyzing the FLE (misloaded or misoriented fuel bundle) as an Infrequent Incident. Should a plant not meet the requirements in Section S.5.3, the event will be analyzed as an AOO.

A. As an *Infrequent Incident, the FLE events are subject to the radiological limits of 10% of 10CFR100, or of 10% of 10CFR50.67 for Alternate Source Term plants. A bounding radiological analysis of the fuel loading error events is referenced in the country–specific supplement to this base document. Individual plants confirm site meteorological and off-gas system parameters such that the bounding analysis is applicable.*

   The consequences of the FLE events are primarily dependent upon each plant's long–term meteorological parameters. As described in Section S.5.3 of the country–specific supplement, the results of the confirmation of meteorological conditions will be included for each plant during each reload analysis.

B. *As an AOO option, the FLE events are subject to the MCPR criteria. (See Section 1.2.5 and 1.2.6)*

The results for A or B will be reported in the supplemental reload licensing report.

1.3 Core Configuration

Each BWR reactor core is comprised of core cells. Each core cell consists of a control rod and four fuel assemblies that immediately surround it (Figure 1–1). Each core cell is associated with a four–lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces. The four fuel assemblies are lowered into the core cell and, when seated, springs mounted at the tops of the channels force the channels into the corners of the cell such that the sides of the channels contact the grid beams (Figure 1–2).

Core lattice designations are based upon relative water gap size between adjacent fuel assemblies and dimensional characteristics of the basic fuel assembly and channel. The specific type of core lattice used for each plant is contained in Table 1-1.

1.4 Fuel Product Line GESTAR II Compliance Reports

The following list documents the GESTAR II compliance reports for recent fuel product lines, including revisions. Note that there will generally be a time delay between the publication of a compliance report and its inclusion in this list. The applicable compliance report for a fuel product line is always the most recent revision even when it is not yet included in this list.

GNF will update this list, without NRC review and approval, following the submittal of an initial compliance report or revision of a compliance report to the NRC.

GE11 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II),
NEDE-31917P, April 1991
NEDE-31917P, E&A No.1, May 1991
NEDE-31917P, Revision 1, August 2017

GE13 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II),
NEDE-32198P, December 1993

GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II),
NEDE-32417P, December 1994

GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II),
NEDC-32868P, December 1998
NEDC-32868P, Revision 1, September 2000
NEDC-32868P, Revision 2, September 2007
NEDC-32868P, Revision 3, April 2009
NEDC-32868P, Revision 4, January 2012
NEDC-32868P, Revision 5, May 2013
NEDC-32868P, Revision 6, March 2016

GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II),
NEDC-33270P, March 2007
NEDC-33270P, Revision 1, August 2008
NEDC-33270P, Revision 2, June 2009
NEDC-33270P, Revision 3, March 2010
NEDC-33270P, Revision 4, October 2011
NEDC-33270P, Revision 5, May 2013
NEDC-33270P, Revision 6, March 2016
NEDC-33270P, Revision 7, October 2016
NEDC-33270P, Revision 8, April 2017
NEDC-33270P, Revision 9, December 2017

GNF3 Generic Compliance with NEDE-24011-P-A (GESTAR II),
NEDC-33879P, Revision 0, March 2017
NEDC-33879P, Revision 1, December 2017
NEDC-33879P, Revision 2, March 2018

1.5 References

1–1 *General Electric Fuel Bundle Designs Evaluated with TEXICO/CLAM Analyses Bases*, NEDE–31151P, Revision 0, April 1986.

1–2 *Global Nuclear Fuels Fuel Bundle Designs*, NEDE–31152P, Revision 9, May 2007, including Supplement 1, June 2000, through Supplement 6, May 2007.

1–3 Letter, J. S. Charnley (GE) to C. H. Berlinger (NRC), *Post–Irradiation Fuel Surveillance Programs*, November 23, 1983.

1–4 Letter, L. S. Rubenstein (NRC) to R. L. Gridley (GE), *Acceptance of GE Proposed Fuel Surveillance Program*, June 27, 1984.

1–5 *General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application*, January 1977 (NEDE–10958–PA and NEDO–10958–A).

1–6 Letter, J. S. Charnley (GE) to C. 0. Thomas (NRC), *Amendment 15 to General Electric Licensing Topical Report NEDE–24011–P–A*, January 25, 1986.

1–7 *Assessment of BWR Mitigation of ATWS, Volume I and II (NUREG–0460 Alternate No. 3)*, December 1979, NEDE–24222.

1–8 *Assessment of BWR/3 Mitigation of ATWS (Alternate 3)*, December 1979, NEDE–24223.

1–9 Letter from R. E. Engel (GE) to T. A. Ippolito (NRC), *Lead Test Assembly Licensing*, August 24, 1981.

1–10 Letter from T. A. Ippolito (NRC) to R. E. Engel (GE), *Lead Test Assembly Licensing*, September 23, 1981.

1–11 Letter, M. A. Smith to Document Control Desk, *10CFR Part 21, Reportable Condition, Safety Limit MCPR Evaluations*, May 24, 1996.

1–12 *ODYSY Application for Stability Licensing Calculations*, NEDC–32992P–A, July 2001.

1-13 *ODYSY Application for Stability Licensing Calculations Including Option I-D and II Long Term Solutions*, NEDE–33213P–A, April 2009.

1-14 Letter from A. A. Lingenfelter (GNF) to Document Control Desk (NRC), *Enhanced Lead Use Channel (LUC) Program for NSF Fuel Bundle Channels*, MFN 12-074, September 25, 2012.

1-15 Letter from S. Bahadur (NRC) to A. A. Lingenfelter (GNF), *Final Safety Evaluation for Global Nuclear Fuel – Americas Topical Report (TR) Enhanced Lead Use Channel Program for NSF Fuel Bundle Channels (TAC No. ME9829)*, MFN 13-020, March 29, 2013.

1-16 General Electric Boiling Water Reactor, Maximum Extended Load Line Limit Analysis Plus, NEDC-33006P-A, Revision 3, June 2009.

1-17 Letter, Glen A. Watford (GNF-A) to U.S. Nuclear Regulatory Commission Document Control Desk with attention to Alan Wang (NRC), *NRC Technology Update–Proprietary Slides–July 31–August 1, 2002*, FLN-2002-015, October 31, 2002.

1-18 Approval of TSTF-564. (TBD)

1-19 *Methodology and Uncertainties for Safety Limit MCPR Evaluation*, NEDC–32601P–A, August 1999.

1-20 *Power Distribution Uncertainties for Safety Limit MCPR Evaluations*, NEDC–32694P–A, August 1999.

Table 1-1 Domestic Plant Information

Domestic Plants	Number of Fuel Bundles	Lattice Type
BWR/2		
Nine Mile Point 1	532	D
Oyster Creek	560	D
BWR/3		
Monticello	484	D
Pilgrim	580	D
Dresden 2, 3	724	D
Quad Cities 1, 2	724	D
BWR/4		
Vermont Yankee	368	D
Duane Arnold	368	D
Cooper	548	D
Fitzpatrick	560	D
Hatch 1, 2	560	D
Brunswick 1, 2	560	D
Peach Bottom 2, 3	764	D
Browns Ferry 1, 2, 3	764	D
Fermi 2	764	C
Hope Creek 1	764	C
Limerick 1, 2	764	C
Susquehanna 1, 2	764	C
BWR/5		
Columbia	764	C
LaSalle 1, 2	764	C
Nine Mile Point 2	764	C
BWR/6		
Clinton 1	624	S
Grand Gulf 1	800	S
Perry 1	748	S
River Bend 1	624	S

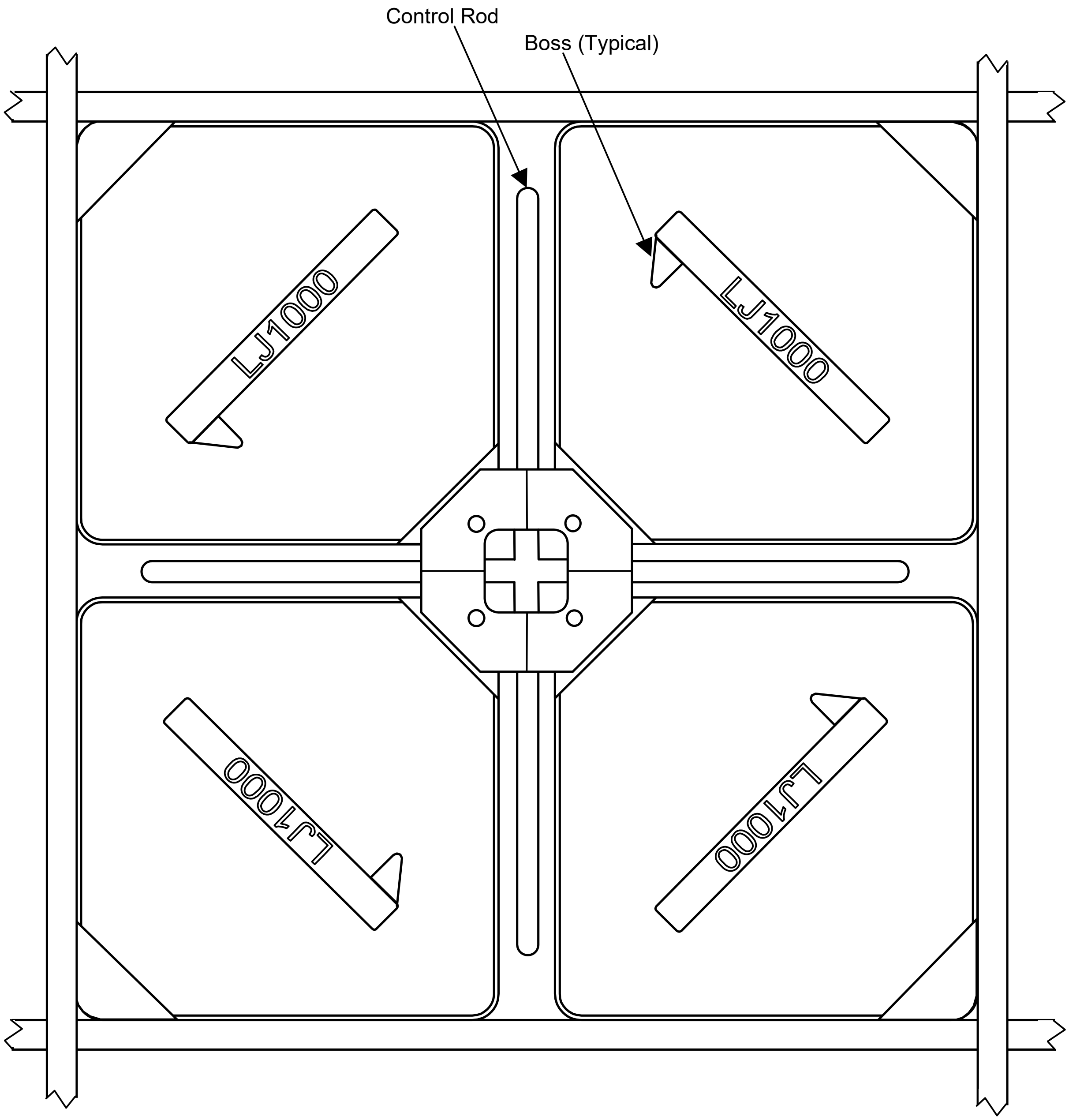

Figure 1–1. Typical Core Cell

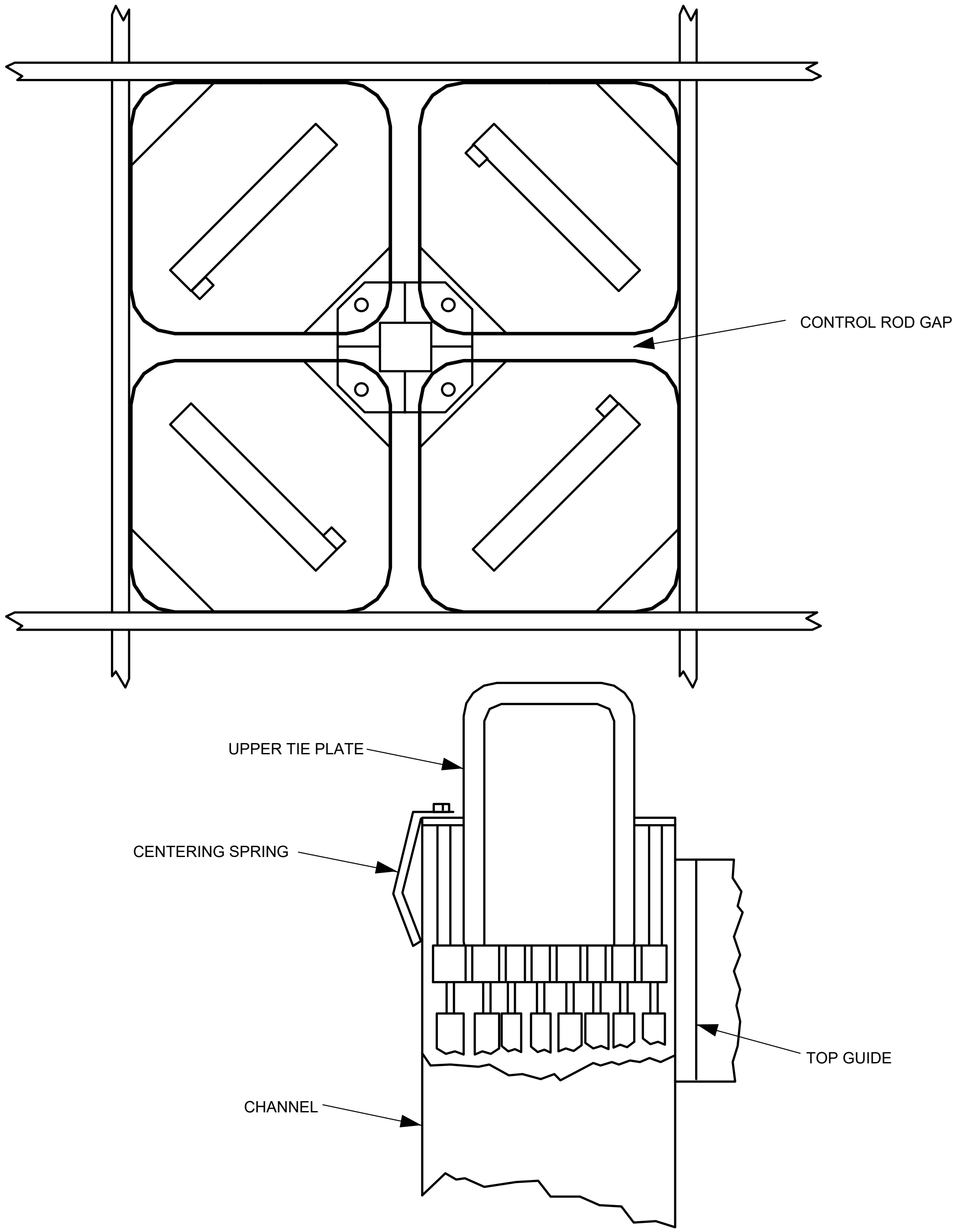

Figure 1-2. Schematic of Four Bundle Cell Arrangement

4. Thermal–Hydraulic Design

4.1 Design Basis

4.1.1 Safety Design Bases

Thermal–hydraulic design of the core shall establish the thermal–hydraulic safety limits for use in evaluating the safety margin relating the consequences of fuel cladding failure to public safety.

4.1.2 Requirements for Steady–State Conditions

For purposes of maintaining adequate fuel performance margin during normal steady–state operation, the MCPR must not be less than the required MCPR operating limit, the APLHGR must be maintained below the required APLHGR limit (MAPLHGR) and the LHGR must be maintained below the required LHGR limit. The steady–state MCPR, MAPLHGR and LHGR limits are determined by analysis of the most severe moderate frequency anticipated operational occurrences (AOOs) to accommodate uncertainties and provide reasonable assurance that no fuel damage results during moderate frequency AOOs at any time in life.

4.1.3 Requirements for Anticipated Operational Occurrences (AOOs)

The MCPR, MAPLHGR and LHGR limits are established such that no safety limit is expected to be exceeded during the most severe moderate frequency AOO event as defined in the country–specific supplement to this document.

4.1.4 Summary of Design Bases

In summary, the steady–state operating limits have been established to assure that the design bases are satisfied for the most severe moderate frequency AOO. Demonstration that the steady–state MCPR, MAPLHGR and LHGR limits are not exceeded is sufficient to conclude that the design bases are satisfied.

4.2 Description of Thermal–Hydraulic Design of the Reactor Core

4.2.1 Critical Power Ratio

A description of the critical power ratio is provided in Subsection 4.3.1. Criteria used to calculate the critical power ratio safety limit are given in Subsection 1.1.5.

4.2.2 Average Planar Linear Heat Generation Rate (APLHGR)

Models used to calculate the APLHGR limit are given in Section 2 as pertaining to the fuel mechanical design limits and in the country–specific supplement to this document as pertaining to 10CFR50 Appendix K limits.

4.2.3 Core Coolant Flow Distribution and Orificing Pattern

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors (References 4–1, 4–2, 4–3). The components of bundle pressure drop considered are friction, local, elevation, and acceleration (Subsections 4.2.4.1 through 4.2.4.4, respectively). Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass flow paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass flow paths. The remainder passes through the orifice in the fuel support plate (experiencing a pressure loss) where some of the flow exits through the fit–up between the fuel support and the lower tieplate and through the lower tieplate holes into the bypass flow region. All initial and reload core fuel bundles have lower tieplate holes. The majority of the flow continues through the lower tieplate (experiencing a pressure loss) where some flow exits through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is lower for those fuel assemblies with finger springs. The bypass flow paths considered in the analysis and typical values of the fraction of bypass flow through each flow path are given in Reference 4–4.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on 1967 or later International Standard Steam–Water Properties. In evaluating fluid properties a constant pressure model is used.

The relative radial and axial power distributions documented in the country–specific supplement are used with the bundle flow to determine the axial coolant property distribution, which gives sufficient information to calculate the pressure drop components within each fuel assembly type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

4.2.4 Core Pressure Drop and Hydraulic Loads

The components of bundle pressure drop considered are friction, local, elevation and acceleration pressure drops. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement.

4.2.4.1 Friction Pressure Drop

Friction pressure drop is calculated with a basic model as follows:

$$\Delta P_f = \frac{w^2}{2g_c\rho}\frac{fL}{D_H A_{ch}^2}\phi_{TPF}^2$$

where

ΔP_f = friction pressure drop

w = mass flow rate

g_c = gravitational conversion factor

ρ = average nodal liquid density

D_H = channel hydraulic diameter

A_{ch} = channel flow area

L = incremental length

f = friction factor

ϕ_{TPF} = two–phase friction multiplier

The formulation for the two–phase multiplier is similar to that presented in References 4–5 and 4–6, and is based on data that is taken from prototypical BWR fuel bundles.

4.2.4.2 Local Pressure Drop

The local pressure drop is defined as the irreversible pressure loss associated with an area change, such as the orifice, lower tieplate, and spacers of a fuel assembly.

The general local pressure drop model is similar to the friction pressure drop and is

$$\Delta P_L = \frac{w^2}{2g_c\rho}\frac{K}{A^2}\phi_{TPL}^2$$

where

ΔP_L = local pressure drop

K = local pressure drop loss coefficient

A = reference area for local loss coefficient

ϕ_{TPL} = two–phase local multiplier

and w, g_c, and ρ are defined above. The formulation for the two–phase multiplier is similar to that reported in Reference 4–6. For advanced spacer designs a quality modifier has been incorporated in the two–phase multiplier to better fit the data. Empirical constants were added to fit the results to data taken for the specific designs of the BWR fuel assembly. These data were obtained from tests performed in single–phase water to calibrate the orifice, the lower tieplate, and the holes in the lower tieplate, and in both single– and two–phase flow, to derive the best fit design values for spacer and upper tieplate pressure drop. The range of test variables was specified to include the range of interest for boiling water reactors. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used.

4.2.4.3 Elevation Pressure Drop

The elevation pressure drop is based on the relation:

$$\Delta P_E = \bar{\rho} \Delta L \frac{g}{g_c}$$

$$\bar{\rho} = \rho_f (1 - \alpha) + \rho_g \alpha$$

where

ΔP_E = elevation pressure drop

Δ_L = incremental length

$\bar{\rho}$ = average mixture density

g = acceleration of gravity

g_c = gravitational conversion factor

α = nodal average void fraction

ρ_f, ρ_g = liquid and saturated vapor density, respectively

The void fraction model used is an extension of the Zuber–Findlay model (Reference 4–7), and uses an empirically fit constant to predict a large block of steam void fraction data. Checks against new data are made on a continuing basis to ensure the best models are used over the full range of interest of boiling water reactors.

4.2.4.4 Acceleration Pressure Drop

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basic formulation for the reversible pressure change resulting from a flow area change in the case of single–phase flow is given by:

$$\Delta P_{ACC} = \left(1 - \sigma_A^2\right) \frac{w^2}{2 g_c \rho_f A_2^2}$$

$$\sigma_A = \frac{A_2}{A_1} = \frac{final\ \ flow\ area}{initial\ \ flow\ area}$$

where

ΔP_{ACC} = acceleration pressure drop

ρ_f = liquid density

g_c = gravitational conversion factor

A_2 = final flow area

A_1 = initial flow area

w = mass flow rate

In the case of two–phase flow, the liquid density is replaced by a density ratio so that the reversible pressure change is given by:

$$\Delta P_{ACC} = \left(1 - \sigma_A^2\right) \frac{w^2 \rho_H}{2 g_c \rho_{KE}^2 A_2^2}$$

where

$\frac{1}{\rho_H} = \frac{x}{\rho_g} + \frac{1-x}{\rho_f}$, homogeneous density,

$\frac{1}{\rho_{KE}^2} = \frac{x^3}{\rho_g^2 \alpha^2} + \frac{(1-x)^3}{\rho_f^2 (1-\alpha)^2}$, kinetic energy density,

α = void fraction at A_2

x = steam quality at A_2

and other terms are as previously defined. The basic formulation for the acceleration pressure change due to density change is:

$$\Delta P_{ACC} = \frac{w^2}{g_c A_{ch}^2} \left[\frac{1}{\rho_{OUT}} - \frac{1}{\rho_{IN}} \right]$$

where ρ is either the homogeneous density, ρ_H, or the momentum density, ρ_M

$$\frac{1}{\rho_M} = \frac{x^2}{\rho_g \alpha} + \frac{(1-x)^2}{\rho_f (1-\alpha)}$$

and is evaluated at the inlet and outlet of each axial node. Other terms are as previously defined. The total acceleration pressure drop in boiling water reactors is on the order of a few percent of the total pressure drop.

4.2.5 Correlation and Physical Data

General Electric Company has obtained substantial amounts of physical data in support of the pressure drop and thermal–hydraulic loads discussed in Subsection 4.2.4. Correlations have been developed to fit these data to the formulations discussed.

4.2.5.1 Pressure Drop Correlations

General Electric Company has taken significant amounts of friction pressure drop data in multi–rod geometries representative of BWR plant fuel bundles and correlated both the friction factor and two–phase multipliers on a best fit basis using the pressure drop formulations reported in Subsections 4.2.4.1 and 4.2.4.3. Tests are performed in single–phase water to calibrate the orifice and the lower tie–plate, and in both single– and two–phase flow to arrive at best fit design values for spacer and upper tie–plate pressure drop. The range of test variables is specified to include the range of interest to boiling water reactors. New data are taken whenever there is a significant design change to ensure the most applicable methods are in use at all times.

Applicability of the single–phase and two–phase hydraulic models discussed in Subsections 4.2.4.1 and 4.2.4.2 for fuel designs as described in Section 1.4, was confirmed by full scale prototype flow tests.

4.2.5.2 Void Fraction Correlation

The void fraction correlation includes effects of pressure, flow direction, mass velocity, quality, and subcooled boiling.

4.2.5.3 Heat Transfer Correlation

The Jens–Lottes (Reference 4–8) heat transfer correlation is used in fuel design to determine the cladding–to–coolant heat transfer coefficients for nucleate boiling.

4.2.6 Thermal Effects of Anticipated Operational Occurrences

The evaluation of the core's capability to withstand the thermal effects resulting from anticipated operational occurrences is covered in Chapter 15 (Accident Analysis) of the plant FSAR.

4.2.7 Uncertainties in Estimates

Uncertainties in thermal–hydraulic parameters are considered in the statistical analysis that is performed to establish the fuel cladding integrity safety limit documented in Subsection 4.3.1.1.

4.2.8 Flux Tilt Considerations

For flux tilt considerations, refer to Subsection 3.2.2.

4.3 Evaluation

The thermal–hydraulic design of the reactor core and reactor coolant system is based upon an objective of no fuel damage during normal operation or during anticipated operational occurrences. This design objective is demonstrated by analysis as described in the following sections.

4.3.1 Critical Power

Notes: (These notes are replicated from Section 1.1.5)

Plants Adopting TSTF-564

For plants that have adopted TSTF-564, the Technical Specification Safety Limit MCPR is cycle-independent as described in Reference 4-51. TSTF-564 uses the term SLMCPR95/95 to define the cycle-independent safety limit that will be applied in Technical Specification (TS) 2.1.1.2. (This TS Section reference is used in the TSTF and is based on the Standard Technical Specifications. Specific plants may have a different TS section for the Safety Limit MCPR.) The cycle specific SLMCPR is termed MCPR99.9% in TSTF-564 and will be included in the cycle-specific Core Operating Limits Report (COLR). The following table summarizes the CPR terminology.

MCPR95/95	Cycle independent value determined based on the GEXL correlation statistics using the expression defined in TSTF-564
SLMCPR95/95	Cycle-independent Technical Specification Safety Limit
MCPR99.9%	Cycle-specific COLR SLMCPR

There is no change in the methodology used to calculate the MCPR99.9%. The cycle-specific SLMCPR methodologies remain as described in Section 1.1.5, in Section 1.2.5, and in Section 4.3.1.1.

The proposed MCPR95/95 values for fuel product lines GE14, GNF2, and GNF3, which may be used to define the SLMCPR95/95, are included in Table 1 of TSTF-564. Section 3.1 of TSTF-564 describes the methodology to be used in the development of the MCPR95/95. For new fuel products, GNF will provide the NRC a letter like Reference 1 of TSTF-564, which may be referenced by a licensee requesting a change to SLMCPR95/95 in their Technical Specifications.

Historically, the term SLMCPR has been used for the statistical limit defined by the approved methodology (References 4-36 and 4-37). This

term is used in GESTAR II, the SLMCPR methodology documents, and in numerous reports. GNF does not intend to change any previous usage.

Plants Not Adopting TSTF-564

For plants that have "not" adopted TSTF-564, the Technical Specification Safety Limit MCPR will remain as the cycle-specific SLMCPR described in Section 1.1.5, in Section 1.2.5, and in Section 4.3.1.1.

The objective for normal operation and AOOs is to maintain nucleate boiling and thus avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. The figure of merit utilized for plant operation is the critical power ratio. This is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure that exist at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR, which corresponds to the most limiting fuel assembly in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected as follows.

Moderate frequency AOOs caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, at least 99.9% of the fuel rods would be expected to avoid boiling transition (Reference 4–9).

Both the transient (safety) and normal operating thermal limits in terms of MCPR are derived from this basis. A discussion of these limits follows.

4.3.1.1 Fuel Cladding Integrity Safety Limit

The generation of the Minimum Critical Power Ratio (MCPR) limit requires a statistical analysis of each reload core near the limiting MCPR condition. The statistical analysis is used to determine the MCPR corresponding to the transient design requirement given in the United States supplement. The MCPR Fuel Cladding Integrity Safety Limit applies not only for core wide AOOs, but is also applied to the localized rod withdrawal error AOO. The cycle–specific Safety Limit MCPR is derived based on the criteria of Subsection 1.1.5.

4.3.1.1.1 Statistical Model

The statistical analysis utilizes a model of the BWR core that simulates the ~~process computer~~**core monitoring** function. This code produces a critical power ratio (CPR) map of the core based on inputs of power distribution, flow and heat balance information. Details of the procedure are documented in Appendix IV of Reference 4–9 and Section 4 of Reference 4–36. Random Monte Carlo selections of all operating parameters based on the uncertainty ranges of manufacturing tolerances, uncertainties in measurement of core operating parameters, calculational uncertainties, and statistical uncertainty associated with the critical power correlations are imposed upon the analytical representation of the core and the resulting bundle critical power ratios are calculated. ~~Uncertainties used in the cycle–specific statistical analysis is presented in References 4–36 and 4–37. Although some of the plant–unique uncertainties may be greater for some plants, other uncertainties for these plants are smaller and the analysis is applicable.~~

The minimum allowable critical power ratio is set to correspond to the criterion that 99.9% of the rods are expected to avoid boiling transition by interpolation among the means of the distributions formed by all the trials.

4.3.1.1.2 BWR Statistical Analysis

Statistical analyses are performed for each operating cycle that provides the fuel cladding integrity Safety Limit MCPR. This Safety Limit MCPR is derived based on the criteria in Subsection 1.1.5. **Uncertainties used in the cycle–specific statistical analysis are presented in References 4–36 and 4–37. These uncertainties are confirmed during the cycle-specific analysis process by the plant. The plant may elect to use larger uncertainties during this process.**

For plants licensed for operation in the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) extended operating domain (Reference 4-49), the required core power and core flow state points and associated uncertainties are defined in Reference 4-49. Applicable additional margin may also be required per the Safety Evaluation (SE) for Reference 4-50.

4.3.1.1.3 Methodology Restrictions

Four restrictions were identified on page 3 of NRC's SE relating to the General Electric (GE) Licensing Topical Reports (LTRs) NEDC-32601P, NEDC-32694P, and in Amendment 25 to NEDE-24011-P-A (Reference 4-43).

The four restrictions were addressed for GE14 in FLN-2001-016 "Confirmation of 10x10 Fuel Design Applicability to Improved SLMCPR" (Reference 4-44).

Section 3.6 of the GNF2 GESTAR II compliance report (Reference 4-45) addresses the restrictions for the GNF2 product line.

Section 3.6 of the GNF3 GESTAR II compliance report (Reference 4-46) addresses the restrictions for the GNF3 product line.

4.3.1.1.4 Deviations from Reference 4-36 Uncertainties

R-Factor

The GEXL R-Factor uncertainty was increased from 1.6% to 2.0% to account for an increase in channel bow due to the phenomena called control blade shadow corrosion-induced channel bow, which is not accounted for in the channel bow uncertainty component of 30 mils for which the Reference 4-36 R-Factor uncertainty of 1.6% is

based on. Reference 4-47 technically justifies that a GEXL R-Factor uncertainty of 2.0% accounts for a channel bow uncertainty of up to 62 mils.

The SE for the NSF LTR (Reference 4-48) allows revisions to the R-factor uncertainty based upon NSF channel distortion measurements under the provisions described in Limitation and Condition (L&C) 7. L&C No. 7 allows future changes in the R-factor uncertainty based upon incorporation of NSF channel distortion measurements which are justified and documented in the annual report required by L&C 4. The cycle-specific SLMCPR will use a GEXL R-Factor uncertainty of 2.0% until such time that this annual report documents that a bow uncertainty of up to 30 mils is appropriate for NSF channels at which time a GEXL R-Factor uncertainty of 1.6% will be used.

Core Flow Rate and Random Effective TIP Reading

As described in Section 1.1.5, SLMCPR analyses are performed at core power/flow points consistent with the licensed domain boundary and specific requirements of expanded operating domains. Section 1.2.5 presents general guidance for the core power and flow state points to be analyzed. The approved cycle-specific SLMCPR methodology is applied at each state point that is analyzed.

For the TLO calculations performed at less than 99% core flow, the approved uncertainty values for the core flow rate (2.5%) and the random effective Traversing In-Core Probe (TIP) reading (1.2%) are conservatively adjusted by dividing them by the percent core flow/100. For example,

Core Flow Rate Uncertainty at 90% Core Flow = 100%*2.5%/90% = 2.78%

The core flow and random TIP reading uncertainties used in the SLO minimum core flow SLMCPR analysis remain the same as in the rated core flow SLO SLMCPR analysis because these uncertainties (which are substantially larger than used in the TLO analysis) already account for the effects of operating at reduced core flow.

Flow Area Uncertainty

The flow area uncertainty for GE14, GNF2, and GNF3 using the process described in Section 2.7 of Reference 4-36 has been recalculated. This recalculation determined that the flow area uncertainty for GE14, GNF2, and GNF3 is larger than the Reference 4-36 value of 2.0%. If the resulting numerical value, using the process described in Section 2.7 of Reference 4-36, of the flow area uncertainty for a fuel product line is greater than 2.0%, then it will be documented in the GESTAR II Compliance Report for that product and used in the cycle specific SLMCPR calculations.

4.3.1.2 MCPR Operating Limit Calculational Procedure

A plant–unique MCPR operating limit is established to provide adequate assurance that the cycle–specific fuel cladding integrity safety limit for that plant is not exceeded for any moderate frequency AOO. This operating requirement is obtained by addition of the maximum ΔCPR value for the most limiting AOO (including any imposed adjustment factors) from conditions postulated to occur at the plant to the cycle–specific fuel cladding integrity safety limit.

4.3.1.2.1 Calculational Procedure for AOO Pressurization Events

Core–wide rapid pressurization events (turbine trip w/o bypass, load rejection w/o bypass, feedwater controller failure) are analyzed using the system model (ODYN) documented in References 4–16 and 4–17. Improvements made in ODYN using the physics methods of Reference 4–18 are documented in References 4–19 and 4–20. An updated version of ODYN using the advanced physics methods of Reference 4–21 is described in Reference 4–22. As described in Reference 4–22, this creates two integrated, self–consistent sets of methods, referred to as GENESIS and GEMINI, for analyzing core–wide rapid pressurization events. For GE11 and later fuel products, the time varying axial power shape is calculated by ODYN (Reference 4–34). TRACG has been approved for application to AOO transients. TRACG uses a multi–dimensional two–fluid model and a three–dimensional kinetics model consistent with the GEMINI method. The application of TRACG is described in Reference 4–40. The set of methods used (GENESIS, GEMINI or TRACG) will be identified in the supplemental reload licensing report; however, application of a different approved method set may be used subsequently for the same cycle.

4.3.1.2.2 Calculational Procedure for AOO Slow Events

The slower core–wide anticipated operational occurrence, loss of feedwater heating, is analyzed using either the steady–state 3–D BWR Simulator Code (Reference 4–18 for GENESIS methods or Reference 4–21 for GEMINI methods), the REDY transient model (References 4–23, 4–24 and 4–25) as described in Reference 4–26, the ODYN system model documented in Reference 4–39, or the TRACG model as described in Reference 4–40. Inadvertent HPCI startup may be bounded by that of the loss of feedwater heating event (Reference 4–35). When necessary, it is analyzed using the REDY transient model, the ODYN system model or the TRACG system model. The scram reactivity used for slow events is shown in Figure 4–1.

4.3.1.2.3 Rod Withdrawal Error Calculational Procedure

The reactor core behavior during the rod withdrawal error transient is calculated by doing a series of steady–state three–dimensional coupled nuclear–thermal–hydraulic calculations using the 3–D BWR Simulator (Reference 4–18 for GENESIS methods or Reference 4–21 for GEMINI methods).

4.3.1.2.4 Event Descriptions

Descriptions of the limiting AOO events are given in the country–specific supplement to this document. The AOO descriptions given in the country–specific supplement to this document are used as a basis for the typical analyses performed. Some plant–unique analyses will differ in certain aspects from the typical calculational procedure. These differences arise because of utility–selected margin improvement options.

4.3.1.2.5 MCPR Operating Limit Calculation

The operating limit MCPR for rapid AOOs is calculated by using the TASC computer program (References 4–28 and 4–41) or TRACG (Reference 4–40). The country–specific supplement to this document lists the plant initial conditions for the MCPR operating limit analysis. Values used in reload analyses may be different from those given in the country–specific supplement to this document. In these cases, the values used appear in the supplemental reload licensing report. Cycle–dependent plant initial conditions for the MCPR operating limit analysis and the resulting parameters are given in the FSAR or in the supplemental reload licensing report.

4.3.1.2.6 MCPR Uncertainty Considerations

The deterministic ΔCPR value that results from ODYN/TASC evaluations (for all rapid pressurization AOOs) must be adjusted such that a 95/95 ΔCPR/ICPR licensing basis is calculated (i.e., 95% probability with 95% confidence that the safety limit will not be violated). The SER, which describes these requirements and procedures, is given in Reference 4–29.

Each utility has the choice of operating under either Option A or Option B.

Option A — For plants operating under Option A with the GENESIS set of methods, an NRC–imposed factor of 1.044 is applied to the MCPR for each event to account for code uncertainties.

With the GEMINI set of methods, the MCPR for each event is determined using statistically evaluated scram times. Plants that do not demonstrate compliance with the statistically evaluated scram times must operate using a higher limit that does not take credit for these scram times. The higher limit will also be referred to as Option A. Details are provided in Reference 4–31.

Option B — Under Option B, the ΔCPR/ICPR ratio for the pressurization events is evaluated on either a plant–unique or generic statistical basis per the methodology and procedures of References 4–29 and 4–30 for GENESIS, and Reference 4–31 for GEMINI. The generic basis utilizes adjustment factors that are dependent on plant and event type. Reference 4–29 summarizes these factors for the GENESIS set of methods. For the GEMINI set of methods, the adjustment factors and their application are described in References 4–31 and 4–38. Since both the GENESIS and GEMINI adjustment factors take credit for conservatism in the scram speed assumed for the transient analyses, each plant operating under Option B must demonstrate that their actual scram speeds are within the distribution assumed in the derivation of the adjustment factors. This conformance procedure is described in Reference 4–29.

The adjusted MCPR values for all rapid pressurization events are given in the FSAR or in the supplemental reload licensing report.

If the ΔCPR is calculated by TRACG (Reference 4–40), the ΔCPR and the OLMCPR are calculated such that less than 0.1% of the fuel rods will be subject to boiling transition during the transient.

4.3.1.2.7 Low Flow and Low Power Effects on MCPR

The operating limit MCPR must be increased at low flow conditions, and the operating limit MCPR must be increased for BWR/6 plants and plants with ARTS at low flow and low power conditions. For low flow conditions this is because, in the BWR, power increases as core flow increases, which results in a corresponding lower MCPR. If the MCPR at a reduced flow condition were at the 100% power and flow MCPR operating limit, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Fuel Cladding Integrity Safety Limit MCPR.

Therefore, the required operating limit MCPR for the BWR/2–6 plants is increased at reduced core flow. This is accomplished by specifying an absolute MCPR as a function of core flow ($MCPR_f$) or as a multiplier (K_f) on the rated OLMCPR.

Plants licensed for the Average Power Range Monitor, Rod Block Monitor and Technical Specification (ARTS) Improvement Program have both power– and flow–dependent limits imposed on the operating limit MCPR (OLMCPR). The flow–dependent required OLMCPR, $MCPR_f$, is defined as a function of the core flow rate and positioning of the scoop tube on the recirculation pump motor or the maximum core flow runout for plants with the recirculation flow control valves or adjustable speed drives. The flow-dependent MCPR limits are provided in the cycle-specific Supplemental Reload Licensing Report.

For powers between 100% of rated and the bypass point for the turbine stop valve/turbine control valve fast closure scram signal (about 30% of rated), the power–dependent OLMCPR, $MCPR_p$, is determined from the product of the OLMCPR at 100% of rated and a power–dependent multiplier, K_p. For powers between threshold for thermal limits monitoring (e.g., 25% of rated) and the bypass point, the $MCPR_p$ limits are absolute values and are defined separately for high core flows (e.g., >50% of rated flow) and for low core flows (e.g., ≤50% of rated flow) conditions. Thermal limits monitoring is not required below approximately 25% of rated power. The power-dependent MCPR limits are provided in the cycle-specific Supplemental Reload Licensing Report. The OLMCPR to be used at powers less than 100% becomes the most limiting value of either $MCPR_f$ or $MCPR_p$.

Plants with a Rod Withdrawal Limiter (RWL) system also require power distribution limits. The RWL system restricts control rod motions as a function of power rather than the local neutron flux used by the Rod Block Monitor (RBM) system.

4.3.1.2.8 End–of–Cycle Coastdown Considerations

AOO analyses are performed at the rated core power, rated core flow, all–rods–out condition referred to as End-of-Rated (EOR). Once an individual plant reaches this condition, it may shutdown for refueling or it may be placed in a coastdown mode of operation. In the end–of–cycle coastdown type of operation the control rods are normally held in the all–rods–out position and the plant is allowed to coastdown to a lower percent of rated core power while maintaining rated core flow. The power profile during this period is assumed to be a linear function with respect to exposure. It is expected that the actual profile will be a slow, exponential curve. An analysis to the linear approximation, however, will be conservative, since it over predicts the core power level for any given exposure.

In Reference 4–32, evaluations were made at 90%, 80%, and 70% core power level points on the linear curve. The results show that the pressure and MCPR from the limiting pressurization AOO exhibit a larger margin for each of these points than the EOR condition. LHGR limits for the EOR condition are conservative for the coastdown period, since the core power will be decreasing and rated core flow will be maintained. Therefore, it can be concluded that the coastdown operation beyond the EOR condition is conservatively bounded by the analysis at the EOR conditions. In Reference 4–33, this conclusion is confirmed for coastdown operation down to 40% power and is shown to hold for analyses performed with ODYN. Analyses with TRACG show the same trends as the evaluation in Reference 4-33, therefore, the same conclusion applies for TRACG based analyses.

4.3.2 Core Hydraulics

Core hydraulics models and correlations are discussed in Section 4.2.

4.3.3 Influence of Power Distributions

The influence of power distributions on the thermal–hydraulic design is discussed in Reference 4–9.

4.3.4 Core Thermal Response

The thermal response of the core for accidents and expected AOO conditions is given in Chapter 15 (Accident Analysis) of the plant FSAR or in the supplemental reload licensing report.

4.3.5 Analytical Methods

The analytical methods, thermodynamic data, and hydrodynamic data used in determining the thermal and hydraulic characteristics of the core are documented in Subsection 4.3.1.2 of this document and the country–specific supplement to this document.

4.3.6 PRIME Transient Methodology

The PRIME transient methodology has been approved for application to AOO transients (Reference 4-42). This methodology is applicable to the analysis of the fuel rod response for all transient events, but is particularly designed to support the analysis of fast (short duration relative to the fuel rod thermal time constant) events.

The PRIME transient methodology may be used to perform a detailed fuel rod thermal-mechanical analysis with inputs from the system transient analysis, such as ODYN or

TRACG. It may also be used to develop screening criteria for use with the transient analysis results to determine that adequate margin exists.

4.4 References

4–1 *Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello*, NEDO–10299A, October 1976.

4–2 H. T. Kim and H. S. Smith, *Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad Cities Unit 1*, NEDO–10722A, August 1976.

4–2 *Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In–Core Vibrations*, NEDO–21215, March 1976.

4–4 *Supplemental Information for Plant Modification to Eliminate Significant In–Core Vibration*, NEDE–21156 (Proprietary), January 1976.

4–5 R. C. Martinelli and D. E. Nelson, *Prediction of Pressure Drops During Forced Convection Boiling of Water*, ASME Trans., 70, 695–702, 1948.

4–6 C. J. Baroozy, *A Systematic Correlation for Two–Phase Pressure Drop*, Heat Transfer Conference (Los Angeles), AICLE, Preprint No. 37, 1966.

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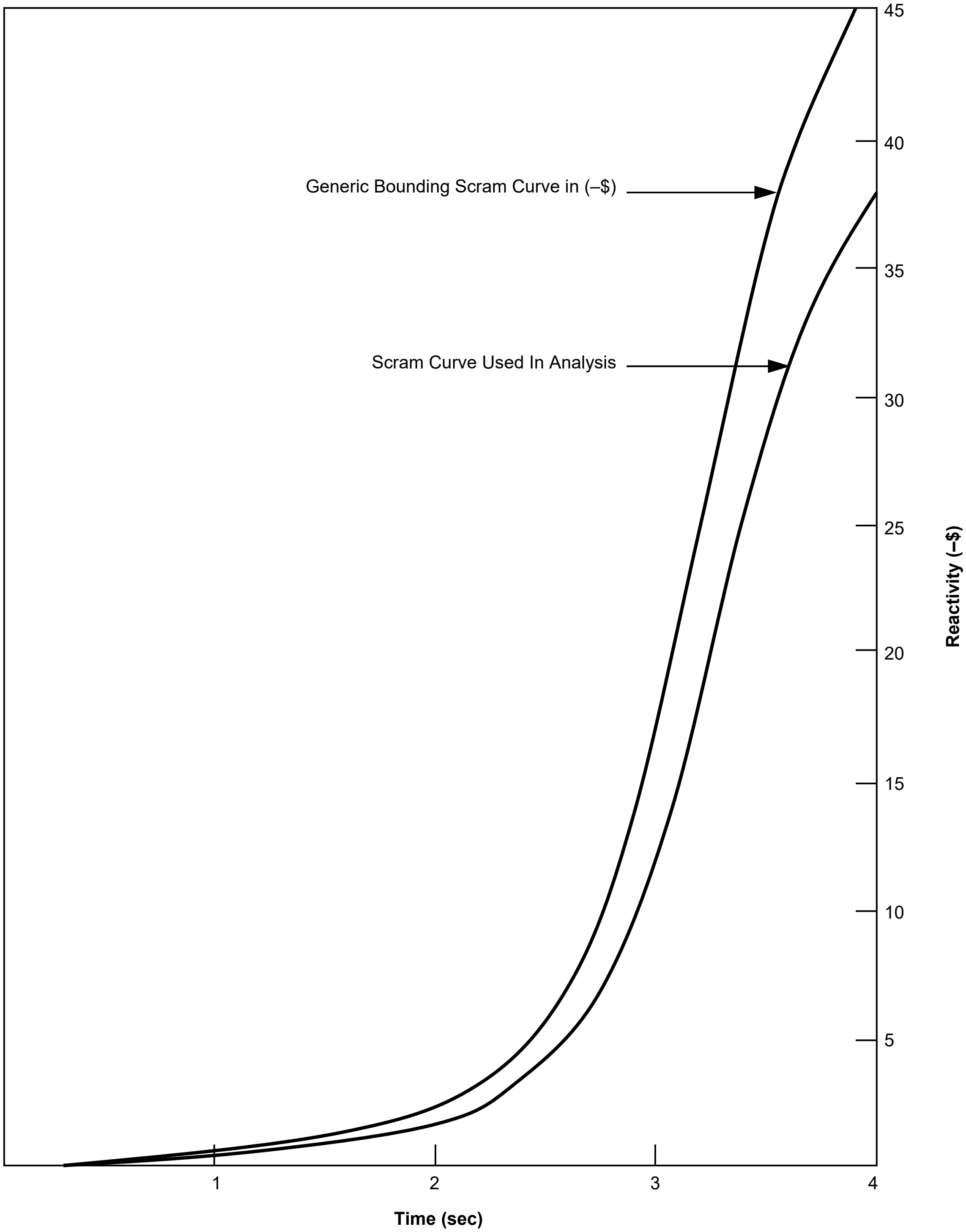

Figure 4–1. Transient Analysis Input–Scram Reactivity (REDY Events)