ENCLOSURE 3

M180050

NEDO-24011-A, Revision 27, General Electric Standard Application for Reactor Fuel (GESTAR II, Main), August 2018

Non-Proprietary Information – Class I (Public)

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Global Nuclear Fuel

NEDO–24011–A–27
August 2018

Non-Proprietary Information – Class I (Public)

Licensing Topical Report

General Electric Standard Application for Reactor Fuel

(GESTAR II)

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GESTAR II

Revision Status Sheet

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
5	5	1 & 2	3	Administrative.
6	6	4	5	Incorporation of barrier clad.
7	7	6	7	GESTR–M fuel mechanical code application.
	8	8	9	Generic stability approach.
	9	10	11	Administrative.
8	10	12	13 & 14	Incorporation of GE8x8E and GE8x8EB fuel designs.
	11	15	16 & 17	Revised ODYN (GEMINI methods).
	12	18	19	Generic CRDA analysis for group notch plants utilizing BPWS.
	13	20	21	Administrative.
9	14	22	23	Safety limit MCPR reduction.
	15	24 & 25	26 & 27	GEXL–Plus correlation.
	16	28	29	Administrative.
	17	30	31	Modified requirements for rod pattern control system.
	18	32	33	Incorporation of GE8x8NB fuel design.
	19	34	35	Changes to technical specifications for power distribution limits.
10	21	36	37	Incorporation of GE8x8NB–1, –2, and –3 fuel designs.
	22	38 & 39	40	Fuel licensing acceptance criteria.
		41	42	Fuel channel bow effect on thermal margins.
11		43	44	TVAPS added and refueling accident updated to incorporate GE11 and later fuel designs.

[a] Only approved amendments are incorporated into a revision to GESTAR. Usually several approved amendments are combined together and incorporated into the document as one revision. GESTAR II began with Revision 4.

[b] See following pages for references.

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
12			45	Superceded by Revision 13.
13		46	47	Cold water event determination.
14	25	48	49	Cycle–specific Safety Limit MCPR.
	26	50	51, 52	Administrative including approvals of Stability and ATWS, inclusion of MLHGR, classifying PRDF, implementing improved GE steady–state methods, et. al.
15	27	53, 54	55, 56	Administrative including approvals of LOCA's SAFER/GESTR, Upper Bound PCT, and TASC; and TRACG. Also, clarified position on stability. Several errors corrected on reload licensing. Other minor errors cleaned up
16	28	57, 58	59	The fuel loading error event is now analyzed as an Infrequent Incident. A generic bounding analysis is applied when the plant confirms site-specific information and that the plant's fuel loading verification procedures meet the requirements defined in the US Supplement Section S.5.3.
	29	60	61	Administrative to incorporate recently approved references.
17	30	62	63	Administrative to implement the referencing of NEDC-33173P, Applicability of GE Methods to Expanded Operating Domains.
	31	64	65	Updates the stability analysis and the SRLR template.
	32	66	67	Temporary LHGR limitation on exposure lifetime for GNF2 fuel design based on current approved thermal-mechanical fuel design method, GESTR-M.
	33	68	69	Implementation of approved PRIME methodology.
18	34	70	71	Administrative to implement the referencing of NEDC-33006P, Maximum Extended Load Line Limit Analysis Plus and NEDE-32906P, Supplement 3-A, Migration to TRACG04 / PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
19	35	72, 73, 74	75	Administrative to implement the referencing of the accepted versions of Supplements 2, 3, and 4 of NEDC-33173P, Applicability of GE Methods to Expanded Operating Domains. Also, corrected a typographical error in a section reference in Section 3.4.2.10 from 4.2 to 3.4.2.
20			76	Administrative to implement the referencing of the accepted version of the Enhanced Lead Use Channel (LUC) Program for NSF Fuel Bundle Channels.
21	36	77, 78, 79	80	Administrative to implement referencing of the accepted version of NEDE-32465 Supplement 1P-A, Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications. Also, updated the reference to NEDO-33173 Supplement 4-A, Implementation of PRIME Models and Data in Downstream Methods, to incorporate the NRC staff audit and updated the reference to NEDC-33173P-A, Applicability of GE Methods to Expanded Operating Domains. Also updated to incorporate the accepted version of Revision 7 of NEDC-33075P, General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density.
	38	81	82	Administrative to clarify the potential effect of unique plant configurations on AOO analyses.
	39	83	84	Updated to state that end-of-cycle coastdown conditions are always bounded by the normal reload analyses when the TRACG04 AOO methodology is used.
	41	85	86	Implementation of the approved GS3 methodology.
22			87	Implementation of the approved Application of NSF to GNF Fuel Channel Designs. Also made administrative corrections to US Supp. Sections S.3, S.6, and Appendix B, as concurred via NRC email dated November 19, 2015.

Revision No.[a]	Amend. No.	GE Amend. Reference[b]	NRC Approval Reference[b]	Amendment Content
23	42	88, 89	90	Updated the core monitoring system from 3D-MONICORE to ACUMEN.
24	37	91-95	96	Incorporated the TRACG-LOCA methodology. Deleted NEDE-31152P as a future requirement and added the lattice table from NEDE-31152P. Added a plant and cycle specific FBIR requirement and added the FBIR shell to Appendix A. Added a new section listing the compliance report references for each fuel product line. Clarified the text with respect to the analyses that support the in-core k_∞ limits. Incorporated SAFER/PRIME. Removed SAFE/REFLOOD as a LOCA evaluation methodology and removed the SAFER/GESTR report list and associated references.
25	40	97, 98	99	Clarified statements regarding 10x10 fuel bundle designs to separate the refueling accident analysis methods from the results, which are product line specific and have been relocated to the fuel product line GESTAR II compliance reports.
	45	100	101	Clarified the application of GESTAR II to plants that have specific approval to operate at greater than 120% OLTP in the MELLLA+ domain.
	--	--	--	Updated the compliance report references for the GNF2 fuel product line and added the GNF3 compliance report to the list. Corrected a reference citation in Section 4.3.1.2.6.
26	44	102	103	Incorporated the PRIME Transient methodology. Updated the compliance report references for the GE11, GNF2, and GNF3 fuel product lines.
27	43	104	105	Clarified statements to make the final core loading pattern criterion more precise and reflective of processes that have evolved over the years.
	--	--	--	Updated the TRACG LOCA methodology reference to Revision 2. Updated the compliance report reference for the GNF3 fuel product line.

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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

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.

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.

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*.

[[

]] 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*.

[[

]] 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*.

[[

]]

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

[[

]]

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

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. 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.*

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

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.*

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 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.

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

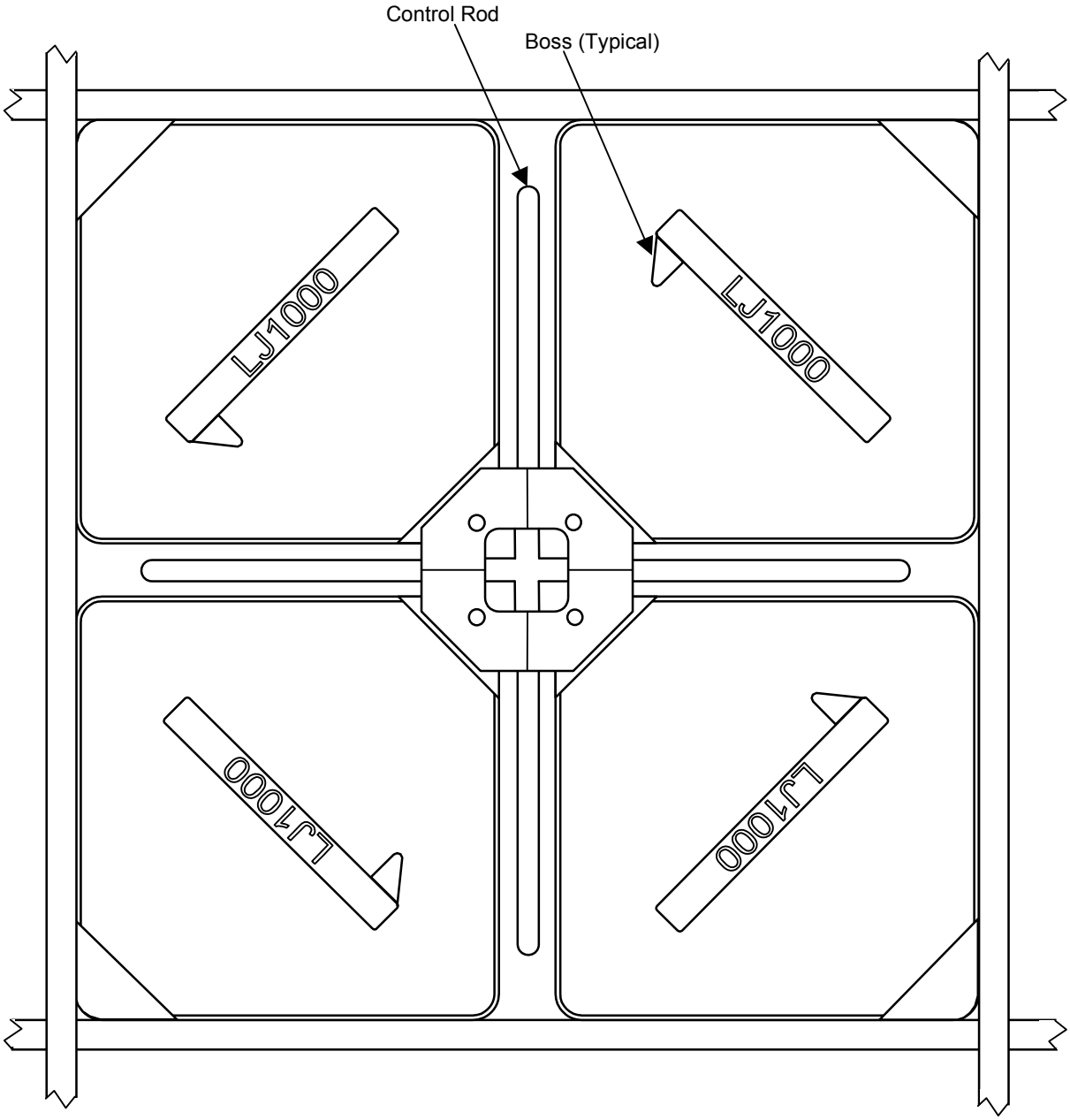

Figure 1–1. Typical Core Cell

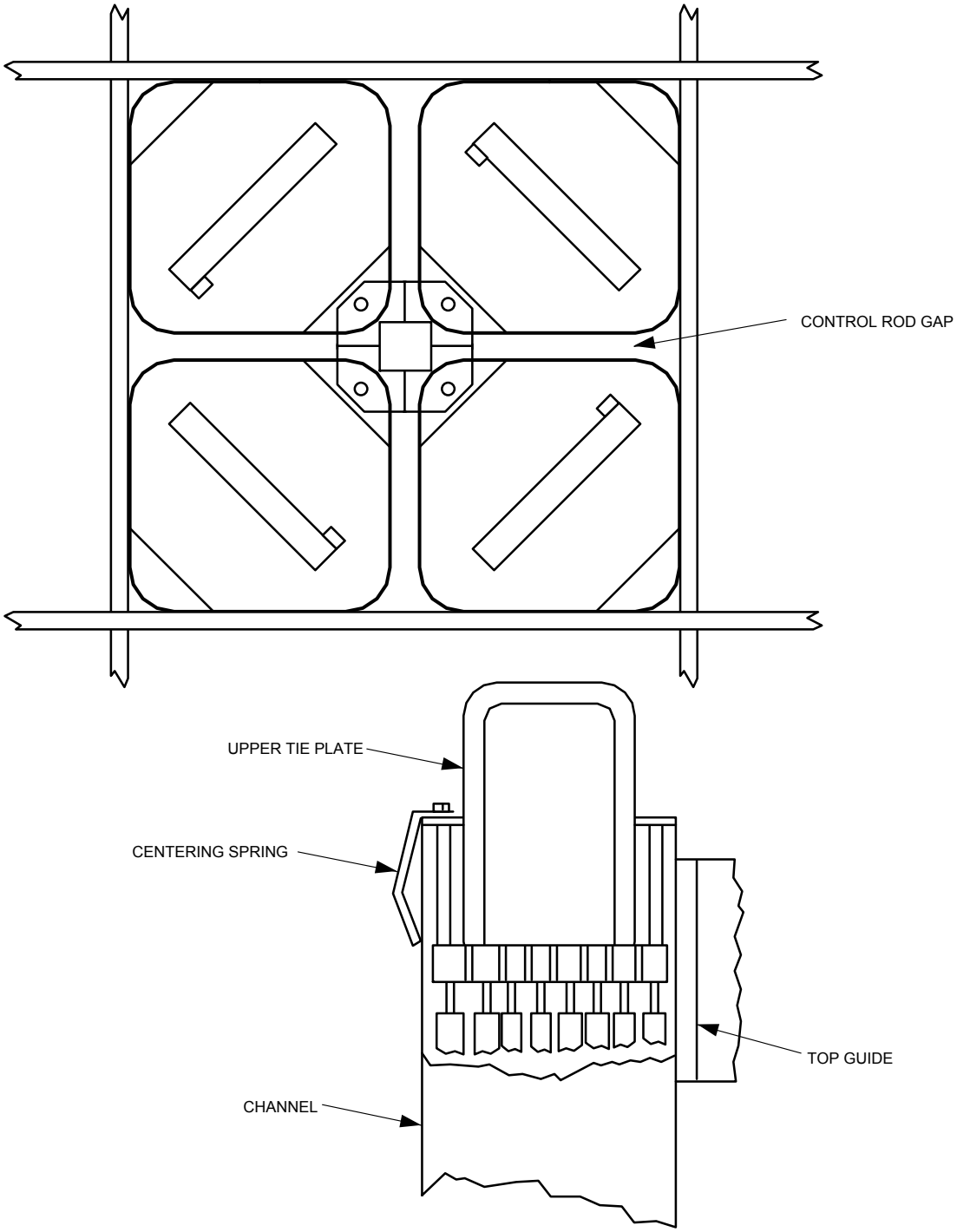

Figure 1-2. Schematic of Four Bundle Cell Arrangement

2. Fuel Mechanical Design

This section contains a description of the fuel thermal–mechanical analyses bases currently in use in the U.S. There were significant changes to the thermal-mechanical design bases in GESTAR II Revision 17. Therefore, the fuel 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 2–2. Reference 2-3 provides a detailed description and 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 for each respective product line.

The format of this section corresponds to Standard Review Plan 4.2 in NUREG–0800. The design bases for each of the fuel system damage, failure, and coolability criteria identified in SRP Section II.A are provided in Subsection 2.2. A description of the fuel assembly (SRP Section II.B) appears in the fuel product specific GESTAR II Compliance Report and in Referenc 2-2 and 2-3. The design evaluations for each of the fuel system damage, failure, and coolability criteria identified in SRP Section II.C are also provided in Subsection 2.2. Fuel assembly testing, inspection, and surveillance plans (SRP Section II.D) are documented in Subsection 2.3.

2.1 Fuel Assembly Description

Descriptions of the fuel assemblies (including fuel rods, water rods, other fuel assembly components and channels) to which the fuel thermal–mechanical (T-M) design bases described in this section apply are given in Reference 2–2 for older fuel product lines. Reference 2-3 provides a detailed description and 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.

2.2 Design Bases, Limits, and Evaluations

Operating limits are established to ensure that actual fuel operation is maintained within the fuel rod thermal–mechanical design and safety analysis bases. These operating limits define the maximum allowable fuel pellet operating power level as a function of fuel pellet exposure. Lattice local power and exposure peaking factors may be applied to transform the maximum allowable fuel pellet power level into Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for individual fuel bundle designs. Otherwise, the LHGR limit is monitored directly. Peak pellet exposure (PPE) for any fuel design is limited to [[ ]] or less, if necessary, to meet the fuel design criteria. Any specific restrictions on the exposure for a specific fuel product will be documented in the GESTAR II compliance report for that product (See Section 1.4).

For the near future, there are two NRC approved Global Nuclear Fuel (GNF) T-M design methodologies that may be used in GNF fuel designs: the older GESTR–MECHANICAL (GESTR-M), and the recently approved PRIME methodology (Reference 2-19). Both

methodologies are valid, subject to their specific limitations. GNF will transition from the GESTR-M to the PRIME T-M methodology basis as quickly as practical. Beginning with the GNF2 fuel product line, the fuel T-M design will use the PRIME methodology. The GESTR-M basis for the GNF2 fuel product as defined in Amendment 32 (Reference 2-20) continues to be a valid basis for GNF2 for the limited lifetime as it is the current basis for some first cycle GNF2 reloads. Fuel products preceding GE14 (e.g., GE11 and 12), which are currently operating, may continue to use the GESTR-M basis. GNF is no longer loading these older fuel products, but some may remain in operating plants for several more cycles. GNF will implement the PRIME T-M basis for the GE14 fuel product line, including GE14 currently in operation, in the reload workscope for new fuel cycle designs initiated following the completion of the downstream codes implementation activities as described in Supplement 4 to NEDC-33173P, *Applicability of GE Methods to Expanded Operating Domains*. (Reference 2-21)

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2.2.1 Fuel System Damage

This subsection applies to normal operation and anticipated operational occurrences except for Subsections 2.2.1.3, 2.2.1.6 and 2.2.1.7, which apply to normal operation only.

2.2.1.1 Stress/Strain

2.2.1.1.1 Bases

The fuel assembly components are evaluated to ensure that the fuel will not fail due to stresses or strains exceeding the fuel assembly component mechanical capability.

2.2.1.1.2 Limits

The limit is patterned after ANSI/ANS–57.5–1981 (Reference 2–5). 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 effective stress or strain is determined by applying the distortion energy theory. The limit is the material ultimate stress or strain. The limit used is that the Design Ratio must be less than or equal to one:

$$\text{Design Ratio} \leq 1.0$$

2.2.1.1.3 Evaluations

[[

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2.2.1.2 Fatigue

2.2.1.2.1 Bases

The fuel assembly and the fuel rod cladding are evaluated to ensure that strains due to cyclic loadings will not exceed the fatigue capability.

2.2.1.2.2 Limits

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2.2.1.2.3 Evaluations

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2.2.1.3 Fretting Wear

2.2.1.3.1 Bases

The fuel assembly is evaluated to ensure that fuel will not fail due to fretting wear of the assembly components.

2.2.1.3.2 Limits

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2.2.1.3.3 Evaluations

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2.2.1.4 Oxidation, Hydriding and Corrosion Products

2.2.1.4.1 Oxidation and Corrosion Products

2.2.1.4.1.1 Bases

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2.2.1.4.1.2 Limits

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2.2.1.4.1.3 Evaluations

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2.2.1.4.2 Hydriding

2.2.1.4.2.1 Bases

The fuel rod is evaluated to ensure that failure will not occur due to internal cladding hydriding.

2.2.1.4.2.2 Limits

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2.2.1.4.2.3 Evaluations

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2.2.1.5 Dimensional Changes

2.2.1.5.1 Bases

The fuel rod is evaluated to ensure that fuel rod bowing does not result in fuel failure due to boiling transition.

2.2.1.5.2 Limits

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2.2.1.5.3 Evaluations

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2.2.1.6 Internal Gas Pressure

2.2.1.6.1 Bases

The fuel rod is evaluated to ensure that the effects of fuel rod internal pressure during normal steady–state operation will not result in fuel failure due to excessive cladding pressure loading.

2.2.1.6.2 Limits

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2.2.1.6.3 Evaluations

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2.2.1.7 Hydraulic Loads

2.2.1.7.1 Bases

The fuel assembly is evaluated to ensure that interference sufficient to prevent control blade insertion will not occur.

2.2.1.7.2 Limits

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2.2.1.7.3 Evaluations

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Two separate aspects of channel box deflection are considered: channel bulge and channel bow. Channel bulge is addressed in Reference 2–4. In response to an NRC question on initial cores, Reference 2–12 provides supplementary information to Reference 2–4, and also contains a discussion of the program GE recommends to utilities obtaining an operating license after May 1982. Channel bow effects on thermal margins are included in Reference 2–16. References 2–4, 2–12 and 2–16 apply only to channels supplied by General Electric.

Channel bow and bulge are material dependent phenomenon. In addition to the standard alloys Zircaloy-2 and Zircaloy-4, GNF also fabricates channels made of a fully recrystallized Zr-Sn-Nb-Fe alloy called NSF. The NRC approved NSF for use in channels and the accepted version of the Topical Report (TR) is included as Reference 2-23. The key difference between the NSF material and the Zircaloy-2 or Zircaloy-4 alloy is the reduction in both fluence and shadow corrosion induced bow. No changes in the channel design requirements or processes are necessary as result of the change to NSF.

The NSF channel bow effects on thermal margins are addressed in the Reference 2-23 TR. The bow inputs for NSF channels, as well as for the transition cycles, were approved in accordance with the Safety Evaluation (SE) included in Reference 2-23.

2.2.1.8 Control Rod Reactivity

Control rod reactivity limits are discussed in Subsections 3.1 and 3.2.4.

2.2.2 Fuel Rod Failure

Subsections 2.2.2.1 through 2.2.2.3 apply to normal operation; Subsections 2.2.2.4, 2.2.2.5 and 2.2.2.7 apply to anticipated operational occurrences; and Subsections 2.2.2.6, 2.2.2.8 and 2.2.2.9 apply to postulated accidents.

2.2.2.1 Hydriding

Hydriding is discussed in Subsection 2.2.1.4.2 of this document.

2.2.2.2 Cladding Collapse

2.2.2.2.1 Bases

The fuel rod is evaluated to ensure that fuel rod failure due to cladding collapse into a fuel column axial gap will not occur.

2.2.2.2.2 Limits

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2.2.2.2.3 Evaluations

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2.2.2.3 Fretting Wear

Fretting wear is discussed in Subsection 2.2.1.3 of this document.

2.2.2.4 Overheating of Cladding

Overheating of the cladding is addressed in Subsection 4.3.1 of this document.

2.2.2.5 Overheating of Pellets

2.2.2.5.1 Bases

The fuel rod is evaluated to ensure that fuel rod failure due to fuel melting will not occur.

2.2.2.5.2 Limits

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2.2.2.5.3 Evaluations

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2.2.2.6 Excessive Fuel Enthalpy

Excessive fuel enthalpy is discussed in the country–specific supplement to this document.

2.2.2.7 Pellet–Cladding Interaction

2.2.2.7.1 Bases

The fuel rods are evaluated to ensure that fuel rod failure due to pellet–clad mechanical interaction will not occur.

2.2.2.7.2 Limits

[[

]] For fuel product lines prior to PRIME implementation, as defined by the compliance reports in Section 1.4, [[

]] For fuel product lines which have implemented PRIME, as defined by the compliance reports in Section 1.4, the strain criteria is defined for two exposure ranges:

Range 1 – [[

]]

Range 2 – [[

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[[

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2.2.2.7.3 Evaluations

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2.2.2.8 Bursting

Bursting is addressed in the country–specific supplement to this document.

2.2.2.9 Mechanical Fracturing

2.2.2.9.1 Bases

The fuel assembly is evaluated under Safe Shutdown Earthquake and Loss–of–Coolant Accident loading conditions to ensure that loss of fuel assembly coolability, and interference to the degree that control blade insertion is prevented, will not occur.

2.2.2.9.2 Limits

The limits used for this evaluation are described in Reference 2–7 and Subsections 2.2.1.1.2 and 2.2.1.2.2.

2.2.2.9.3 Evaluations

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2.2.2.9.3.1 Dynamic Analysis and Component Seismic Loads

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2.2.2.9.3.2 LOCA Loads

The pressure differentials on the BWR/4–6 lower tieplates, upper tieplates, and spacers resulting from a recirculation line break or from a steam line break are greater than or equal to the corresponding pressure differentials for BWR/2, 3 fuel assembly components. Water rod pressure differentials are insignificantly small. The methodology for evaluating LOCA pressure differentials for BWR/2, 3 fuel is similar to that used for BWR/4–6 fuel assemblies.

2.2.2.9.3.3 Component Evaluations

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2.2.3 Fuel Coolability

This subsection applies to postulated accidents.

2.2.3.1 Cladding Embrittlement

Cladding embrittlement is addressed in the country–specific supplement to this document.

2.2.3.2 Violent Expulsion of Fuel

Violent expulsion of fuel is addressed in the country–specific supplement to this document.

2.2.3.3 Generalized Cladding Melting

Generalized cladding melting is bounded by the cladding embrittlement criteria of Subsection 2.2.3.1.

2.2.3.4 Fuel Rod Ballooning

Fuel rod ballooning is addressed in the country–specific supplement to this document.

2.2.3.5 Structural Deformation

Structural deformation is addressed in Subsection 2.2.2.9 of this document.

2.3 Testing, Inspection and Surveillance Plans

2.3.1 Testing and Inspection of New Fuel

The General Electric quality assurance program is documented in Reference 2–15. The reference covers the quality control areas associated with the manufacture and inspection of new fuel for the areas of:

1. Material and component procurement.
2. Fabrication and assembly of components and systems.
3. Inspection and testing.
4. Cleaning, packaging, and shipping.
5. Installation and erection of systems and components.
6. Pre–operational and startup testing.

The reference further describes that these quality control plans are implemented using the following document types:

1. Acceptance standards.
2. Audit plans and procedures.
3. Calibration procedures.

4. Corrective action procedures.
5. Design control procedures.
6. Engineering drawings and specifications.
7. Handling, storage, packing and shipping procedures.
8. Inspection instructions.
9. Inspection and tester stamp control procedures.
10. Material identification and control procedures.
11. Measuring and test equipment control procedures.
12. Nonconforming material control procedures.
13. Pre–production quality evaluation procedures.
14. Process and personnel qualification procedures.
15. Process control procedures.
16. Product/process quality plans.
17. Purchased material quality control plans.
18. Quality assurance document control procedures.
19. Quality assurance records specifications and instructions.
20. Quality control standards instructions.
21. Receiving inspection plans.
22. Shipment release control procedures.
23. Supplier evaluation and selection procedures.
24. Test instructions.

The quality assurance program described in Reference 2–15 applies explicitly to the Wilmington manufacturing site; however, similar quality assurance programs are implemented in the overseas manufacturing facilities.

2.3.2 On–Line Fuel System Monitoring

Provided by Applicant.

2.3.3 Post–Irradiation Surveillance

General Electric has an active program of interim and post–irradiation surveillance of both lead use assemblies and developmental BWR fuel. The schedule of inspection is contingent on both the availability of the fuel as influenced by plant operation and the expected value of the information to be obtained.

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2.4 References

2–1 Letter from J. S. Charnley (GE) to R. Lobel (NRC), *Implementation of GESTR–M*, April 24, 1984.

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

2–3 *General Electric Fuel Bundle Designs Evaluated with TEXICO/CLAM Analyses Bases*, April 1986 (NEDE–31151–P).

2–4 *BWR Fuel Channel Mechanical Design and Deflection*, NEDE–21354–2–P (Proprietary) and NEDO–21354–2, July 1977.

2–5 American National Standard for Light Water Reactors Fuel Assembly Mechanical Design and Evaluation, American Nuclear Society Standards Committee Working Group ANS 57.5, ANSI/ANS–57.5–1981.

2–6 W. G. Jameson, Jr., *Fuel Assembly Evaluation of Shipping and Handling Loadings*, NEDE–23542–P (Proprietary), March 1977.

2–7 *Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss–of–Coolant Accident (LOCA) Loadings (Amendment No. 3)*, NEDE–21175–3–P–A (Proprietary) and NEDO–21175–3–A, October 1984.

2–8 Letter from J. S. Charnley (GE) to M. S. Dunenfeld (NRC), *1984 Fuel Experience Report*, October 14, 1985.

2–9 K. W. Hill, et al., *Effect of a Rod Bowed to Contact on Critical Heat Flux in Pressurized Water Reactor Rod Bundles*, American Society of Mechanical Engineers Publication 75–WA/ HT–77.

2–10 E. S. Markowski, et al., *Effect of Rod Bowing on CHF in PWR Fuel Assemblies*, American Society of Mechanical Engineers Publication 77–HT–91.

2–11 Letter from R. L. Gridley (GE) to D. G. Eisenhut (NRC), *Evaluation of Potential Fuel Bundle Lift at Operating Reactors*, July 11, 1977.

2–12 Letter from G. G. Sherwood (GE) to D. G. Eisenhut (NRC), *In the Matter of 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II) Docket No. 50–447*, February 2, 1983.

2–13 *Creep Collapse Analysis of BWR Fuel Using SAFE–COLAPS Model*, NEDO–20606A and NEDE–20606–PA (Proprietary), August 1976.

2–14 Memo from L. S. Rubenstein (NRC) to R.L. Tedesco (NRC), *SER Input for WNP–2*, February 24, 1982.

2–15 *GE Hitachi Nuclear Energy Quality Assurance Program Description, NEDO-11209-A, Most Recent Accepted Version*.

2–16 Letter, J. S. Charnley (GE) to R. C. Jones, Jr. (NRC), *Fuel Channel Bow Assessment*, MFN086–89, November 15, 1989.

2–17 *GE11 Critical Power Test with Rod Bow to Contact*, NEDE–31829P, April 1990.

2-18 Robert A. Rand, *Cladding Creep Collapse Licensing Topical Report*, NEDC-33139P-A, July 2005.

2-19 GNF Licensing Topical Report, *The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance*, Technical Bases - NEDC-33256P-A, Qualification - NEDC-33257P-A, and Application Methodology - NEDC-33258P-A, September 2010.

2-20 Letter from AA Lingenfelter (GNF) to Document Control Desk (NRC), Subject: *Amendment 32 To NEDE–24011–P, General Electric Standard Application For Reactor Fuel (GESTAR II)*, FLN-2008-011, October 15, 2008, and Letter from TB Blount (NRC) to AA Lingenfelter (GNF), Subject: *Final Safety Evaluation for Amendment 32 to Global Nuclear Fuel Topical Report NEDE-24011-P General Electric Standard Application for Reload (TAC NO. MD9939)*, July 30, 2009.

2-21 *Implementation of PRIME Models and Data in Downstream Methods,* NEDO-33173 Supplement 4-A, Revision 1, November 2012.

2-22 Letter from JC Kinsey (GEH) to Document Control Desk (NRC), Subject: *Response to Portion of NRC Request for Additional Information Letter No. 110 - Related to ESBWR Design Certification Application - RAI Numbers 4.2-2 Supplement 3, 4.2-4 Supplement 2 and 4.8-6 Supplement 1*, MFN 08-347, May 9, 2008.

2-23 *Application of NSF to GNF Fuel Channel Designs*, NEDE-33798P-A, Revision 1, September 2015.

3. Nuclear Design

This section describes the nuclear design basis and the models used to analyze the core and fuel. The nuclear design bases for older fuel product lines are given in Reference 3–2. Reference 3-3 provides a detailed description and the nuclear 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. All GE and GNF fuel designs either meet the criteria of Subsection 1.1.3 or are separately approved by the NRC.

3.1 Design Bases

The design bases are those that are required for the plant to operate, meeting all safety requirements. Safety design bases fall into two categories: (1) the reactivity basis, which prevents an uncontrolled positive reactivity excursion, and (2) the overpower bases, which prevent the core from operating beyond the fuel integrity limits.

3.1.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being made subcritical at any time or at any core condition with the highest worth control rod fully withdrawn.

3.1.2 Overpower Bases

The Technical Specification limits on Minimum Critical Power Ratio (MCPR), the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and the Linear Heat Generation Rate (LHGR) are determined such that the fuel will not exceed required licensing limits during abnormal operational occurrences or accidents.

3.2 Description

The BWR core design consists of a light–water moderated reactor, fueled with slightly enriched uranium–dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The BWR design provides a system for which reactivity is reduced by an increase in the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the BWR system. Any system input that increases reactor power, either in a local or gross sense, produces additional steam voids that reduce reactivity and thereby reduce the power.

3.2.1 Nuclear Design Description

The reference loading pattern for each cycle is documented in the FSAR or in the Supplemental Reload Licensing Report.

The reference loading pattern is the basis for all fuel licensing. It is designed with the intent that it will represent, as closely as possible, the actual core loading pattern; however, there will be occurrences where the number and/or types of bundles in the reference design and the actual core loading do not agree exactly.

Any differences between the reference loading pattern and the actual loading pattern are evaluated as described in Section 3.4.

3.2.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MAPLHGR, LHGR operating limit and MCPR (defined in Table 3–1), limit unacceptable core power distributions.

3.2.2.1 Power Distribution Measurements

The techniques for measurement of the power distribution within the reactor core, together with instrumentation correlations and operation limits, are discussed in Reference 3–1.

3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distributions is discussed in References 3–4, 3–5, 3–16, 3–17 and 3–18.

GNF has changed the name of the updated core monitoring system from 3D-MONICORE to ACUMEN. However, from a technical analysis standpoint, the inputs, internal algorithms and models, and key outputs are the same as 3D-MONICORE. The change in name from 3D-MONICORE to ACUMEN does not affect the methodology and uncertainty basis approved by the NRC.

GNF has chosen to not change the current 3D-MONICORE citations to ACUMEN. 3D-MONICORE is mentioned in References 3-17 and 3-18.

3.2.2.3 Power Distribution Anomalies

The power distribution anomaly resulting from a fuel loading error does not generally result in the limiting delta-CPR compared to the other events analyzed for each reload cycle. As such, the event has a very remote likelihood of resulting in fuel failures. The fuel loading error is analyzed as an Infrequent Incident when appropriate core verification procedures are utilized to ensure the correct arrangement of the core following fuel loading. Fuel loading error is discussed further in the country–specific supplement to this document.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the in–core instrumentation system, together with the on–line computer, provides the operator with prompt information on the

power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods and flow, then the total core power would have to be reduced.

3.2.3 Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor. The coefficients of interest, relative to BWR systems, are discussed here individually.

There are two primary reactivity coefficients that characterize the dynamic behavior of boiling water reactors; these are the Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the BWR are a power reactivity coefficient and a temperature coefficient. The power coefficient is a combination of the Doppler and void reactivity coefficients in the power operating range, and the temperature coefficient is merely a combination of the Doppler and moderator temperature coefficients. Power and temperature coefficients are not specifically calculated for reload cores.

3.2.3.1 Doppler Reactivity Coefficient

The Doppler coefficient is of prime importance in reactor safety. The Doppler coefficient 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 in question. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among BWR designs. For most structural and moderator materials, resonance absorption is not significant, but in U–238 and Pu–240 an increase in temperature produces a comparatively large increase in the effective absorption cross–section. The resulting parasitic absorption of neutrons causes a significant loss in reactivity. In BWR fuel, in which approximately 97% of the uranium in UO_2 is U–238, the Doppler coefficient provides an immediate negative reactivity response that opposes increased fuel fission rate changes.

Although the reactivity change caused by the Doppler effect is small compared to other power–related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe power excursions are those associated with rod drop accidents. A local Doppler feedback associated with a 3000°F to 5000°F temperature rise is available for terminating the initial excursion.

The Doppler coefficient is determined using the theory and methods described in Reference 3–6.

3.2.3.2 Moderator Void Coefficient

The moderator void coefficient should be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation. In addition, the void coefficient in a BWR has the ability to flatten the radial power distribution and to provide ease of reactor control due to the void feedback mechanism. The overall void coefficient is always negative over the complete operating range since the BWR design is undermoderated.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analyses, is presented in Reference 3–6.

3.2.4 Control Requirements

The General Electric BWR control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the plant operation. The shutdown capability is evaluated assuming a cold, xenon–free core.

3.2.4.1 Shutdown Reactivity

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code (see Section 3.3) to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon–free condition in order to ensure that the calculated values are conservative. Further discussion of the uncertainty of these calculations is given in References 3–7 and 3–8.

As exposure accumulates and burnable poison depletes in the lower exposure fuel bundles, an increase in core reactivity may occur. The nature of this increase depends on specifics of fuel loading and control state.

The cold k_{eff} is calculated with the strongest control rod out at various exposures through the cycle. A value R is defined as the difference between the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point in the cycle is equal to or less than:

$$k_{eff} = k_{eff}\ (Strongest\ rod\ withdrawn)_{BOC} + R,$$

where

R is always greater than or equal to 0. The value of R includes equilibrium Sm.

The calculated values of k_{eff} with the strongest rod withdrawn at BOC and of R are reported in the FSAR or in the supplemental reload licensing report. For completeness, the uncontrolled

k_{eff} and fully controlled k_{eff} values are also reported in the FSAR or in the supplemental reload licensing report.

3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia–urania fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

3.2.4.3 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, to a subcritical condition with the reactor in the most reactive xenon–free state with all of the control rods in the full–out condition. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full–power and cold, xenon–free condition. The shutdown capability of the SLCS is given in the FSAR or the supplemental reload licensing report.

3.2.5 Criticality of Reactor During Refueling

The core is subcritical at all times.

3.2.6 Stability

3.2.6.1 Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by: (1) never having observed xenon instabilities in operating BWRs, (2) special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability, and (3) calculations. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

Analysis and experiments conducted in this area are reported in Reference 3–9.

3.2.6.2 Thermal Hydraulic Stability

This subject is covered in the country–specific supplement to this document.

3.3 Analytical Methods

The nuclear evaluations of all General Electric BWR cores are performed using the analytical tools and methods described in this section. There are two sets of procedures available for fuel design and licensing analysis: GENESIS and GEMINI. The nuclear physics methods described in References 3–4, 3–7, 3–10 and 3–11 are utilized as part of the GENESIS group. The advanced physics methods described in References 3–5 and 3–16 are utilized as part of the GEMINI group. The particular procedure that can be utilized is optional. In either case,

the nuclear evaluation procedure is best addressed as two parts: lattice analysis and core analysis.

The lattice analyses are performed during the bundle design process. The results of these single bundle calculations are reduced to "libraries" of lattice reactivities, relative rod powers, and few group cross–sections as functions of instantaneous void, exposure, exposure–void history, exposure–control history, control state, and fuel and moderator temperature, for use in the core analysis. These analyses are dependent upon fuel lattice parameters only and are, therefore, valid for all plants and cycles to which they are applied.

The core analysis is unique for each cycle. It is performed in the months preceding the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three–dimensional Boiling Water Reactor Simulator code, which computes power distributions, exposure, and reactor thermal–hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

3.4 Final Core Loading Pattern Comparison (Reload Cores)

3.4.1 Introduction and Bases

Because the reload licensing process requires an assumption as to the condition of the core at the end of the previous cycle, the final core loading pattern may not be identical to the reference core loading pattern. To ensure that the results of the licensing analysis performed on the reference core loading pattern are applicable to the final core loading pattern, certain key parameters, which affect the licensing analysis, are examined to ensure that there is no significant adverse effect. Only when this examination has been completed and determined that the final core loading pattern satisfies the licensing basis will the core be operated.

3.4.2 Acceptable Deviations from the Reference Core Loading Pattern

The key parameters that measure the deviation between the reference core loading pattern and the final core loading pattern have been identified and are discussed in this section. The key parameters described below were determined to be important to the licensing analysis, and the allowed deviations are defined in the following sections. Section 3.4.3 must be followed if any of the following criteria is not met.

3.4.2.1 Previous EOC Exposure

The previous end-of-cycle (EOC) cycle exposure deviation (i.e., actual EOC cycle exposure versus assumed EOC cycle exposure) must be within the range considered in the licensing analysis.

3.4.2.2 Core Average Axial Exposure Distribution

The final core loading pattern axial exposure shape is demonstrated to be bounded by the axial exposure shapes used for the licensing analysis.

3.4.2.3 Number and Fuel Type of New Bundles

The number of new bundles for each fuel type loaded in the final core loading pattern must match the number in the reference core loading pattern.

3.4.2.4 Number and Fuel Type of Irradiated Bundles

If the number of available irradiated bundles of a given fuel type is less than specified in the reference core loading pattern, bundles of a different fuel type but of lower reactivity may be substituted.

Note that a bundle is not available for loading into the core if it is damaged or is not suitable for continued operation.

3.4.2.5 Locations of New Bundles

A new bundle of a particular fuel type must be loaded only into the location that has been designated to receive this new bundle fuel type in the reference core loading pattern.

3.4.2.6 Locations of Non-Peripheral Irradiated Bundles

Criterion 1: The locations of each individual non-peripheral irradiated bundle as designated in the reference core loading pattern is preserved, except for changes necessitated by the available inventory or necessitated by a different operating history relative to the reference core loading pattern.

Criterion 2: If a change is necessitated, an irradiated bundle may be replaced with an irradiated bundle of lower reactivity for regions of high importance. Shuffles to remedy a different operating history can result in replacing an irradiated bundle with an irradiated bundle of higher reactivity for regions of low importance (within three rows from the periphery).

Regions of high importance are defined as locations that are at least four rows in from the periphery.

3.4.2.7 Locations of Peripheral Irradiated Bundles

Bundles on the periphery may be shuffled to other locations on the periphery relative to the reference core loading pattern.

3.4.2.8 Shutdown Margin

Adequate cold shutdown margin is demonstrated for the final core loading pattern.

3.4.3 Re–Examination of Bases

If the criteria of Section 3.4.2 are not met, a re–examination of the reference core loading pattern licensing analysis is performed based on the final core loading pattern. This

re-examination may consist of evaluating parameters that affect the licensing analysis or re-performing the licensing analysis for the limiting condition. The licensing analyses re-examined are:

1. Cold shutdown margin
2. Standby liquid control system shutdown margin
3. Safety Limit MCPR
4. Core-wide AOOs
5. Rod withdrawal error
6. Misloaded fuel assembly (when analyzed as an AOO)
7. Stability

The listed licensing analyses were chosen because they are reported in the cycle-specific licensing documentation (see Appendix A of the country-specific supplement) and can be affected by changes between the final core loading pattern and the reference core loading pattern.

The evaluations in this section will either conclude that the current licensing documentation is adequate or that the licensing documentation needs to be updated.

3.5 Reactivity of Fuel in Storage

The basic criterion in 10CFR50.68 associated with the storage of both irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal and abnormal conditions will be ≤ 0.95 for GE low-density and high–density racks over a temperature range of 4°C to 100°C. For cases where optimum moderation is a credible event for the storage of fresh fuel in GE low-density racks, the maximum k-effective corresponding to the optimum moderation condition will be ≤ 0.98 per 10CFR50.68. These storage criteria will be satisfied if the cold uncontrolled in core k_∞ for a lattice calculated in the normal reactor core configuration meets the following condition for General Electric designed fuel storage racks.

(a) $k_\infty \leq 1.28$ for low-density spent fuel storage racks with an interrack spacing ≥ 11.70 inches.

(b) $k_\infty \leq 1.33$ for high-density spent fuel storage racks with an interrack spacing ≥ 6.563 inches.

(c) $k_\infty \leq 1.31$ for low-density new fuel vault storage racks with an interrack spacing ≥ 10.50 inches.

If the new fuel vault storage racks are in use and there are no administrative controls and/or design features to prevent optimum moderation from occurring, a checkerboard array must be employed where only one out of every three storage locations in either linear direction contains a fuel bundle.

3.6 References

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Table 3–1
Definition of Fuel Design Limits

Linear Heat Generation Rate (LHGR) Operating Limit The LHGR operating limit is the maximum linear heat generation rate expressed in kW/ft for the fuel rod with the highest surface heat flux at a given nodal plane in the bundle. The LHGR operating limit is bundle type dependent. The LHGR operating limit can be monitored to assure that all mechanical design requirements will be met.
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the plant Technical Specifications for that fuel type. This parameter is obtained by averaging the linear heat generation rate over each fuel rod in the plane, and its limiting value is selected such that (a) the peak clad temperature during the design basis loss–of–coolant accident will not exceed 2200°F in the plane of interest, and (b) all fuel rod thermal–mechanical design limits specified in Section 2 will be met if the exposure–dependent LHGR operating limit is not monitored for that purpose.
Minimum Critical Power Ratio (MCPR) The critical power ratio 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 exists 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.
Operating Limit MCPR The MCPR operating limit is the minimum CPR allowed by the plant Technical Specifications for a given bundle type. The minimum CPR is a function of several parameters, the most important of which are bundle power, bundle flow and bundle R–factor. The R–factor is dependent upon the local power distribution and details of the bundle mechanical design (Reference 3–15). The limiting value of CPR is selected for each bundle type such that, during the most limiting event of moderate frequency, the calculated CPR in that bundle is not less than the safety limit CPR. The MCPR operating limit is attained when the bundle power, R–factor, flow, and other relevant parameters combine to yield the technical specification value.

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:

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

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

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

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:

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

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

where

ΔP_E = elevation pressure drop

Δ_L = incremental length

$\overline{\rho}$ = 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

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 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.

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

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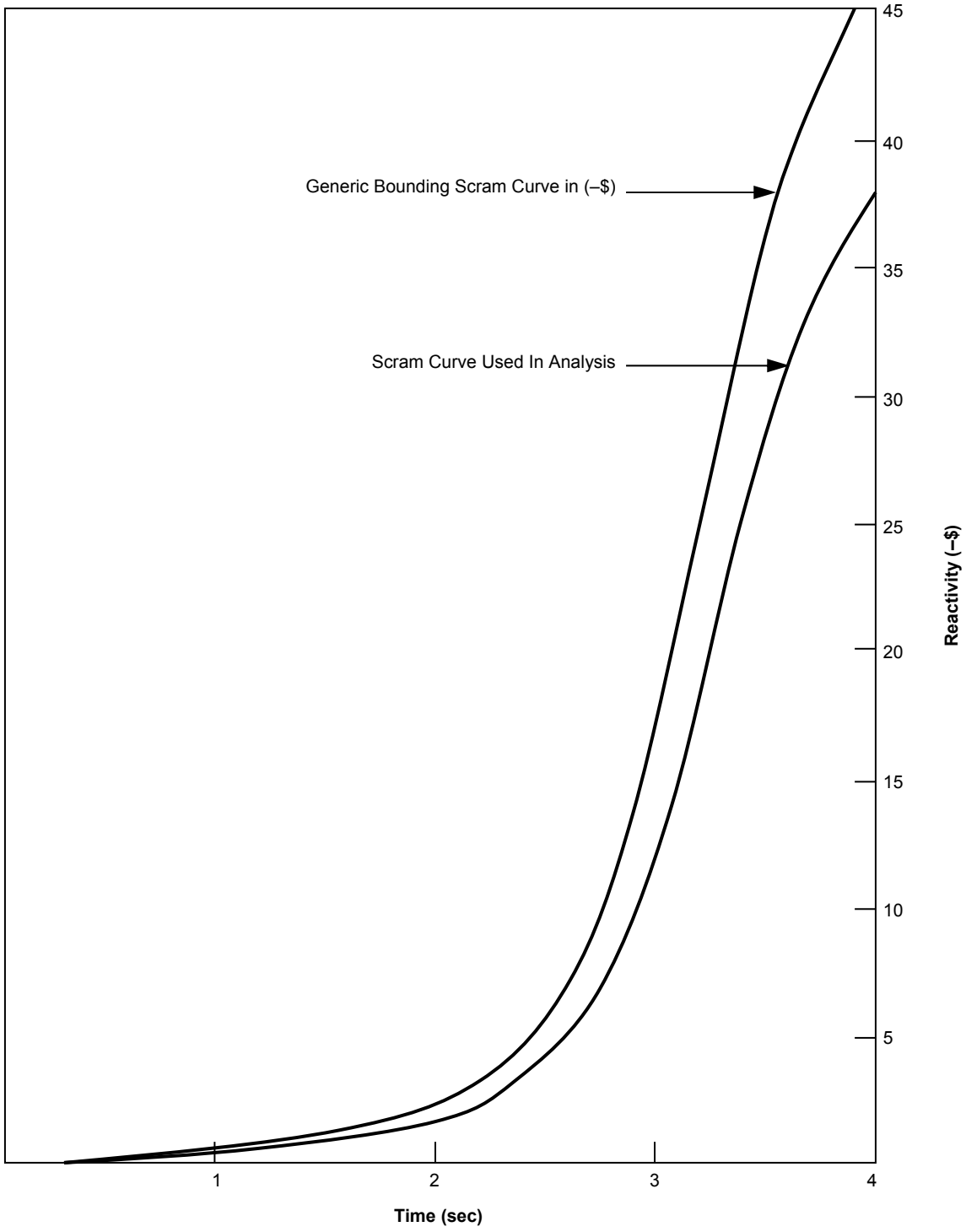

Figure 4–1. Transient Analysis Input–Scram Reactivity (REDY Events)

Appendix A

Safety Analysis Report Road Map

Appendix A
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(Only FSAR Sections Applicable to Fuel are Addressed)

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A. Summary

The purpose of this appendix is to provide a road map for incorporating the nuclear fuel design and analysis characteristics described in this document into the standard Final Safety Analysis Report (FSAR) format. This format is consistent with that specified by the Nuclear Regulatory Commission (NRC) in References A–1 and A–2.

Only those subsections that pertain to fuel design and analysis are addressed. For each of those subsections either an approved document or a particular reference to the GESTAR II document (or its country–specific supplement) is given.

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 A-3 and A-4. A detailed description of the 8x8 and 8x8R fuel designs is given in Reference A-4 while the newer designs are described in Reference A-3. 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 A-3 is found in the plant-cycle specific Fuel Bundle Information Report (FBIR).

Utilities that follow the standard FSAR format need only reference the appropriate section in this appendix in the corresponding section of their FSAR. This minimizes the approval effort, since GESTAR II has been generically approved for use with GE–designed BWRs in the United States and many foreign countries.

Only the specific sections in the plant FSAR related to fuel design or analysis are listed in this appendix. All other sections are provided by the applicant.

A.1.2 General Plant Description

A.1.2.2.3.1 Reactor Core and Control Rods

The reactor fuel and core designs are described in References A–3 or A–4. Fuel bundle design information for bundles more recent than those included in Reference A-3 is found in the compliance report for each fuel product line (See Section 1.4) and in the plant-cycle specific FBIR. The design of the control rods is described in the plant–specific FSAR.

A.1.3 Comparison Tables

A.1.3.1 Comparison with Similar Facility Designs

A comparison of plant–specific fuel information is usually documented in Table 1.3–1 of the FSAR. The fuel information for this table is taken from References A–3 or A–4. Fuel bundle design information for bundles more recent than those included in Reference A-3 is found in

the compliance report for each fuel product line (See Section 1.4) and in the plant-cycle specific FBIR.

A.4.2 Fuel System Design

The content of this section corresponds to Regulatory Guide 1.70 and Standard Review Plan 4.2 (References A–1 and A–2). Most of the information presented will be by reference to the approved subsections of this document.

A.4.2.1 Design Bases

The design bases for each of the fuel system damage, failure, and coolability criteria identified in Section II.A of Standard Review Plan 4.2, except control rod reactivity, are provided in Subsection 2.2 of this document. Control rod reactivity is discussed in Reference A–5. Additional information required by Reference A–1 is to be provided by the Applicant.

A.4.2.2 Description and Design Drawings

The fuel assembly is described in References A–3 or A–4. Fuel assembly design information for bundles more recent than those included in Reference A-3 is found in the compliance report for each fuel product line (See Section 1.4) and in the plant-cycle specific FBIR. The reactivity control assembly description is to be provided by the applicant.

A.4.2.3 Design Evaluation

The design evaluations for each of the fuel system damage, failure, and coolability criteria identified in Section II.C of Standard Review Plan 4.2, except control rod reactivity, are provided in Subsection 2.2. Control rod reactivity is discussed in Reference A–5. Additional information to be provided by the Applicant.

A.4.2.4 Testing Inspection and Surveillance Plans

Fuel assembly testing, inspection and surveillance plans are documented in Subsection 2.3 of this document.

A.4.3 Nuclear Design

The content of this section corresponds to Regulatory Guide 1.70 and Standard Review Plan 4.3 (References A–1 and A–2). Most of the information presented will be by reference to the approved subsections of this document.

A.4.3.1 Design Bases

See Section 3.1.

A.4.3.1.1 Reactivity Basis

See Section 3.1.1.

A.4.3.1.2 Overpower Bases

See Section 3.1.2.

A.4.3.2 Description

See Section 3.2.

A.4.3.2.1 Nuclear Design Description

See Section 3.2.1. The reference core loading pattern is to be provided by the applicant in the format shown in Appendix A of the country–specific supplement.

A.4.3.2.2 Power Distribution

See Section 3.2.2.

A.4.3.2.2.1 Power Distribution Calculations

To be provided by Applicant.

A.4.3.2.2.2 Power Distribution Measurements

See Section 3.2.2.1.

A.4.3.2.2.3 Power Distribution Accuracy

See Section 3.2.2.2.

A.4.3.2.2.4 Power Distribution Anomalies

See Section 3.2.2.3.

A.4.3.2.3 Reactivity Coefficients

See Section 3.2.3.

A.4.3.2.4 Control Requirements

See Section 3.2.4.

A.4.3.2.4.1 Shutdown Reactivity

See Section 3.2.4.1.

The cold shutdown margin for the reference core loading pattern is to be supplied by the applicant in the format shown in Appendix A of the country–specific supplement.

A.4.3.2.4.2 Reactivity Variations

See Section 3.2.4.2.

A.4.3.2.5 Control Rod Patterns and Reactivity Worths

To be provided by Applicant.

A.4.3.2.6 Criticality of Reactor During Refueling

See Section 3.2.5.

A.4.3.2.7 Stability

A.4.3.2.7.1 Xenon Transients

See Section 3.2.6.1.

A.4.3.2.7.2 Thermal Hydraulic Stability

See Section S4.

A.4.3.2.8 Vessel Irradiations

To be provided by Applicant.

A.4.3.3 Analytical Methods

See Section 3.3.

A.4.3.4 Changes

General Electric fuel design philosophy is based on three principles: (1) standardization; (2) evolution, and (3) test before use. This process has resulted in a series of fuel designs. Details of these designs are provided in References A–3 or A–4. Fuel bundle design information for bundles more recent than those included in Reference A-3 is found in the compliance report for each fuel product line (See Section 1.4) and in the plant-cycle specific FBIR.

A.4.4 Thermal–Hydraulic Design

A.4.4.1 Design Basis

A.4.4.1.1 Safety Design Bases

See Subsection 4.1.1.

A.4.4.1.2 Requirements for Steady–State Conditions

See Subsection 4.1.2. The design steady–state operating limit MCPR and the peak MAPLHGR are provided by the Applicant in the format shown in Appendix A of the country–specific supplement.

A.4.4.1.3 Requirements for Anticipated Operational Occurrences (AOOs)

See Section 4.1.3.

A.4.4.1.4 Summary of Design Bases

See Section 4.1.4.

A.4.4.2 Description of Thermal–Hydraulic Design of the Reactor Core

A.4.4.2.1 Summary Comparison

A tabulation of thermal and hydraulic parameters of the core is provided by the Applicant in the plant FSAR. Any changes for reload cores will be indicated in the Supplemental Reload Licensing Report.

A.4.4.2.2 Critical Power Ratio

See Subsections 4.2.1 and 4.3.1.

A.4.4.2.3 Average Planar Linear Heat Generation Rate (APLHGR)

See Subsection 4.2.2.

A.4.4.2.4 Void Fraction Distribution

The core average and maximum exit void fractions in the core at rated condition are provided by the Applicant. The axial distribution of core void fractions for the average radial channel and the maximum radial channel (end of node value) for the core are provided by the Applicant. The core average and maximum exit value are also provided by the Applicant. Similar distributions for steam quality are provided by the Applicant. The core average axial power distribution used to produce the above results is provided by the Applicant.

A.4.4.2.5 Core Coolant Flow Distribution and Orificing Pattern

See Subsection 4.2.3.

A.4.4.2.6 Core Pressure Drop and Hydraulic Loads

See Subsection 4.2.4.

A.4.4.2.7 Correlation and Physical Data

See Subsection 4.2.5.

A.4.4.2.8 Thermal Effects of Operational Transients

See Subsection 4.2.6.

A.4.4.2.9 Uncertainties in Estimates

See Subsections 4.2.7 and 4.3.1.1.

A.4.4.2.10 Flux Tilt Considerations

See Subsection 3.2.2.

A.4.4.3 Description of the Thermal and Hydraulic Design of the Reactor Coolant System

A.4.4.3.1 Plant Configuration Data

The Applicant is to provide reactor coolant system geometric data as well as other information required by Reference A–1.

A.4.4.3.2 Operating Restrictions on Pumps

To be provided by Applicant.

A.4.4.3.3 Power–Flow Operating Map

To be provided by Applicant.

A.4.4.3.4 Temperature–Power Operating Map (PWR)

Not applicable.

A.4.4.3.5 Load–Following Characteristics

To be provided by Applicant.

A.4.4.3.6 Thermal and Hydraulic Characteristics Summary Table

The thermal–hydraulic characteristics are to be provided by the Applicant.

A.4.4.4 Evaluation

See Subsection 4.3.

A.4.4.4.1 Critical Power

See Subsection 4.3.1.

A.4.4.4.2 Core Hydraulics

See Subsection 4.2.

A.4.4.4.3 Influence of Power Distributions

See Subsection 4.3.1.

A.4.4.4.4 Core Thermal Response

See Subsection 4.3.4.

A.4.4.4.5 Analytical Methods

See Subsection 4.3.1.2 and the country–specific supplement to this document.

A.4.4.4.6 Thermal–Hydraulic Stability Analysis

See the country–specific supplement to this document.

A.4.4.5 Testing and Verification

The testing and verification techniques to be used to assure that the planned thermal and hydraulic design characteristics of the core have been provided, and will remain within required limits throughout core lifetime, are discussed in Chapter 14 (Initial Test Program) of the plant FSAR.

A.4.4.6 Instrumentation Requirements

The reactor vessel instrumentation monitors the key reactor vessel operating parameters, during planned operations. This ensures sufficient control of the parameters. The reactor vessel sensors are discussed in Subsections 7.6 and 7.7 of the plant FSAR.

A.4.4.6.1 Loose Parts

To be provided by Applicant.

A.5.2.2.2 Design Evaluation

A.5.2.2.2.1 Method of Analysis

The model used to analyze overpressurization is provided in the country–specific supplement to this document.

A.5.2.2.2.2 Transients

The overpressure protection system must accommodate the most severe pressurization event described in the country–specific supplement to this document.

A.5.2.2.2.3 Evaluation of Results

A.5.2.2.2.3.1 Safety/Relief Valve Capacity

The required SRV capacity is determined by analyzing the pressure rise from an MSIV closure with flux scram transient as documented in the country–specific supplement to this document. Results of this analysis are to be provided by the Applicant.

A.6.3.3 ECCS Performance Evaluation

The performance of the ECCS is determined through application of the 10CFR50 Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. Analytical models are described in the country–specific supplement to this document.

The accidents, as listed in Chapter 15 of the plant FSAR, for which ECCS operation is required are:

FSAR	Title

Subsection	
15.2.8	Feedwater Piping Break.
15.6.4	Spectrum of BWR Steam System Piping Failures Outside of Containment.
15.6.5	Loss–of–Coolant Accidents.

Radiological consequences of the above listed events are provided by the Applicant in Chapter 15 of the plant FSAR.

A.6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates (MAPLHGR) calculated in this performance analysis provide the basis for Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections A.6.3.3.4 and A.6.3.3.5, and testing requirements are discussed in Subsection 6.3 of the plant FSAR. Limits on minimum suppression pool water level are to be provided by the Applicant.

A.6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria are extracted from 10CFR50.46. Conformance to each criterion is to be demonstrated by the Applicant. A detailed description of the methods used to show compliance is provided in the country–specific supplement to this document.

A.6.3.3.3 Single–Failure Considerations

To be provided by the Applicant.

A.6.3.3.4 System Performance During the Accident

To be provided by the Applicant.

A.6.3.3.5 Use of Dual Function Components for ECCS

To be provided by the Applicant.

A.6.3.3.6 Limits on ECCS System Parameters

The limits on the ECCS parameters are discussed in Subsections A.6.3.3.1 and A.6.3.3.7.1.

Any number of components in any given system may be out of service, up to and including the entire system. The maximum allowable out–of–service time is a function of the level of redundancy and the specified test intervals.

A.6.3.3.7 ECCS Analyses for LOCA

A.6.3.3.7.1 LOCA Analysis Procedures and Input Variables

The procedures approved for LOCA analysis conformance calculations are described in detail in the country–specific supplement. These procedures were used in the calculations documented in Subsection A.6.3.3.

A.6.3.3.7.2 Accident Description

A detailed description of the LOCA calculation is provided in the country–specific supplement.

A.6.3.3.7.3 Break Spectrum Calculations

To be provided by the Applicant.

A.6.3.3.7.4 Large Recirculation Line Break Calculations

To be provided by the Applicant.

A.6.3.3.7.5 Transition Recirculation Line Break Calculations

To be provided by Applicant.

A.6.3.3.7.6 Small Recirculation Line Break Calculations

To be provided by Applicant.

A.6.3.3.7.7 Calculations for Other Break Locations

To be provided by Applicant.

A.6.3.3.8 LOCA Analysis Conclusions

To be provided by Applicant.

A.9.1.2.3 Safety Evaluation

A.9.1.2.3.1 Criticality Control

To be provided by Applicant.

A spent fuel storage area provided by General Electric will accommodate all fuel types designed by General Electric, as noted in Subsection 3.5 of this document.

A.15.0 Accident Analysis

In this chapter the effects of anticipated process disturbances and postulated component failures are examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and events.

General Electric has developed a unique systematic approach to plant safety consistent with the General Electric Boiling Water Reactor technology base. The key to the General Electric approach to plant safety is the Nuclear Safety Operational Analysis. A generic nuclear safety operational analysis (NSOA) has been developed for each of the recent General Electric Boiling Water Reactor product lines. It has then been modified to be compatible with the specific plant configuration being evaluated. Key inputs into the nuclear safety operational analysis are derived from the applicable regulations and through industry codes and standards.

The nuclear safety operational analysis (NSOA) is provided by the Applicant.

General Electric evaluates the entire spectrum of events in the NSOA in order to establish the most limiting or design basis events in a meaningful manner. It is the design basis events that are quantified in this chapter.

The scope of the situations analyzed includes anticipated (expected) operational occurrences (e.g., loss of electrical load), off–design abnormal (unexpected) transients that induce system operations condition disturbances, postulated accidents of low probability (e.g., the sudden loss of integrity of a major component), and, finally, hypothetical events of extremely low probability (e.g., an anticipated transient without the operation of the entire control rod drive system).

A.15.0.1 Nuclear Safety Operational Analysis

In the nuclear safety operational analyses (NSOA) given in each FSAR, all unacceptable safety results and all required safety actions are identified. In addition, an evaluation of the entire spectrum of events is consistently carried out for all plant designs to demonstrate that a consistent level of safety has been attained.

The NSOA acceptance criteria are based on event probability. This means that events more likely to occur are tested against more restrictive limits. This is consistent with industry practice and the applicable regulatory requirements.

The starting point for the NSOA is the establishment of unacceptable safety results. This concept enables the results of any safety analysis to be compared to applicable criteria. Unacceptable safety results represent an extension of the nuclear design criteria for plant systems and components that are used as the basis for system design. The unacceptable safety results have been selected so that they are consistent with applicable regulations and industry code and standards.

The focal point of the NSOA is the event analysis. In the event analysis, all essential protection sequences are evaluated until all required safety actions are successfully completed. The event analysis identifies all required frontline safety systems and their essential auxiliaries.

The full spectrum of initial conditions limited by the constraints placed on planned operation for AOOs, accidents, and plant capability demonstrations are evaluated. All events are

analyzed until a stable condition is obtained. This assures that the event being evaluated does not have a characteristic for long–term consideration that is important.

In the event analysis all essential system, operator actions, and limits to satisfy the required safety actions are identified. Limits are derived only for those parameters continuously available to the operator. Credit for operator action is taken only when an operator can be reasonably expected to perform the required action based on the information available to him.

In the NSOA a complete and consistent set of safety actions has been developed. These safety actions are those required to prevent unacceptable results. For transients and accidents, a single failure proof path to plant shutdown must be shown. The application of a single failure criterion to these events is imposed as an additional measure of conservatism in the nuclear safety operational analysis process.

A.15.0.2 Event Analytical Objective

The spectrum of postulated initiating events developed from the NSOA is divided into categories based upon the type of disturbance and the expected frequency of the initiating occurrence; the limiting events in each combination of category and frequency are quantitatively analyzed. The plant safety analysis evaluates the ability of the plant to operate without unacceptable safety results within regulatory guidelines.

A.15.0.3 Analytical Categories

Transient and accident events are discussed in individual categories as required by Reference A–1. Documentation of each event appears in References A–7, A–8, A–9 and A–10; however, documentation of the following events is to be provided by the Applicant: Failure of Small Lines Carrying Primary Coolant Outside Containment, Radioactive Gas Waste System Leak or Failure, Liquid Radioactive System Failure, and Postulated Radioactive Release Due to Liquid Radwaste Tank Failure. Each event evaluated is assigned to one of the eight categories listed in Chapter 15 of Regulatory Guide 1.70 (Reference A–1).

A.15.0.4 Event Evaluation

A.15.0.4.1 Identification of Causes and Frequency Classification

Situations and causes which lead to the initiating event analyzed are described within the categories designated above. The frequency of occurrence of each event is summarized based upon the nuclear safety operational analysis and currently available operating plant history for the transient event. Events for which inconclusive data exist are discussed separately within each event section.

Each initiating event within the major groups is assigned to one of three frequency groups defined in Regulatory Guide 1.70 (Reference A–1).

A.15.0.4.2 Identified Unacceptable Results

The unacceptable results for each frequency group are defined for the U.S. plants in Subsection S.2.1 of the GESTAR II U.S. Supplement (Reference A–6). For the foreign

plants, the customer must supply this information if the applicable licensing authority requires it.

A.15.0.4.3 Sequence of Events and Systems Operations

Each transient or accident (except those to be provided by the Applicant as discussed in Subsection A.15.0.3) is discussed and evaluated in terms of:

1. a step–by–step sequence of events from initiation to final stabilized condition;
2. the extent to which normally operating plant instrumentation and controls are assumed to function;
3. the extent to which plant and reactor protection systems are required to function;
4. the credit taken for the functioning of normally operating plant systems; and
5. the operation of engineered safety systems that is required.

This sequence of events is supported by the NSOA for the transient or accident. The effect of a single equipment failure or malfunction or an operator error on the event is shown in the NSOA provided in each FSAR.

A.15.0.4.4 Analysis Basis

The analyses documented in this chapter are for the plant core used for the nuclear evaluations given in Section A.4.3 of this document.

A.15.0.4.4.1 Evaluation Models

The models used to analyze the core and system performance during AOO events are given in Subsection 4.4.1.2 of this document. Models for accident analyses and dose calculations are given in the documentation for the applicable event in the country–specific supplement.

A.15.0.4.4.2 Input Parameters and Initial Conditions for Analyzed Events

In general, the events analyzed have values for input parameters and initial conditions as specified in the format shown in Table A.15.0–1 (to be provided by Applicant). Analyses that assume data inputs different than these values are designated accordingly in the appropriate event discussion.

The dynamic parameters assumed in Chapter 15 of the FSAR are much more conservative than the normal operating values. The scram reactivity presents a conservative lower bound on the minimum scram reactivity and also defines the minimum scram characteristics for permissible operation.

The analytical values for some system characteristics, like SRV delay/stroke time, recirculation pump trip coastdown time constant, etc., bound the design specification for that system.

In conclusion, the values used in FSAR Chapter 15 analyses are conservative values and bound the operating band. Therefore, Chapter 15 analyses will cover all operating conditions and cycle points.

A.15.0.4.4.3 Initial Power/Flow Operating Constants

The analyses basis for most of the transient safety analyses is the thermal power at rated core flow (100%) corresponding to the power designated in the FSAR. This operating point is the apex of a bounded operating power/flow map that, in response to any classified AOOs, will yield the minimum pressure and thermal margins of any operating point within the bounded map.

Any other constraint that may truncate the bounded power/flow map must be observed, such as the recirculation valve and pump cavitation regions, the licensed power limit and other restrictions based on pressure and thermal margin criteria.

The upper operating power/flow limit of a reactor is predicated on the operating basis of the analysis and the corresponding constant rod pattern line. This boundary may be truncated by the licensed power and the MCPR operating limit.

Certain localized events are evaluated at other than the above mentioned conditions. These conditions are discussed pertinent to the appropriate event in References A–7, A–8, A–9 and A–10.

A.15.0.4.5 Evaluation of Results

For each event, the results of standard transient analyses are presented in References A–7, A–8, A–9 and A–10. Results of the transient analyses for individual plants may differ from these results; however, the relative results between events will not change. Therefore, based on these transient results, the limiting events have been identified. Only the results of the limiting events are provided in the format shown in Appendix A of the country–specific supplement to this document. This information should be provided in the FSAR.

The limiting events are listed below and descriptions of the typical analyses performed for these events are given in the country–specific supplement to this document. Reasons why the other events are not limiting are provided in the event documentation in References A–7 through A–10 and supported by the analytical results in these references.

1. Limiting Pressurization Events: Pressure Controller Downscale Failure (BWR/6 only), Generator Load Rejection without Bypass, and Turbine Trip Without Bypass,

2. Limiting Decrease in Core Coolant Temperature event: Loss of Feedwater Heating (manual control), and

3. Limiting Temperature Decrease/Pressurization event: Feedwater Controller Failure (maximum demand).

The Load Rejection and Turbine Trip without Bypass events are categorized as infrequent events but are still included in this list.

A.15.0.4.5.1 Effect of Single Failures and Operator Errors

The effect of a single equipment failure or malfunction, or operator error is provided in the NSOA of each FSAR.

A.15.0.4.5.2 Analysis Uncertainties

Model uncertainties are documented in Subsection 4.3.1.2.6 of this document.

In Table A.15.0–2, a summary of applicable accidents is provided. This table compares the GE calculated amount of failed fuel to that used in worst–case radiological calculations. Most of these results are applicable to all core configurations and can be referenced by the Applicant. Applicability is given in the event documentation.

The Applicant is to provide results for the following events: Failure of Small Lines Carrying Primary Coolant Outside Containment, Radioactive Gas Waste System Leak or Failure, Liquid Radioactive System Failure, and Postulated Radioactive Release Due to Liquid Radwaste Tank Failure.

A.15.0.4.5.3 Barrier Performance

The significant areas of interest for internal pressure damage are the high pressure portions of the reactor coolant pressure boundary (the reactor vessel and the high pressure pipelines attached to the reactor vessel). The overpressure criteria are identified in the country–specific supplement to this document. The limiting overpressurization event analysis is described in the country–specific supplement to this document.

A.15.0.4.5.4 Radiological Consequences

In this chapter, the consequences of radioactivity release during the three types of events: (a) incidents of moderate frequency (anticipated operational occurrences); (b) infrequent incidents (abnormal operational occurrences); and (c) limiting faults (design basis accidents), are considered. For all events whose consequences are limiting, a detailed quantitative evaluation is presented. For nonlimiting events, a qualitative evaluation is presented or results are referenced from a more limiting or enveloping case or event.

The Applicant is to provide results for the following events: Failure of Small Lines Carrying Primary Coolant Outside Containment, Radioactive Gas Waste System Leak or Failure, Liquid Radioactive System Failure, and Postulated Radioactive Release Due to Liquid Radwaste Tank Failure.

For limiting faults (design basis accidents), two quantitative analyses are considered:

The first is based on conservative assumptions considered to be acceptable to the NRC for the purposes of worst–case bounding the event and determining the adequacy of the plant design to meet 10CFR100 guidelines. This analysis is referred to as the "design basis analysis."

The second is based on realistic assumptions considered to reflect expected radiological consequences. This analysis is referred to as the "realistic analysis."

Results for both are shown to be within NRC guidelines.

A.15.1 through A.15.8

Event descriptions are provided in Sections 15.1 through 15.8 of References A–7, A–8, A–9 and A–10. Results of limiting events are given in Section 15.0 of the FSAR.

Description and results of the following events are to be provided by the Applicant: Failure of Small Lines Carrying Primary Coolant Outside Containment, Radioactive Gas Waste System Leak or Failure, Liquid Radioactive System Failure, and Postulated Radioactive Release Due to Liquid Radwaste Tank Failure.

Table A.15.0–1

Input Parameters and Initial Conditions for Anticipated Operational Occurrences (To be provided by Applicant)

Parameters and Conditions
1. Thermal Power Level (MWt) License Value Analysis Value
2. Steam Flow (lb/hr) License Value Analysis Value
3. Core Flow (lb/hr)
4. Feedwater Flow Rate (lb/sec) License Value Analysis Value
5. Feedwater Temperature (°F)
6. Vessel Dome Pressure (psig)
7. Vessel Core Pressure (psig)

Parameters and Conditions
8. Turbine Bypass Capacity (% NBR)
9. Core Coolant Inlet Enthalpy (Btu/lb)
10. Turbine Inlet Pressure (psig)
11. Fuel Lattice
12. Core Leakage Flow (%)
13. Required MCPR Operating Limit First Core Reload Core
14. MCPR Safety Limit First Core Reload Core
15. Doppler Coefficient (–¢/°F T_{avg}) Analysis Data (REDY only)
16. Void Coefficient (–¢/% rated voids) Analysis Data for Power Increase Events (REDY only)[a]
17. Analysis Data for Power Decrease Events (REDY only)[a] Core Average Rated Void Fraction (%) (REDY only)[a]
18. Scram Reactivity, $\Delta K Analysis Data (REDY only)[a]
19. Control Rod Drive Position versus time
20. Nuclear characteristics used in ODYN simulations
21. Jet Pump Ratio (M)
22. Safety/Relief Valve Capacity (% NBR) at 1210 psig Manufacturer Quantity Installed
23. Relief Function Delay (sec)
24. Relief Function Response Time Constant (sec)
25. Safety Function Delay (sec)
26. Safety Function Response Time Constant (sec)
27. Setpoints for Safety/Relief Valves Safety Function (psig)

[a] For transients simulated on the ODYN computer model, this input is calculated by ODYN.

Parameters and Conditions
28. Relief Function (psig) Number of Valve Groupings Simulated Safety Function (No.) Relief Function (No.)
29. S/R Valve Reclosure Setpoint – Both Modes (% of setpoint) – Maximum Safety Limit (used in analysis – Minimum Operational Limit
30. High Flux Trip (% NBR) Analysis setpoint
31. High Pressure Scram Setpoint (psig)
32. Vessel Level Trips (ft above bottom of separate skirt bottom) Level 8 – (L8) (ft) Level 4 – (L4) (ft) Level 3 – (L3) (ft) Level 2 – (L2) (ft)
33. APRM Simulated Thermal Power Trip Scram (% NBR) Analysis Setpoint Time Constant (sec)
34. Recirculation Pump Trip Delay (sec)
35. Recirculation Pump Trip Inertia Time Constant for Analysis (sec)[b]
36. Total Steamline Volume (ft^3)
37. Set pressure of Recirculation pump trip (psig) (Nominal)

Table A.15.0–2
Summary of Accidents

Title	Failed Fuel Rods

[b] The inertia time constant is defined by the expression:

$$t = \frac{2\pi J_0 n}{g T_0}$$

where:

t = Inertia time constant (sec);
J_0 = pump motor inertia (lb–ft);
n = rated pump speed (rps);
g = gravitational constant (ft/sec^2); and
T_0 = pump shaft torque (lb–ft).

	GE Calculated Value	NRC Worst–Case Assumption
Seizure of one recirculation pump.	None.	
Recirculation pump shaft break.	None.	
Rod drop accident.	[c]	[c]
Instrument line break.	None.	None.
Steam system pipe break outside containment.	None.	None.
LOCA within RCPB.	None.	100%.
Feedwater line break.	None.	None.
Main condenser gas treatment system failure.	N/A.	N/A.
Liquid radwaste tank failure.	N/A.	N/A.
Fuel–handling accident.	[c]	[c]
Cask drop accident.	None.	None.
ATWS.	Fuel product line dependent. See Section 1.	

A.16 References

A–1 United States Nuclear Regulatory Commission, Regulatory Guide 1.70, Revision 3, *Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)*, November 1978.

A–2 *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, LWR Edition, United States Nuclear Regulatory Commission, September 1975 (NUREG–75/087).

A–3 *Global Nuclear Fuels Fuel Bundle Designs*, NEDE–31152P, Revision 9, May 2007, including Supplement 1, June 2000, through Supplement 6, May 2007.

A–4 *General Electric Fuel Bundle Designs Evaluated with TEXICO/CLAM Analysis Bases*, NEDE–31151–P, April 1986.

A–5 K. W. Brayman and K. W. Cook, *Evaluation of Control Blade Lifetime with Potential Loss of B_4C*, NEDO–24226, December 1979; and Supplement 1 (Proprietary), March 1981.

A–6 *General Electric Standard Application for Reactor Fuel Supplement for the United States*, General Electric Company Licensing Topical Report, NEDE–24011–P–A–US, (Latest Approved Revision).

[c] To be supplied by applicant.

A–7 *General Electric Standard Safety Analysis Report*, 22A7007.

A–8 *Final Safety Analysis Report for LaSalle County Station*, Dockets 50–373 and 50–374, Commonwealth Edison Co.

A–9 *Final Safety Analysis Report for Susquehanna Steam Electric Station*, Dockets 50–387 and 50–388, Pennsylvania Power and Light Co.

A–10 *Final Safety Analysis Report for Grand Gulf Nuclear Station, Units 1 and 2*, Dockets 50–416 and 50–417, Mississippi Power and Light Co.