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.

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 has 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.

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. These bases 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.

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

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.

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.

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

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.

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.

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:

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

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.*

The basic criterion associated with the storage of both irradiated and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be less than or equal to 0.90 for regular density racks and less than or equal to 0.95 for high density racks. Abnormal storage conditions are limited to a k_{eff} of less than or equal to 0.95 for both high and regular density designs. For GE designed fuel storage racks, these 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

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.*

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 and Table 3–3 of Reference 1–2. 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 fuel design information report.

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 fuel design information report.

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.

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.
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.

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.

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 core lattice descriptions and a definition of the specific type of lattice used for each plant are contained in Reference 1–2.

1.4 References

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

1–2 *General Electric Fuel Bundle Designs*, NEDE–31152P, Revision 8, April 2001.

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.

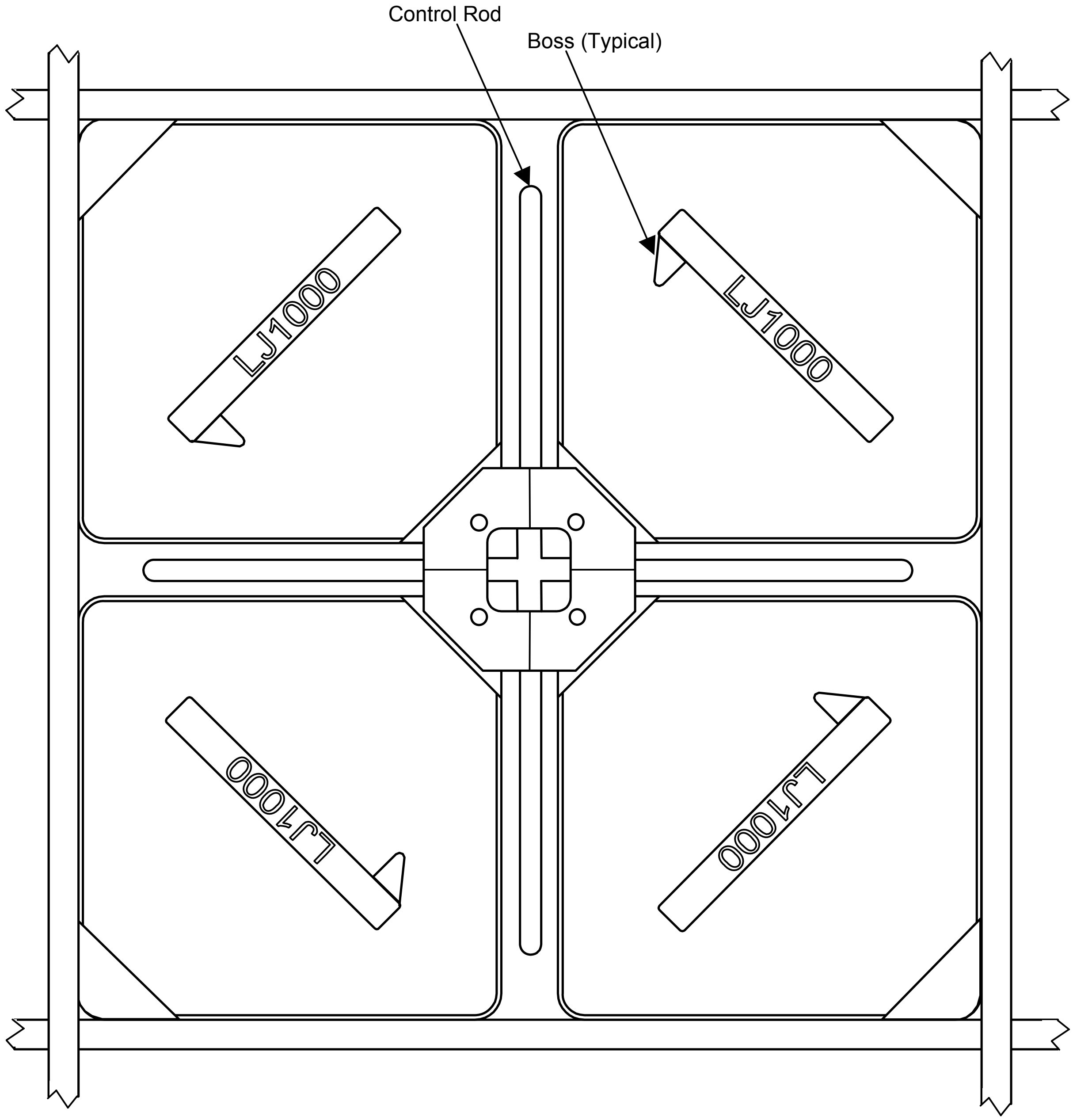

Figure 1–1. Typical Core Cell

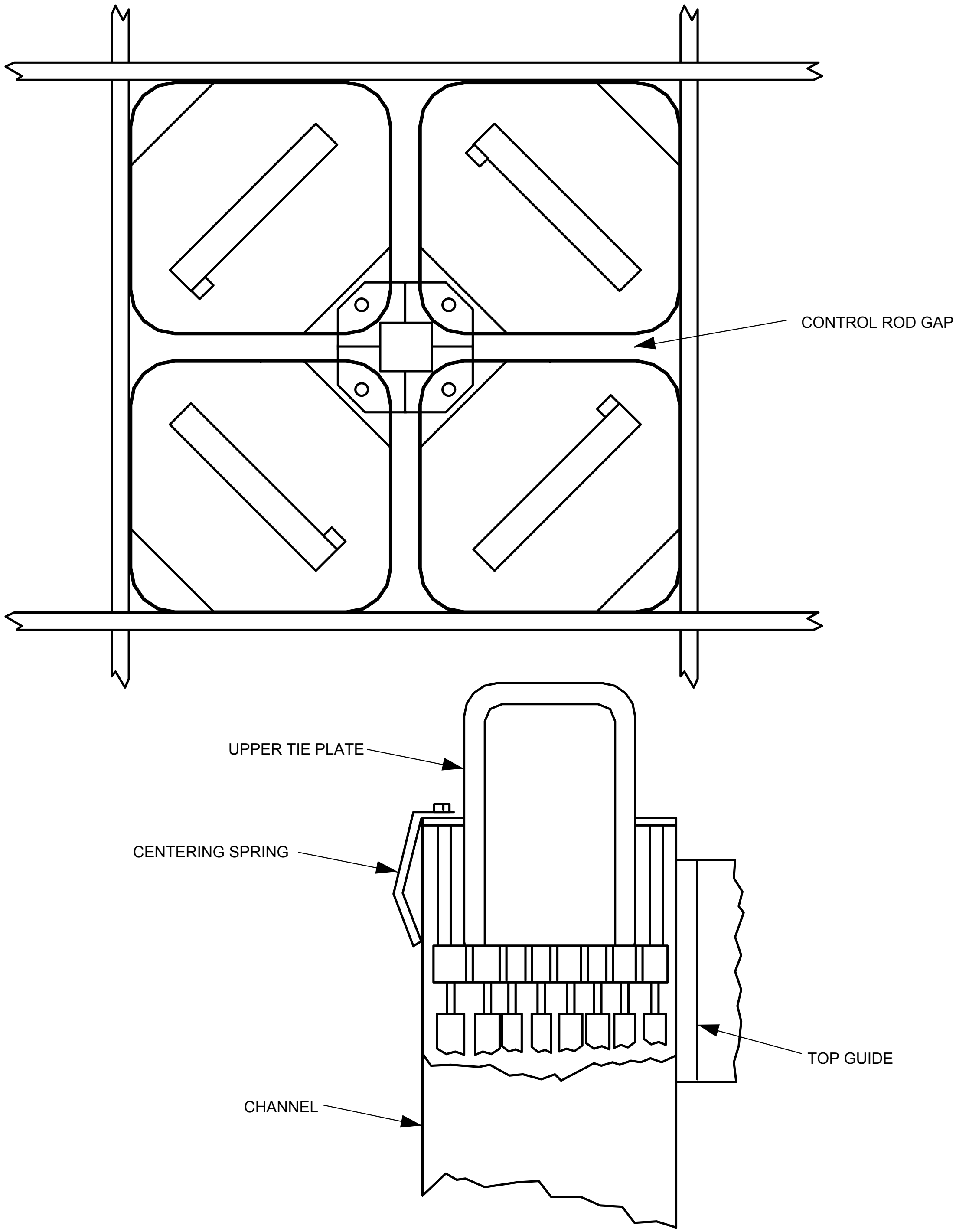

Figure 1-2. Schematic of Four Bundle Cell Arrangement