February 22, 2017

Mr. Jerald G. Head
Senior Vice President, Regulatory Affairs
General Electric-Hitachi
Nuclear Energy Americas, LLC
P.O. Box 780, M/C A-18
Wilmington, NC 28401-0780

SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY - AMERICAS, LLC TOPICAL REPORT NEDE-33005P AND NEDO-33005, REVISION 0, "LICENSING TOPICAL REPORT TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEMS / LOSS-OF-COOLANT-ACCIDENT ANALYSES FOR BWR/2-6" (CAC NO. ME5405)

Dear Mr. Head:

By letter dated January 27, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110280323), GE Hitachi Nuclear Energy Americas LLC (GE-H) submitted Topical Report (TR) NEDE-33005P/NEDO-33005, Revision 0, "Licensing Topical Report TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6," to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

By letter dated January 18, 2017, an NRC draft safety evaluation (SE) regarding our approval of TR NEDE-33005P, Revision 0, "Licensing Topical Report TRACG Application for Emergency Core Cooling Systems / Loss-of-Coolant-Accident Analyses for BWR/2-6," was provided for your review and comment (ADAMS Accession No. ML17012A287). By letter dated January 27, 2017, you provided comments on the draft SE (ADAMS Accession No. ML17027A094). The NRC staff's disposition of the GE-H comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR NEDE-33005P, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The enclosed final SE is a publicly available version with GE-H proprietary material redacted. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in licensing applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that GE-H publish approved proprietary and non-proprietary versions of TR NEDE-33005P, Revision 0, within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved versions shall include a "-A" (designating approved) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, GE-H will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

/RA/

Kevin Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 710

Enclosure:
Final SE (Non-Proprietary)

SUBJECT: FINAL SAFETY EVALUATION FOR GE HITACHI NUCLEAR ENERGY - AMERICAS, LLC TOPICAL REPORT NEDE-33005P AND NEDO-33005, REVISION 0, "LICENSING TOPICAL REPORT TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEMS / LOSS-OF-COOLANT-ACCIDENT ANALYSES FOR BWR/2-6" (CAC NO. ME5405) DATED: February 22, 2017

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

GENERAL ELECTRIC-HITACHI LICENSING TOPICAL REPORT NEDE-33005P AND

NEDO-33005, "TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEMS/

LOSS-OF-COOLANT ACCIDENT ANALYSES FOR BWR/2-6"

EXECUTIVE SUMMARY

Licensing Topical Report (LTR) NEDE-33005P, Revision 0, "TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6" was submitted to the U.S. Nuclear Regulatory (NRC) by General Electric-Hitachi Nuclear Energy (GEH, or the vendor) by letter dated January 27, 2011, for NRC staff review and approval. The LTR describes an emergency core cooling system (ECCS) evaluation model (EM) that GEH developed to analyze boiling water reactor (BWR) loss-of-coolant accidents (LOCAs). The EM is referred to as TRACG-LOCA. The NRC staff review and basis for approving TRACG-LOCA is provided in this safety evaluation (SE). This SE is also intended to provide guidance to NRC staff for reviewing plant-specific requests for licensing action that are based on, or supported by, TRACG-LOCA.

This SE is organized into eleven successive chapters. The first two chapters provide background and regulatory basis for the NRC staff review. Chapter 3 discusses the accident scenario specification and delineates the applicability of the TRACG-LOCA approval. Chapter 4 presents a review of the TRACG code and its execution.

Chapters 5 through 8 focus on the application of TRACG to ECCS evaluation; while Chapter 5 is focused on the analytic methodology, Chapter 6 addresses the LOCA-relevant key models that were reviewed by the NRC staff. Chapter 7 provides a review of the overall combination of uncertainty associated with TRACG-LOCA. Although Chapter 6 reviews the key models in consideration of their respective qualification, Chapter 8 additionally provides a review of the analysis of qualification events (i.e., large-scale LOCA tests) and a demonstration analysis for the BWR/2 event, which is expected to be the most challenging application of the EM within its plant design applicability.

Finally, Chapters 9 through 11 discuss licensing aspects of the NRC staff review. Chapter 9 discusses licensing considerations of TRACG-LOCA, including its relationship to other GEH LTRs and expectations regarding plant-specific submittals based on TRACG-LOCA. Chapter 10 provides the limitations associated with the TRACG-LOCA approval, and Chapter 11 presents the overall review conclusion.

Based on its detailed review, the NRC staff determined that NEDE-33005P is acceptable for referencing in licensing actions. For the purpose of compliance with Title 10 of the *Code of Federal Regulations* 50.46 requirements, TRACG-LOCA, as documented in NEDE-33005P, may be considered an acceptable evaluation model. With regard to referencing in licensing actions, NEDE-33005P may be considered approved for use.

Contents

LIST OF ACRONYMS

ABB	Allmänna Svenska Elektriska Aktiebolaget Brown Boveri
ADAMS	Agencywide Documents Access and Management System
AEC	Atomic Energy Commission
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASEA	Allmänna Svenska Elektriska Aktiebolaget
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
BEPU	Best-Estimate Plus Uncertainty
B-J	Baker-Just
BWR	Boiling Water Reactor
CHF	Critical Heat Flux
CCFL	Counter-Current Flow Limitation
CLTR	Constant Pressure Power Uprate Licensing Topical Report
C-P	Cathcart-Pawel
CPPU	Constant Pressure Power Update
CSAU	Code Scaling, Applicability, and Uncertainty
CSHT	Core Spray Heat Transfer
CWO	Core-Wide Oxidation
DEGB	Double-Ended Guillotine Break
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
ECR	Equivalent Cladding Reacted
ELTR	Extended Power Uprate Licensing Topical Report
EM	Evaluation Model
EPU	Extended Power Uprate
EOOS	Equipment Out-of-Service
ESBWR	Economic, Simplified Boiling Water Reactor
FIST	Full Integral Scale Test
FIX-II	*Series of integral effects tests simulating a Swedish-designed BWR*
FSAR	Final Safety Analysis Report
GE	General Electric
GEH	GE Hitachi Nuclear Energy Americas
GESTAR-II	General Electric Standard Application for Reactor Fuel
GEXL	General Electric Critical Quality Boiling Length
GIRAFFE	*Gravity-driven integral full-height test for passive heat removal*
GNF	Global Nuclear Fuel
ICF	Increased Core Flow
IET	Integral Effect Test
IMLTR	Interim Methods Licensing Topical Report
LCO	Limiting Condition for Operation
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LTR	Licensing Topical Report
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MCPR	Minimum Critical Power Ratio
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MLO	Maximum Local Oxidation

M+SAR	Maximum Extended Load Line Limit Analysis Plus Safety Analysis Report
NRR	Office of Nuclear Reactor Regulation
OLMCPR	Operating Limit Minimum Critical Power Ratio
OLTP	Original Licensed Thermal Power
OPA	Offsite Power Available
ORNL	Oak Ridge National Laboratory
PCT	Peak Cladding Temperature
PIRT	Phenomena Identification and Ranking Table
PLHGR	Peak Linear Heat Generation Rate
PSTF	Pressure Suppression Test Facility
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RIL	Research Information Letter
RLA	Request for Licensing Action
RG	Regulatory Guide
ROSA	Rig of Safety Assessment
SBWR	Simplified Boiling Water Reactor
SE	Safety Evaluation
SGT	Sun-Gonzalez-Tien
SIT	Systems Interaction Test
SRP	Standard Review Plan
SSTF	Steam Sector Test Facility
THTF	Thermal Hydraulic Test Facility
TLTA	Two-Loop Test Apparatus
TMOL	Thermal-Mechanical Operating Limit
TR	Topical Report
TRAC	Transient Reactor Analysis Code
TRACG	GEH-proprietary version of the Transient Reactor Analysis Code
TS	Technical Specification
UFM	Ultrasonic Flow Meter

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

GENERAL ELECTRIC-HITACHI LICENSING TOPICAL REPORT NEDE-33005P, AND

NEDO-33005, "TRACG APPLICATION FOR EMERGENCY CORE COOLING SYSTEMS/

LOSS-OF-COOLANT ACCIDENT ANALYSES FOR BWR/2-6"

1.0 INTRODUCTION

By letter dated January 27, 2011, General Electric (GE)-Hitachi Nuclear Energy (GEH, or the vendor) submitted for US Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) NEDE-33005P, Revision 0, and NEDO-33005, Revision 0, "TRACG Application for Emergency Core Cooling Systems/Loss-of-Coolant-Accident Analyses for BWR/2-6" (Reference 1). The TR describes an emergency core cooling system (ECCS) evaluation model (EM) that GEH developed to analyze boiling water reactor (BWR) loss of coolant accidents (LOCAs). For the sake of brevity, this safety evaluation (SE) will hereafter refer to the TR as NEDE-33005P, and to the EM itself as TRACG-LOCA.

1.1. CORRESPONDENCE SUMMARY

During the NRC staff review, GEH supplemented the TR with six submittals. The vendor provided supporting information related to TRACG and its post-processing tools by letter March 30, 2012 (Reference 2). The vendor supplemented the TR with responses to NRC staff requests for additional information (RAIs) in five batches, as shown in Table 1, below (References 3 - 7). Table 1 also provides the GEH letter number and the Agencywide Documents Access and Management System (ADAMS) Accession Number for each RAI response transmittal, for convenient reference.

Batch	Date	RAIs	Letter No.	Accession No.
1	October 7, 2014	1 - 66	MFN 14-064	ML14281A014
2	February 19, 2016	67 - 98	MFN 16-008	ML16050A138
3	June 13, 2016	33*, 99, 100	MFN 16-020	ML16165A348
4	June 21, 2016	101, 102	MFN 16-039	ML16173A330
5	October 21, 2016	33*, 65*, 103, 104	MFN 16-072	ML16295A253

*The responses to RAIs 33 and 65 were revised.

Table 1. Index of RAI Responses.

1.2. REVIEW APPROACH

As discussed in Chapter 2 of this SE, there are numerous sources of regulatory guidance for performing the review of an ECCS evaluation model. Some of these sources were published concurrent with efforts to revise the ECCS performance requirements to permit realistic ECCS evaluation in 1988, and others were developed more recently. The 1988 vintage guidance pertains most specifically to ECCS evaluation, and despite that the state of the art has evolved significantly since the publication of this guidance, it remains most applicable to the review of an ECCS EM. However, more recent guidance, which is more generalized to the review of accident and transient analysis methods, provides a reasonable framework to perform the review, including the concept of a graded approach to reviewing the evaluation model. Unfortunately, using a single element of the available guidance leads to an incomplete review, while using all of the guidance would lead to significant amounts of repetition of effort. Therefore, the NRC staff review was accomplished by hybridizing the available guidance to construct an evaluation model review that balances the more general aspects of the EM review with recognition of the fact that TRACG is a mature code that has been accepted by the NRC staff in previous applications. This SE focuses on the detailed code features that are somewhat unique and important to ECCS evaluation among other applications.

2.0 REGULATORY BASIS

The TRACG-LOCA EM was developed in accordance with the regulatory requirements established in Title 10, "Energy," of the *U.S. Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities," Section 46, "Acceptance criteria for ECCSs for light-water nuclear power reactors" (10 CFR 50.46). In developing TRACG-LOCA, GEH considered guidance contained in two NRC Regulatory Guides (RGs). These include: (1) RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," and (2) RG 1.203, "Transient and Accident Analysis Methods" (References 8 and 9).

The NRC staff reviewed NEDE-33005P to determine whether TRACG-LOCA is an acceptable evaluation model as set forth in 10 CFR 50.46. In its review, the NRC staff relied on the regulatory guidance described above, as well as applicable chapters contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Light Water Reactor Edition." Principally, the NRC staff relied on Standard Review Plan (SRP), Chapter 15.0.2, "Review of Transient and Accident Analysis Methods" (Reference 10).

As described in Section 2.1, below, the TRACG-LOCA EM is required to provide an estimated uncertainty associated with its results, and comparisons must be made to experimental data to show that its results realistically describe reactor behavior under hypothetical LOCA conditions. The NRC provides an acceptable approach to determining uncertainty associated with safety analysis methods in NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break, Loss-of-Coolant Accident" (Reference 11). In addition, the NRC provides a compendium of experimental data pertinent to ECCS evaluation model in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis" (Reference 12). These documents provide additional, supporting guidance for the documentation contained in NEDE-33005P and the related NRC review.

2.1. APPLICABLE REGULATORY REQUIREMENTS

Holders of operating licenses under 10 CFR Part 50 are required, pursuant to subsection 50.34, to submit final safety analysis reports (FSARs) to the NRC. In part, 50.34(b)(4) states:

> …Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46…

GEH developed TRACG-LOCA as a realistic, best-estimate, or best-estimate plus uncertainty (BEPU) evaluation model, [A] to be used to evaluate ECCS performance at BWRs. The enabling regulatory framework is established in Paragraph 50.46(a)(1)(i) which states, in part:

> Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated… [T]he evaluation model must include sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a loss-of-coolant accident. Comparisons to applicable experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated. This uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded. Appendix K, Part II Required Documentation, sets forth the documentation requirements for each evaluation model…

Paragraph (b) of 50.46, as referenced in the above excerpt, provides the acceptance criteria for ECCS evaluation. The acceptance criteria are as follows:

> (b)(1) *Peak cladding temperature.* The calculated maximum fuel element cladding temperature shall not exceed 2200°F.[B]
>
> (2) *Maximum cladding oxidation.* The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation…
>
> (3) *Maximum hydrogen generation*. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
> (4) *Coolable Geometry*. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

[A] In addition to the term, "realistic," the terms, "best-estimate," and, "BEPU," are also frequently used in a similar context. It should be understood that, in order for the analytic results to be considered acceptable for the purposes of demonstrating compliance with § 50.46 requirements, the realistic, or so-called best-estimate, results must be expressed at some upper level that includes an allowance for estimated uncertainty. The distinction among these terms is discussed in further detail in Enclosure C, "ACRS Comments on Code Scaling, Applicability and Uncertainty Associated with the use of Realistic ECCS Evaluation Models," to SECY-88-162, "Revision of the ECCS Rule Contained in Appendix K and Section 50.46 of 10 CFR Part 50."
[B] Note that Reference 1 documents results and compares peak cladding temperatures using Kelvin, as opposed to degrees Fahrenheit. Expressed in Kelvin, 2200 °F is approximately 1478 K.

> (5) *Long-Term Cooling.* After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The evaluation model addresses Criteria (b)(1) through (b)(3) by providing analytic results to compare directly to the acceptance criteria. The evaluation model provides the peak cladding temperature (PCT), the maximum local oxidation (MLO), which is also known as equivalent cladding reacted (ECR), and core-wide oxidation (CWO). Additional discussion pertaining to Criteria (b)(4) and (b)(5) are located in Section 3.2, "Scenario Specification," of this SE.

Additional requirements, which govern assumptions that must be employed in the ECCS evaluation, are contained in 10 CFR Part 50 Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 35, "Emergency core cooling," which states:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

These requirements – § 50.34, § 50.46, and GDC 35 – form the regulatory basis for the NRC staff review.

2.2. STANDARD REVIEW PLAN GUIDANCE

As discussed in Section 2.0, the NRC staff performed its review using the SRP.

The NRC staff relied principally on the over-arching guidance in SRP Chapter 15.0.2. This SRP chapter documents the areas of an evaluation model to be reviewed, including: (1) Documentation; (2) Evaluation Model; (3) Accident Scenario Identification Process; (4) Code Assessment; (5) Uncertainty Analysis; and (6) Quality Assurance Plan. The SRP chapter provides guidance intended to focus review resources, primarily, on ensuring (1) that the scenario to be analyzed is appropriately specified, (2) that previously un-reviewed aspects of the evaluation model that are pertinent to the scenario under consideration are appropriately reviewed, and (3) that the application of the model evaluates uncertainties appropriately. Accomplishing these review objectives provides assurance that the results obtained from TRACG-LOCA, with uncertainty accounted for, demonstrate with high probability that the criteria of § 50.46(b)(1-3) are not exceeded, consistent with § 50.46(a)(1)(i) requirements.

Chapter 15.0.2 of the SRP provides guidance to the NRC staff in performing the safety review of NEDE-33005P. It describes methods or approaches that the NRC staff has found acceptable for meeting NRC requirements. For the purposes of reviewing an ECCS EM, however, the SRP is not considered a complete, standalone reference to provide all the required review guidance. Additional documents, discussed in Sections 2.3 and 2.4 of this SE, also apply.

2.3. NRC REGULATORY GUIDES

Regulatory Guides provide guidance to licensees, applicants, and vendors on implementing specific parts of the NRC's regulations, techniques used by the NRC staff in evaluating specific problems or postulated accidents, and data needed by the staff in its review of applications for permits or licenses. The particular guides applicable to ECCS evaluation model development include RG 1.203 and RG 1.157.

Regulatory Guide 1.203 is the analog to SRP 15.0.2. It describes a process that the NRC staff considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant.

RG 1.157 was developed in concert with the revision to § 50.46 that permitted the use of realistic ECCS EMs. This RG describes models, correlations, data, model evaluation procedures, and methods that are acceptable to the NRC staff for meeting the requirements for a realistic or best-estimate calculation of ECCS performance during a LOCA and for estimating the uncertainty of that calculation.

2.4. ADDITIONAL LITERATURE

2.4.1. Code Scaling, Applicability, and Uncertainty

The NRC staff review was based, in part, on NUREG/CR-5249, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation to a Large-Break, Loss-of-Coolant Accident" (Reference 11). This NUREG/CR report describes an uncertainty evaluation methodology called code scaling, applicability, and uncertainty (CSAU). The NRC staff considered the information contained in NUREG/CR-5249 in its review.

2.4.2. Compendium of ECCS Research

The requirements contained in § 50.46(a)(1)(i) state, in part, that "comparisons to applicable experimental data must be made…" Accordingly, the guidance in RG 1.203 and RG 1.157 frequently indicate that models, correlations, formulas, etc., will be considered acceptable, provided they are checked against or compared to relevant data sets. The NRC provides a set of such relevant data sets in NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis" (Reference 12).

2.4.3. Additional Sources

The NRC-published regulatory guidance and technical reference materials specifically pertinent to ECCS EMs were all published prior to 1990. The state of the art for LOCA research and EM development has evolved since then. Hence, additional sources of relevant research, which are not necessarily included in the literature reviewed above, were considered as appropriate.

3.0 ACCIDENT SCENARIO IDENTIFICATION

3.1. NUCLEAR POWER PLANT SELECTION

The EM is applicable to GEH-designed nuclear steam supply systems of the BWR/2 through BWR/6 product lines. This applicability is identified in Chapter 2 of Reference 1, and the qualification and demonstration analyses focus on BWR/2 through BWR/6 analyses. Based on these two considerations, the NRC staff determined that GEH is consistent with this element of the accident scenario identification process as set forth in SRP 15.0.2.

The applicability is considered a limitation to the NRC staff approval of TRACG-LOCA. The applicability of the EM to other BWR designs, such as ASEA Brown Boveri (ABB)-designed BWRs, was not considered. The applicability to evolutionary designs, such as the Economic Simplified Boiling Water Reactor (ESBWR), is addressed in separate applications. The model is not considered applicable to pressurized water reactor designs. Refer to Limitation 1.1, “Nuclear Power Plant Specification,” in Chapter 10 of this SE for additional discussion.

3.1.1. Fuel Design-Specific Considerations

In addition to the BWR plant category designation, it is important also to recognize that TRACG-LOCA was reviewed considering the GE14 and GNF2 fuel product lines. Although minor upgrades to these fuel designs can probably be readily accommodated by the evaluation model, the introduction of new fuel design features that require substantial revision to the evaluation model or methodology would require additional review. This is Limitation 1.2, “Fuel System Design Applicability,” in Chapter 10 of this SE.

It should be noted also that GEH has established means to account for the characteristics of other fuel designs. These include limiting power shapes, cladding material properties, and critical quality-boiling length correlations. Limitation 1.3, “Competitor and Co-Resident Fuel System Applicability,” permits modeling of competitor or co-resident fuel to the extent that TRACG-LOCA can accommodate the design features of such fuel, but requires that operating constraints on such fuel remain supported by, or more conservative than, the analytic methods furnished by the vendor(s) of that fuel.

In this sense, operating constraints include applicable fuel design operating limits, but also fuel and core parameters that are initial conditions to the ECCS evaluation. As this limitation applies to operating characteristics of what would be legacy fuel, the practical implication is that the legacy fuel must be operated within the constraints specified in the legacy ECCS evaluation. If, as an initial condition in the legacy evaluation, a MAPLHGR of 12.5 kW/ft is assumed, then the fuel is expected to be operated at or below this value, regardless of whether the TRACG-LOCA analysis supports a higher value. This limitation is necessary because the TRACG-LOCA application, and approval basis, considers thermal-hydraulic modeling features that are confirmed to be applicable to GNF fuel designs. Other ECCS evaluation models reflect similar, proprietary characteristics of specific fuel designs.

Limitation 1.3 specifies that, if TRACG-LOCA is used to establish less limiting operating characteristics for legacy fuel than those established within the design and licensing basis for that fuel, then the implementing licensee would be required to submit a request for licensing action (RLA)[C] for prior NRC staff review and approval. This is because the NRC staff review did not consider any demonstration analyses, nor evaluate specific modeling techniques or experimental basis, that justify application of TRACG-LOCA analytic results to other vendor fuel designs.

3.2. SCENARIO SPECIFICATION

In Section 2.5.2 of Reference 1, GEH states:

> The LOCA scenarios include the full range of pipe breaks for the distinct BWR product lines (BWR/2; BWR/3,4; and BWR/5,6). The scenarios are differentiated by break size and location. Typical BWR LOCA scenarios are described in Section 3.2 [of NEDE-33005]. The LOCA transient is divided into Blowdown and Refill/Reflood phases so that the application methodology can be focused on the processes and components that are important to each phase.

The scenarios described above are evaluated with respect to the critical safety parameters, PCT, ECR, and CWO. These critical safety parameters tie directly to the § 50.46(b)(1) through (b)(3) acceptance criteria. GEH also states, in Section 2.5 of Reference 1, that by satisfying the PCT and ECR acceptance criteria, the TRACG-LOCA results also demonstrate that Criterion (b)(4), related to coolable geometry, is satisfied.

This assertion is acceptable relative to TRACG-LOCA, because separate GEH fuel assembly design criteria require fuel assembly structural integrity and control blade insertability under seismic and LOCA conditions. Specifically, the thermal-mechanical design criteria provided in Section 1.1.2.B of the General Electric Standard Application for Reactor Fuel (GESTAR II) establish these requirements (Reference 13). When the PCT and ECR remain below regulatory limits, the fuel cladding remains sufficiently ductile that the thermal-mechanical design criteria assure that the fuel will remain in a coolable configuration. Thus, the NRC staff accepts GEH's disposition for Criterion (b)(4) of § 50.46.

GEH states further that existing analyses demonstrate that Criterion (b)(5), regarding long-term core cooling, is satisfied, and that the criterion need not be evaluated as part of the TRACG ECCS/LOCA analysis. This position is reflected in Limitation 1.5, "Regulatory Compliance," of the TRACG-LOCA approval. Refer to Chapter 10 of this SE for additional detail.

Because the vendor has identified the scenario under consideration – a BWR hypothetical LOCA – the NRC staff concluded that GEH has addressed this element of the accident scenario identification process as set forth in SRP 15.0.2. Based on these considerations, the NRC staff determined that TRACG-LOCA is acceptable with respect to scenario specification.

[C] The term *requested licensing action* is adopted within Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-109, "Acceptance Review Procedures," to include, more broadly than license amendment requests, activities that require NRC approval prior to implementation. Such activities include license amendments, relief requests, exemptions, security and emergency plan changes, etc. Refer to ADAMS Accession ML091810088 for additional discussion.

3.3. PHENOMENA IDENTIFICATION AND RANKING

The ranking and identification of phenomena relevant to a specified accident scenario is a structured process described in Reference 11. As described in SRP 15.0.2, it is an acceptable means to approach the accident scenario identification process (Reference 10).

GEH presents its phenomena identification and ranking table (PIRT) in Chapter 3 of Reference 1. As discussed above, the PIRT identifies critical safety parameters, which tie directly to the § 50.46(b)(1) through (b)(3) acceptance criteria. The PIRT also identifies downcomer level and core water level as "intermediate safety parameters," because, according to GEH, "These safety parameters are the criteria used to judge the performance of the safety systems and the margins in the design" (Reference 1, Page 3-1).

The NRC staff review was based on the regulatory guidance contained in References 10 and 11. The RAIs listed in Table 2, below, are relevant to the NRC staff review.[D]

Batch	RAI	Topic	Ref.
1	14	Treatment of Medium-Ranked PIRT Parameters	3
1	22	Treatment of Jet Pump Critical Flow	3
1	37	Statement in PIRT Table Entry F3	3
1	38	Air/Two-Phase CCFL in BWR/2	3
1	58	Emergency Core Cooling in Upper Plenum	3

Table 2. RAI Responses Related to Phenomena Identification and Ranking.

GEH initiates Chapter 3 of the Licensing Topical Report (LTR) by presenting several demonstration analyses for the variety of reactor designs and for various break locations and sizes. Building on these analyses, the PIRT offers the GEH interpretation of the overall significance of the various phenomena on the critical safety parameters for both small breaks and large breaks for the three main product groupings, i.e., BWR/2, BWR/3-4, and BWR/5-6. GEH assigns importance values of high, medium, and low. Section 3.4 indicates that those phenomena assigned high importance have a significant effect on the primary safety parameters, while medium-ranked phenomena have a small effect on the primary safety parameters. Although Revision 0 of the LTR indicated that medium-ranked parameters "may be excluded in the overall uncertainty evaluation" (Reference 1, Page 3-16), GEH clarified, in response to RAI 14.a, that all medium-ranked PIRT parameters would be included in the statistical analysis, along with the high-ranked phenomena (Reference 3). Further, the response to RAI 14.b indicated analyses would generically be performed in accordance with the highest rank assigned to a particular phenomenon, regardless of the specific plant class. Low-importance phenomena have insignificant or no effect on the primary safety parameters and "need not be considered in the overall uncertainty evaluation," states GEH (Reference 1, Page 3-16).

3.3.1. PIRT Process – NRC Staff Evaluation

As per Chapter 15.0.2 of the SRP, Section III, "Review Procedures," Sub-section C, "Accident Scenario Identification Process," the NRC staff review of the PIRT process confirms that the

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[D] These RAIs address other issues besides the PIRT importance of one particular phenomenon or another. This section of the SE, however, considers only those portions of the RAI responses that address the PIRT importance of phenomena. Additional aspects of the RAI response are evaluated, as appropriate, in other sections of the SE.

dominant physical phenomena influencing the outcome of the accident are correctly identified and ranked. A structured process for this effort is described in CSAU. The staff review included an evaluation of the PIRT process relative to that described in CSAU, comparison to other relevant PIRTs (References 14 and 15), and additional input from the vendor through the RAI process.

In its review, the NRC staff determined that the GEH PIRT was generally consistent with the analogous PIRTs. The RAIs were generated to address discrepancies between the analogous PIRTs, and to address demonstration analysis results that suggested higher importance was attributable to specific phenomena than GEH assigned. Specifically, the vendor responses to RAIs 22, 37, 38, and 58 were considered in the PIRT review (Reference 3).

The NRC staff review approach was simplified, somewhat, by GEH's treatment of medium-ranked PIRT parameters. Since the medium-ranked phenomena are included in the statistical analysis, they are treated effectively the same as the high-ranked phenomena. Thus, the distinction between medium- and high-ranked phenomena is insignificant. For example, the issue discussed in RAI 22, related to the medium ranking assigned to a phenomenon in the jet pumps, was resolved in part because the particular phenomenon is treated the same as a highly ranked phenomenon.

The issue addressed in RAI 22 is the PIRT treatment of jet pump reverse flow. While GEH characterizes this phenomenon as having medium importance, two other sources rank the phenomenon highly. In particular, Reference 15 observed that jet pump reverse flow was important during blowdown, because it helps to determine the total core flow. In its response, GEH acknowledged that the jet pump reverse flow phenomenon may have a higher importance than many other medium-ranked parameters, but that the use of a three-tiered (i.e., high, medium, and low) ranking process does not allow for distinguishing such importance. GEH also stated that the PIRT process is somewhat subjective, and added that the assignment of a medium rank to the jet pump reverse flow importance ensures that it is treated consistently with high-rank parameters, in that the calculations include the biases and uncertainties associated with the phenomenon. Since the calculations include the phenomenon along with its biases and uncertainties, the NRC staff determined that the distinction between medium and high importance for the jet pump reverse flow is insignificant for the purposes of LOCA analysis, and thus accepted the RAI response provided by the vendor.

In RAI 37, the NRC staff noted that the PIRT entry for Item F3, "Noncondensible Return at Low Pressures," included a statement that the PCT transient is over before the vessel is depressurized to containment pressure. However, the statement was inconsistent with demonstration calculations in Chapter 3 of Reference 1, as well as NRC staff audit calculations. In response to the RAI, GEH agreed to remove the statement. The statement had no effect on the PIRT treatment, as the highest ranking for the phenomenon was high, specifically for the BWR/2 event. Since the statement in the table had no effect on GEH's treatment of the parameter, the NRC staff accepted the vendor's deletion of the statement.

The response to RAI 38 discusses PIRT Entry M3, counter-current flow limitation (CCFL): Air In/Two-Phase Flow Out (Reference 3). In the RAI, the NRC staff questioned the assignment of medium importance to the phenomenon, even in consideration of the fact that the BWR/2 demonstration calculations appear to indicate that the phenomenon has a more direct impact on the PCT, suggesting a higher ranking would be more appropriate. The response notes that **[[**

]] Thus, the vendor stated, uncertainty regarding the total amount of ingested air has a lower effect on PCT than the simpler consideration of whether or not air is ingested into the reactor coolant system. Sensitivity studies performed in response to RAI 13.b corroborate this assertion: [[

]] Moreover, the phenomenon is assigned a medium value, meaning its uncertainty is addressed in the analysis.[E] The response to RAI 13 is discussed in greater detail in Section 8.2 of this SE.

The response to RAI 58 addresses the importance of PIRT Item F2, which is ECC Interaction/Mixing/Subcooling in the Upper Plenum. The phenomenon has a high ranking, but GEH states, in Section 5.1.6.2 of Reference 1, that its uncertainty is addressed by other parameters, [[
]]. In the RAI, the NRC staff noted the upper plenum spray model as discussed in Section 7.8.2 of Reference 16 provides some discussion that appears to contradict. In particular, the calculated upper plenum spray trajectories could affect mixing in the upper plenum.

The vendor begins its response to RAI 58 by stating, "The premise of the question is that item F2 is important. This is not the case." Even so, Reference 1 assigns a high importance to the phenomenon. The discussion in the PIRT table provides the basis: [[

]] (Reference 1, Table 4.2-1, Entry F3). In the RAI response, however, GEH explains that emergency core cooling interaction and mixing in the upper plenum is addressed by several other phenomena in addition to those listed in Section 5.1.6.2 of Reference 1 (i.e., [[
]]). This includes the upper plenum spray distribution and CCFL. The vendor makes this observation from the standpoint that the importance is assessed relative to the phenomenon's impact on the critical safety parameters, which relate to the core. The key impact of upper plenum mixing is the availability of liquid to flow into a hot channel. [[

]] Based on the consideration of these additional phenomena in play relative to the influence of upper plenum mixing on the critical safety parameters, the NRC staff determined that the GEH assessment of the importance of upper plenum mixing and ECC interaction is acceptable.

In conclusion, the NRC staff review determined that there was generally acceptable agreement between the GEH PIRT for TRACG-LOCA, and other, contemporary PIRTs. In cases where GEH assigned a lower-rank importance to a particular phenomenon as compared to other PIRTs, the NRC review determined that (1) GEH provided an adequate basis for its particular ranking, and (2) despite a lower ranking, the phenomenon is either treated statistically (if ranked

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[E] The uncertainty attributed to PIRT Item M3 defers to PIRT Item F3. The results shown in response to RAI 13 indicate that [[

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medium), or the uncertainty associated with the phenomenon is addressed by other relevant phenomena that are also considered in the PIRT. Based on these considerations, the NRC staff determined that GEH's PIRT for TRACG-LOCA was acceptable.

3.4. ACCIDENT SCENARIO IDENTIFICATION – CONCLUSION

Based on the considerations discussed in the preceding sections, the NRC staff determined that GEH is consistent with the guidance associated with specifying an accident scenario. The review established that TRACG-LOCA is to be used for analyzing ECCS performance in the BWR/2-6 product line, for GNF fuel. Specific considerations for other fuel designs are reflected in Section 3.1 of this SE. The EM will be used to determine compliance with the acceptance criteria contained in paragraphs (b)(1) though (b)(3) of § 50.46, with additional consideration of the remaining criteria as discussed in Section 3.2 of this SE. Finally, GEH has applied the PIRT process in an acceptable way to ensure that the important phenomena relative to the critical safety parameters are appropriately treated in the evaluation model.

4.0 TRACG BACKGROUND AND EXECUTION

This section provides a brief background on the origin and development of TRACG for an unfamiliar reader. Sections 4.1.1 and 4.1.2 are included as a confirmation that the vendor identified the "frozen version" of the computer code that will be used for ECCS evaluation, and the appropriate, supporting documentation. This content is provided in accordance with the review procedures outlined in SRP 15.0.2 (Reference 10, Page 3). Section 4.2 identifies the prior applications to operating BWR analysis, for which TRACG has been approved for use. Finally, Section 4.3 reviews topics related to the TRACG numerical methods used in ECCS evaluation, consistent with guidance provided in RG 1.157 (Reference 8).

4.1. TRACG OVERVIEW

The Transient Reactor Analysis Code (TRAC) family of codes began as a pressurized water reactor analysis code developed for the NRC at Los Alamos National Laboratory. A BWR version of the code was developed jointly by the NRC and GEH at the Idaho National Engineering Laboratory as TRAC-BD1/MOD1; the primary objective of this effort was to develop the capability to simulate BWR LOCAs.

In the mid-1980s, GEH developed a proprietary version of the code designated as TRACG. The objective of the proprietary code development was to have a code capable of realistic analyses of transients, stability, and anticipated transients without scram (ATWS) events. Further developments for the TRACG code have included the implementation of a three-dimensional kinetics model and an implicit integration scheme. The basic thermal-hydraulic model is a two-fluid model explicitly represented in the code with six conservation equations and appropriate closure relationships.

4.1.1. Frozen Code Version Selection

TRACG04P is selected as the computer code used in the analysis.

4.1.2. Provision of Complete Code Documentation

TRACG is described in GEH LTR NEDE-32176P, "TRACG Model Description" (Reference 16). Its general purpose qualification is provided in NEDE-32177P, "TRACG Qualification"

(Reference 17). Specific applications of the TRACG code are documented in NRC-approved LTRs (References 18 - 22). Table 3 summarizes the TRACG AOO applications for US operating BWRs.

4.2. SUMMARY OF PREVIOUS REVIEW FINDINGS RELATED TO TRACG FOR AOO APPLICATIONS

LTR Number	Title	Notes	Accession No.
NEDE-32906P-A, Rev. 1 (Reference 18)	TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis	Approved TRACG02A for realistic analysis of AOOs	ML060390557
NEDE-32906P-A, Rev. 3 (Reference 19)	TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analysis	Corrected a small error in the quantification of the accuracy of the void coefficient	ML062720163
NEDE-32906P-A, Supplement 1 (Reference 20)	TRACG Application for Anticipated Transient Without Scram Analyses	Extended TRACG02A acceptance for modeling anticipated transients without scram	ML033381073
NEDE-32906P-A, Supplement 2 (Reference 21)	TRACG Application for Anticipated Operational Occurrences Transient Analysis	Modified approach for calculating critical power ratio during transient analysis	ML060800312
NEDE-32906P-A, Supplement 3 (Reference 22)	Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients	Implemented TRACG04P and PANAC11 for use in AOO, ASME Overpressure, and ATWS Overpressure analyses	ML110970401

Table 3. Summary of Previous Review Findings Related to TRACG for AOO Applications.

4.3. NUMERICAL METHODS

In the course of its review, the NRC staff evaluated the TRACG numerical methods with regard to their adequacy for ECCS evaluation. This aspect of the evaluation considered the TRACG nodalization, time discretization, and steady-state initialization. The NRC staff generated four RAIs related to the execution of TRACG for ECCS evaluation. These RAIs, summarized in Table 4, are addressed in this section of the SE.

Batch	RAI	Topic	Ref.
1	41	6-Sector VSSL Nodalization Sensitivity	3
2	71	Changes to Nodalization	4
1	10	Time Step Sensitivity	3
2	77	Steady-State Initialization	4

Table 4. RAI Responses Related to Numerical Methods.

This portion of the review evaluated GEH's consistency with Regulatory Position 2.1.1, "Numerical Methods," of RG 1.157 (Reference 8).

4.3.1. Nodalization

Regulatory Position 2.1.1, "Numerical Methods," of RG 1.157 states, "*Numerical simulations of complex problems such as those considered here, treat the geometry of the reactor in an approximate manner, making use of discrete volumes or nodes to represent the system. Sensitivity studies and evaluations of the uncertainty introduced by noding should be performed.*"

The basic nodalization philosophy used in TRACG-LOCA is provided in Chapter 6 of Reference 17. The general approach is to apply nodalization detail to the plant model that is consistent with the models used in the integral effects test (IET) comparisons. As applied to a plant, the axial levels and node boundaries in the vessel are selected so as to resolve important geometric features such as the recirculation pump suction, the top of the jet pump, and the location of important core features such as the core support plate and the lower tie plate.

Nodalization sensitivity is discussed in Section 5.2 of Reference 1, and is also addressed, with respect to ECCS-LOCA analysis, in Section 6.9.2 of Reference 17. The vendor addressed nodalization sensitivity by increasing the noding detail in regions of the system model, using separate studies for each region in which detail was increased. Generally, the studies showed that **[[**

]] which is discussed in Section 4.3.2 of this SE. These studies show that increasing the noding detail in various regions of the model can cause variations in PCT, but that **[[**

]].

[[

]]

It should be noted, however, that GEH attributed a significant amount of variability associated with its results to the parallel channel effect, which is defined and discussed in Section 7.4 of this SE. In short, the parallel channel effect introduced additional variability when attempts were made to quantify code or computational (i.e., non-parametric) uncertainty. As such, the variability associated with the timestep sensitivity study, or with other such measures like small perturbation analyses, tended to be inflated. The same variability was observed in the BWR/3-6 nodalization sensitivity study results, making it difficult to discern the true adequacy of the nodalization.

During the review process, GEH added significantly more detail to the core model, which had the effect of reducing the model sensitivity to the parallel channel effect. Because of this, the variability associated with both time step sensitivity and small parametric perturbations was reduced. Given this reduced variability, GEH incorporated additional nodalization sensitivity studies specifically for the detailed, BWR/2 model. The more detailed, BWR/2 model exhibited significantly less PCT sensitivity to changes in the level of detail in the vessel nodalization. For example, increasing detail in the vessel component, as well as in selected regions within the vessel, such as the upper plenum, bypass, and lower plenum, showed less than **[[ ]]** fluctuation in PCT, in each case. For comparison, the BWR/2 timestep sensitivity study indicated that the standard deviation associated with time step variation was **[[ ]]**, and the small perturbation analyses performed in response to RAI 9 showed a variation (min to max for a sample of 59 cases) of **[[ ]]**.

The NRC staff issued RAI 41 to query the existence of a **[[ ]]** azimuthal sensitivity study, which is referenced in Section 5.1.1.9 of the LTR. This section discusses the uncertainty associated with 3-dimensional effects in the lower plenum, and GEH referenced the **[[ ]]** sensitivity study. In the response to RAI 41, GEH clarified that this study had been performed in 2002, but was not included in the qualification LTR. The vendor provided further information quantifying that the increased detail caused the PCT to **[[ ]]**. GEH used this information to conclude that the vessel azimuthal detail in the standard nodalization was adequate. Based on the minor change in PCT, the NRC staff agrees with GEH's conclusion.

Based on the NRC staff review of the information contained in LTR Section 5.2, as discussed above, the NRC staff determined that GEH has investigated the effect of nodalization and concluded that the default nodalization is adequate for ECCS-LOCA analysis. Specifically, the BWR/2, detailed core model nodalization sensitivity studies showed acceptably small fluctuation in PCT when compared to more detailed nodalization schemes. On this basis, the NRC staff determined that GEH's approach to nodalization is consistent with the guidance in RG 1.157 and hence acceptable. The following paragraphs address a generic process for increasing nodalization detail, and also provide a limitation requiring that the nodalization sensitivity studies for the BWR/4 plant be updated, and results reported to the NRC staff, prior to plant-specific implementation of TRACG-LOCA for BWR/3-6s.

In LTR Section 5.2, GEH provides a process to modify the default nodalization for LOCA analysis, stating, "Additional details may be added or changed from the standard nodalization provided the changes are shown not to invalidate the qualification bases and the effect on modeling biases and uncertainties are assessed." In RAI 71, the NRC staff requested additional information concerning this process. In the response to RAI 71, GEH stated that the LTR specifies the least-detailed nodalization that is acceptable, meaning that such changes would only serve to increase the level of detail. GEH provided an example, wherein **[[**

]] The process requires confirming that the added detail does not cause results to differ in a statistically significant way.

In considering this information, the NRC staff determined that this process, if implemented administratively, would be subject to the requirements contained in 10 CFR 50.59, "Changes, Tests, and Experiments." Specifically, an update to the nodalization would be considered a change to an element of TRACG-LOCA, and would need to satisfy the *conservative or essentially the same* clause that excludes such a change from requiring prior NRC review and approval in facility-specific implementation.[F]

In addition to the above considerations, the NRC staff recognizes that the initial nodalization sensitivity studies were performed for the BWR/4 demonstration plant, but not subsequently updated using the more detailed core model, as the BWR/2 demonstration plant was used instead. Thus, the NRC staff will require GEH to perform updated nodalization sensitivity analyses and report the results to the NRC staff for each of a BWR/4 and BWR/6 demonstration plant prior to allowing BWR/3-6 implementation. This is Limitation 7, "BWR/3-6 First-of-a-Kind Application," as discussed in Section 10.7 of this SE.

4.3.2. <u>Time Discretization</u>

Regulatory Position 2, "Considerations for Thermal-Hydraulic Best-Estimate Codes," of RG 1.157, reflects the NRC staff guidance applicable to the discretization of transient time. Regulatory Position 2.1.1, "Numerical Methods," states, in part, "*Numerical methods treat time in a discrete manner, and the effect of time-step size should also be investigated*."

This topic is addressed by the vendor in Section 6.9.2 of Reference 17. Reference 17 provides a sensitivity study that varies the maximum time step between **[[**
]] for a BWR-4 large-break LOCA. In RAI 10, the NRC staff requested that the vendor expand this sensitivity study to small- and intermediate-break events for the BWR/4, and to a BWR/2 large-break.

In the response, GEH updated the time step sensitivity study to include the requested plants and break sizes, and also to use the updated, more detailed core model (see SE Section 5.3). The studies included **[[ ]]** cases with increasing time step sizes. The results associated with the more detailed core model, which will be used in production safety analysis, showed very little variation in PCT as a function of the time step size. The standard deviation in PCT of the **[[ ]]** cases associated with each sensitivity study ranged from **[[ ]]**. Shown in Figure 1 is an example of the sensitivity study result, which applies to a BWR/4 intermediate break using the detailed core model (Reference 3, Figure R10-5).

[F] Although 10 CFR 50.59 requirements do not apply when a more specific regulation such as 10 CFR 50.46 provides more explicit requirements for accomplishing a change, both requirements must be considered individually, each on its own merits. Additional discussion on the interplay between these requirements is available in RIS 2016-04, "Clarification of 10 CFR 50.46 Reporting Requirements and Recent Issues with Related Guidance Not Approved for Use" (Reference 23). In short, to be implemented administratively, the nodalization change would not be permitted to constitute a *departure from a method…* and reporting the effects of such a change would fall under the requirements of 10 CFR 50.46(a)(3).

[[

]]

Figure 1. Time Step Sensitivity for BWR/4 Intermediate Break with Detailed Core Model.

Based on this result, the NRC staff determined that GEH has investigated the effect of time-step size and concluded that, in the range of the TRACG default maximum time step size of **[[**

]]. On this basis, the NRC staff determined that the time discretization used in TRACG-LOCA is consistent with the guidance in RG 1.157 and hence acceptable.

4.3.3. <u>Initialization</u>

Although not addressed explicitly in the applicable regulatory guidance, it is important, when analyzing a thermal-hydraulic transient system response, to ensure that the system model can achieve a converged, steady-state initial condition. This is an indication that the code performs in a stable manner, and a confirmation that the initial system parameters have been represented correctly.

In the response to RAI 77, GEH explained its process for ensuring that a steady-state calculation has adequately converged prior to performing transient calculations. The process includes checking a limited set of heat balance parameters, to ensure that any fluctuations are within a specified tolerance. GEH stated that the tolerances are approximately **[[**
]]. The steady-state case is typically run **[[** **]]** after reaching the specified tolerances to ensure that the parameters continue to remain within tolerance.

The approach described above will ensure that the steady-state condition is properly modeled prior to initiating a transient analysis. Based on this consideration, the NRC staff determined that the GEH process for ensuring steady-state convergence is acceptable.

5.0 TRACG-LOCA METHODOLOGY

The TRACG-LOCA methodology relies on a combination of extensive system analysis and simplifying assumptions to identify appropriately limiting conditions prior to performing a statistical uncertainty analysis. Most variability in initial and boundary conditions is addressed through detailed, deterministic system analysis that identifies several sets of plant-specific limiting conditions to evaluate using the statistical analysis. The EM also addresses some aspects of initial and boundary condition variability through the use of simplifying conservative or bounding assumptions. Detailed core modeling, including the use of multiple limiting channels located in two separate rings of the core, addresses the variability of cycle design-dependent parameters, such as power distribution, peaking factors, and exposure.

The NRC staff review of the analytic methodology is based on Regulatory Position 3.1, “Initial and Boundary Conditions and Equipment Availability,” of RG 1.157, which states as follows (Reference 8):

> *The heat generated by the fuel during a loss-of-coolant accident depends on the power level of the reactor at the time of the loss-of-coolant accident and on the history of operation. […] Given the assumed initial conditions, relevant factors such as the actual total power, actual peaking factors, and actual fuel conditions should be calculated in a best-estimate manner.*
>
> *The calculations performed should be representative of the spectrum of possible break sizes from the full double-ended break of the largest pipe to a size small enough that it can be shown that smaller breaks are of less consequence than those already considered. The analysis should also include the effects of longitudinal splits in the largest pipes, with the split area equal to twice the cross-sectional area of the pipe. The range of break sizes considered should be sufficiently broad that the system response as a function of break size is well enough defined so that interpolations between calculations, without considering unexpected behavior between the break sizes, may be made confidently.*
>
> *Other boundary and initial conditions and equipment availability should be based on plant technical specification limits. These other conditions include, but may not be limited to, availability and performance of equipment, automatic controls, and operator actions. Appendix A to 10 CFR Part 50 requires that a single failure be considered when analyzing safety system performance and that the analysis consider the effect of using only onsite power and only offsite power.*

Based on the above guidance, the NRC staff review of the initial and boundary conditions and equipment availability addressed the following:

(1) Time in cycle
(2) Operating statepoints
(3) Power distribution and channel groupings
(4) Break spectrum analysis
(5) Adherence to TS LCOs and equipment performance
(6) General Design Criterion 35 compliance

A complete description of the way that GEH treats initial and boundary conditions requires review of both Chapters 6 and 8 of Reference 1, as well as consideration of additional commitments made in response to RAIs. Such discussion is provided in detail in the succeeding subsections of this SE. Generally, GEH distinguishes between initial conditions and plant parameters, and the items identified in the list above fall within these two categories. Section 6.2 of Reference 1 addresses initial conditions, and Section 6.3 addresses plant parameters.

The RAI responses relevant to this review are listed in Table 5, below. They are listed in the order in which they are discussed.

Batch	RAI	Topic	Ref.
1	27	Operating Domain Applicability	3
2	84	Disposition for Increased Core Folow	4
1	3	Channel Grouping	3
1	6	Channel Grouping	3
1	7	Channel Grouping and Analysis Resolution	3
1	9	Channel Grouping	3
2	72	Hot Channel Power Distribution Initial Conditions	4
2	73	Hot Channel Power Distribution Initial Conditions	4
2	74	Hot Channel Power Distribution Initial Conditions	4
4	102	Hot Channel Power Distribution Initial Conditions	6
1	4	Single Parameter Sensitivity Studies	3
1	31	Use of De-biased Simulations for Analysis	3
1	57	Discharge Flow Uncertainty	3
2	89	Break Spectrum Analysis	4
2	86	Treatment of TS-Controlled Parameters	4
2	87	Applicability of Generic Scram Time Curve	4
2	75	Control Blade Interference/Scram Time	4
1	12	Treatment of GDC 35	3
2	76	Treatment of GDC 35	4

Table 5. RAI Responses Related to TRACG-LOCA Methodology.

5.1. TIME IN CYCLE

Fuel rod exposure is addressed in Section 6.2.7 of the LTR. The LTR provides a general description of the cycle-dependent burnup characteristics of GE14 fuel. The LTR states, "fuel rod power is restricted based on the fuel rod exposure (i.e., PLHGR limit). The thermal mechanical design envelope for allowable PLHGR versus exposure provides the limiting conditions for the rod exposure." GEH considers a range of fuel rod exposures, including for both UO_2 and Gadolinia rods. Limiting channels are included in the core model to represent the life of the fuel. While the minimum number of hot channels associated with a core model is **[[ ]]**, the vendor will include additional hot channel components to represent additional burnup points in the plant MAPLHGR curve. This topic is addressed in more detail in Section 5.3 of the SE.

Many exposure-dependent effects correlate to the time-in-cycle, but are addressed by other aspects of the model. For example, Section 6.2.5 of the LTR describes the exposure-dependent characteristics of the radial and axial power peaking. In addition, the stored energy and decay heat models require either explicit or bounding treatment of time-in-cycle. Fuel rod design characteristics, such as fuel thermal conductivity and pellet-cladding gap thermal conductivity, are also exposure-dependent.

5.2. OPERATING STATEPOINTS

Permissible operating statepoints for a BWR are defined by the power-to-flow operating domain. The power-to-flow operating domain governs important initial conditions such as the axial void profile and the total core power. At the time of the review of NEDE-33005P, the maximum operating domain for BWRs extended to 120-percent of the original licensed thermal power level (OLTP), and at that power level, the permissible flow range extended from a minimum of

80-percent of recirculation flow, up to 108-percent. This is known as the Maximum Extended Load Line Limit Analysis Plus (MELLLA+) operating domain.

A typical MELLLA+ operating domain is illustrated in Figure 2, which is a reproduction of Figure 6.2-1 of NEDE-33005P.

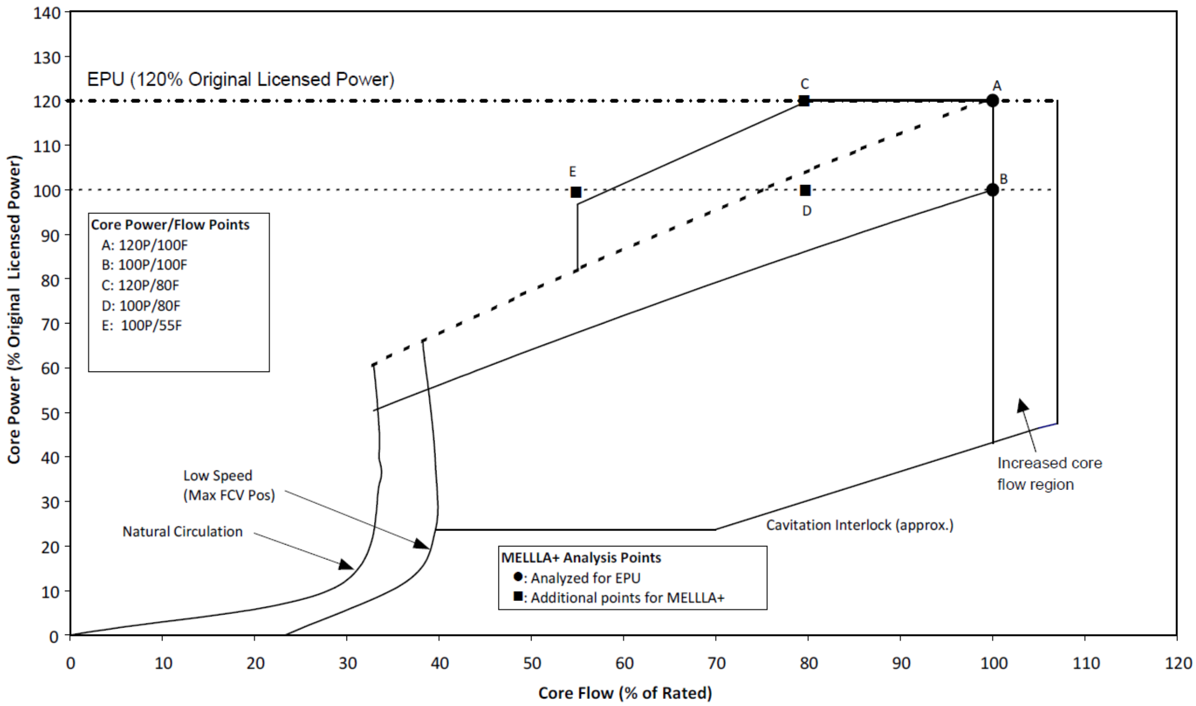

Figure 2. MELLLA+ Power-to-Flow Operating Domain.

The system response is studied for a variety of statepoints, to envelope the operating domain. For MELLLA+ plants, this means that small- and large-break LOCAs are analyzed to cover points A, C, and E in Figure 2. As noted in the response to RAI 27, the power-to-flow operating domain is plant specific in that not all plants are licensed to operate in the MELLLA+ operating domain, and not all plants have received license amendments to implement EPUs.[G] Thus, as noted in the response to RAI 27, "Each application will use the plant-specific power-flow map and determine the power-flow condition that leads to the highest PCT."

The Increased Core Flow (ICF) region may not always be analyzed. As noted in the response to RAI 84, **[[**

]] Thus, GEH states that the ICF region will be evaluated on an application-specific basis to determine if analysis is required.

GEH will deterministically increase core power by two-percent to account for calorimetric uncertainty. In some cases, NRC licensees have installed ultrasonic flow meters to improve this

[G] In addition, not all BWRs that implement EPUs uprate to the full 120%-original licensed thermal power (OLTP) value. An EPU is considered an uprate that increases the licensed thermal power output anywhere between 7- and 20-percent of OLTP.

value, and have reduced the calorimetric uncertainty. If this is the case, licensees implementing TRACG-LOCA may use an appropriately justified thermal power uncertainty. This does not apply, however, to any reduced calorimetric uncertainty associated with a UFM for which the NRC has withdrawn its approval. Such licensees must revert to the generic, 2-percent uncertainty. This is Limitation 2.6, as discussed in Chapter 10 of this SE. Refer also to Reference 24 for additional details.

5.3. POWER DISTRIBUTION AND CHANNEL GROUPINGS

The treatment of the power distribution is described in Section 6.2.5 of the LTR. The treatment was revised significantly during the NRC staff review. While the original LTR specified the use of **[[ ]]** channel groupings in a typical core model, the response to RAIs 3, 6, 7, and 9 indicated that GEH would instead use a more detailed core model, comprised of at least **[[ ]]** channel groups; **[[ ]]**. In addition, the treatment of power peaking factors in the hot bundle was revised by the response to RAIs 72-74. As noted in the response to RAI 102, the revision requires **[[ ]]**. The NRC staff evaluation pertains to the revised model, and is supplemented with additional approval conditions that ensure that relevant factors such as the actual total power, actual peaking factors, and actual fuel conditions are calculated in a best-estimate manner, as recommended by RG 1.157.

The peak bundle power at a given plant has several constraints that are important in the ECCS evaluation. These include the axial power distribution, the peak linear heat generation rate, and the critical power ratio. The axial power distribution tends to be tightly correlated with cycle burnup, with bottom-peaked power shapes more likely for fresh fuel bundles, and top-peaked power shapes becoming more dominant as the exposure increases. GEH stated that, for a given bundle power, and axial and local power shapes, the maximum LHGR or the minimum MCPR will yield the highest PCT. The demonstration analyses indicate that a basis for a limiting power shape can be identified, and the BWR/4 demonstration analyses, in particular, include sensitivity studies of the limiting axial node.

The following excerpt from LTR Section 6.2.5 provides a detailed description of the constraints on bundle peaking:

> Three limits constrain the design and operation of fuel bundles: Thermal Mechanical Operating Limit (TMOL), which is the limiting Peak Linear Heat Generation Rate (PLHGR), MAPLHGR [Maximum Average Planar Linear Heat Generation Rate] and Operating Limit Minimum Critical Power Ratio (OLMCPR).
> **[[ ]]**

The LTR includes several figures, Figures 6.2-2 to 6.2-6, which illustrate cycle performance with respect to margin to LHGR and MCPR operating limits for a variety of plant vintages and fuel bundle designs. The response to RAI 73 also includes similar data for more recent bundle designs, including GE11, GE12, and GNF2 fuel. The figures in the response to RAI 73 confirm that modern fuel bundle designs perform consistently with the designs considered in the LTR. All of these figures illustrate the general trend in cycle performance described above, namely that [[

]]

In the response to RAI 102, GEH summarized its revised approach to modeling the hot channel groupings in the core. Based on the RAI response, GEH will ensure that each core is modeled [[ ]]. As described in the RAI response, the channels capture the variations in axial peaking [[ ]]. The vendor also captures variations in limiting fuel and thermal-hydraulic conditions [[

]] Variability in core spray flow availability is addressed [[

]]

Sensitivities to the axial location of the peak node are addressed in the demonstration analyses. An explicit sensitivity study is provided in Section 8.1.4.2 for the BWR/4 analysis. A similar study is provided for a BWR/2 in response to RAI 29, which is addressed in Section 8.2 of this SE. Regarding the axial power distribution, GEH notes that the results for the critical safety parameters are sensitive to the initial conditions and a basis for the limiting initial condition can be established. For the demonstration analyses, the limiting heated nodes are [[ ]] for the bottom-peak, and [[ ]] for the top-peak.[H] Sensitivity studies confirm that this is the case; for the BWR/4, the study results are provided in Table 8.1-5 of Reference 1.

The NRC staff determined that this analytic approach provides an acceptably detailed core model to capture the effects of variation in power distribution, time-in-cycle, and steady-state thermal-hydraulic performance, as recommended by RG 1.157. Since the model and sensitivity studies account for all of these effects, the NRC staff determined that TRACG-LOCA is acceptable in this regard. Noting the processes described in response to RAI 6 and RAI 102, Limitations 2.1, "Core Detail," and 2.2, "Hot Channels," apply as described in Chapter 10 of this SE.

5.4. BREAK SPECTRUM ANALYSIS

The break spectrum analysis is performed for the limiting statepoint. The spectrum includes both recirculation discharge and suction line breaks, breaks in the main steam lines, and ancillary line breaks like the core spray line. The response to RAI 4 discusses the use of sensitivity studies to identify limiting conditions. The response to RAI 31 discusses the use of de-biased code runs to complete the spectral analysis. The response to RAI 89 follows on to RAI 31. The response to RAI 57 discusses uncertainty in the critical flow model. The response is tangentially related to the break spectrum, since the effective break area is the product of the break area and the discharge flow uncertainty; however, the response to RAI 57 is more appropriately evaluated in Section 6.2 of this SE.

[H] Note that the standard channel nodalization includes [[ ]] axial nodes, of which 25 are heated.

The deterministic break spectrum includes a variety of split break sizes, as well as a double-ended guillotine break (DEGB). The split breaks are analyzed at a resolution that is sufficient to permit overlap, in terms of effective break area, between successive break sizes when considering the discharge flow uncertainty. The statistical analysis includes consideration of the discharge flow uncertainty, and is performed for the limiting break size. If a conclusively limiting break size is not identified, the potentially limiting breaks are treated as sensitivity studies. These modeling approaches are revisited in Chapter 7, "Estimation of Overall Calculational Uncertainty," of this SE. Limitation 4.1 in Chapter 10 of the SE requires GEH to adhere to this analytic approach.

As described in the demonstration analyses in LTR Chapter 8, the analytic method is not prescriptive with regard to whether model biases should be removed from, or compensated in, the computer code prior to analyzing the break spectrum to determine the limiting break size (i.e., "de-biased").[I] The NRC staff questioned this approach in RAI 31, seeking justification for analyzing the break spectrum without first de-biasing TRACG. Similar considerations would also apply in performing system analyses to identify the limiting initial conditions. Namely, performing a series of analyses without first correcting the code for biases could result in the erroneous identification of the limiting break size or other initial conditions. In the response to RAI 31, GEH provided several comparisons to illustrate that implementing the more detailed core model had a more significant effect on the break spectrum than did the compensation for model bias in the code. For the limiting intermediate break size region, this behavior is apparent in Figure R31-1 of the RAI response. When comparing the detailed core model using both biased and de-biased TRACG models, it is clear that the break spectrum performs, for the limiting break size region, very similarly. Nonetheless, GEH pointed out that the break spectrum is traditionally determined from best-estimate nominal results, from which the NRC staff understands that GEH will analyze plant-specific break spectra using de-biased code runs.

The NRC staff continued its evaluation of this issue with follow-on RAI 89. While RAI 31 and its response focused on the use of biased or de-biased results relative to the evaluation of the break spectrum, RAI 89 focused on whether the nominal break spectrum appropriately identified the break characteristics that would be limiting when the statistical analysis was performed and the corresponding upper tolerance limits were determined. In the response to RAI 89, GEH executed statistical analysis for **[[ ]]** break sizes clustered around the limiting break size. The results showed that, among the **[[ ]]** cases, the one-sided upper tolerance limit PCT varied within a range of **[[ ]]**. GEH compared this to the estimated analysis resolution[J] associated with the detailed channel grouping model, which is on the order of **[[ ]]**, and concluded that the variation was insignificant. Moreover, the limiting one-sided upper tolerance limit was **[[ ]]** than that associated with the one-sided upper tolerance limit determined for the nominally limiting break size.

Based on its review of the sensitivity studies provided in the LTR, as supplemented with analyses provided in the RAI responses, the NRC staff determined that GEH appropriately analyzes the break spectrum to determine the limiting breaks. Thus, the NRC staff concluded

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[I] GEH introduces the term "de-biased" in the response to RAI 89, stating, "The biased results in the response to… RAI-31 refer to the TRACG calculation results in which those TRACG model biases determined in Section 5 of the LTR are not removed. On the contrary, the non-biased results in the response to… RAI-31 refers to the results in which the TRACG model biases are removed. The non-biased results are actually, more precisely, de-biased results, which will be called hereafter."

[J] Refer to Section 7.4 for discussion of the analysis resolution.

that GEH is consistent with the portion of Regulatory Position 3.1 of RG 1.157, which recommends that calculations be representative of the full spectrum of possible break sizes. The GEH approach also conforms to the passage in 10 CFR 50.46(a)(1)(i), requiring postulated LOCAs of different sizes, locations, and other properties be calculated, sufficient to provide assurance that the effects of the most severe hypothetical LOCAs are calculated. Based on these considerations, the NRC staff determined that TRACG-LOCA is acceptable with respect to the break spectrum analysis. Limitation 2.3, "Break Spectrum Analysis," specifies the appropriate approach to identify the limiting break for statistical analysis.

5.5. ADHERENCE TO TS LCOS AND EQUIPMENT PERFORMANCE REQUIREMENTS

Chapter 6 of the LTR describes the EM approach with regard to initial conditions and plant parameters. Many parameters that fall into either category are controlled by facility TS or related, design-basis requirements. The LTR makes a distinction between plant parameters and initial conditions as follows:

> A distinction is made in [NEDE-33005] between *initial conditions* and *plant parameters*. The initial rated conditions for a nuclear power plant, specified in absolute units, are considered as plant parameters in certain contexts. In [NEDE-33005], however, those key plant inputs that determine the overall steady-state nuclear and hydraulic conditions prior to the transient are considered to be *initial conditions*…
>
> The term *plant parameter* is reserved for quantities such as protection system setpoints, valve capacities and stroke times, and scram characteristics that influence the characteristics of the transient response but do not have an effect on steady-state operation.

5.5.1. Initial Conditions

A number of initial conditions, including total core power and flow, limiting bundle power distribution, average bundle power distribution, and fuel rod exposure, are addressed in the preceding sections of this chapter. Remaining initial conditions include feedwater temperature, steam dome pressure, and downcomer water level.

The treatment of feedwater temperature is described in Section 6.2.2; **[[**

]]

The treatment of steam dome pressure is described in 6.2.3 of the LTR. Steam dome pressure is a TS-controlled parameter. The vendor stated that **[[ ]]** is expected to be limiting, but evaluated the sensitivity of the results to the dome pressure in the demonstration analyses. The sensitivity studies documented in Section 8 of the LTR indicate that the **[[ ]]**. This is an item subject to plant-specific confirmation as a part of preparation of a qualified base deck and break spectrum analysis.

The downcomer water level is controlled between two alarm setpoints: the low level (L4) and the high level (L7). In a typical BWR/4, this may correspond to a level control range of approximately 0.2 meters. Similar to feedwater temperature and steam dome pressure, sensitivity studies demonstrated [[ ]] The studies discussed in Section 8.1 of the LTR indicated that varying the initial level by [[ ]] In practice, GEH will confirm on a plant-by-plant basis that the downcomer water level does not significantly affect the TRACG-LOCA results, or will use a limiting initial condition. With regard to feedwater level control, this disposition is appropriate in consideration of the fact that the level control system has no safety basis, and the reactor trip and ECCS actuation signals associated with reactor vessel water level are not received until inventory is depleted below L4.

In summary, the analytic sensitivity to initial conditions is evaluated in the TRACG-LOCA analysis. [[ ]] The approach is presented in Chapter 6 of the LTR, and its application is demonstrated in Chapter 8. If the results of the sensitivity studies indicate that the critical safety parameters are sensitive to a particular initial condition, the uncertainty associated with that initial condition will be considered in the analysis, either by using a bounding input, or by applying the uncertainty in the uncertainty analysis. The GEH approach is acceptable, subject to Limitation 2.4, "Initial Conditions and Plant Parameters," as discussed in Chapter 10 of this SE.

5.5.2. Plant Parameters

Regarding the ECCS performance evaluation, GEH states that the plant FSAR provides a list of significant inputs. These inputs are generally analyzed at TS analytic limits, as indicated in Table 6.3-1 of the LTR. The use of TS analytic limits for plant parameters is generally acceptable, because it introduces conservatism into the analysis relative to a given plant's actual, permissible, operating state.

In the LTR, GEH states that realistic uncertainty distributions for TS-controlled parameters may be used. In response to RAI 86, GEH clarified that the use of analytic limits precludes the need to develop and implement realistic uncertainty distributions; however, plants using an NRC-approved instrument setpoint methodology may replace analytic limits with realistic uncertainty distributions. The vendor further stated that NRC licensees seeking to implement such realistic distributions, but without having previously implemented an NRC-approved instrument setpoint methodology, would need to seek prior NRC review and approval. Since the licensees would be using uncertainties associated with an NRC-approved methodology, the NRC staff determined that the adoption of this more realistic approach would be acceptable. Limitation 2.4, "Initial Conditions and Plant Parameters," establishes the requirement for treatment of plant-specific initial conditions, and allows for the approach whereby initial condition uncertainty may be treated more realistically using an NRC-approved instrument setpoint methodology.

Chapter 6 of the LTR introduces and justifies [[ ]] Section 6.3.1 describes the methods [[ ]] and the response to RAI 87 provided additional information supporting the use [[

]].[K] It was determined by modeling steamline breaks for a variety of reactor and containment designs, [[

]] In the response to RAI 87, GEH provided a comparison of the scram times [[

]]

The scram time is set either by the high drywell pressure scram signal, as described above, or by the L3 signal, which is determined mechanistically from the TRACG trip logic. Then, the scram speed is modeled using the TS scram speed applicable to the plant. RAI 75 sought clarification whether scram speed could be affected by equipment degradation, such as shadow corrosion-induced channel bow, or by the attendant effects of a seismic event. In response, GE stated that TS LCOs, Surveillance Requirements, and seismic analysis requirements ensure that such mechanisms do not adversely affect the scram performance relative to the TS requirement. Based on this explanation, the NRC staff agrees that the TS scram speed is an appropriately bounding approach to modeling the scram.

The review described above established that GEH treats initial conditions and plant operating parameters in an appropriate fashion. However, the NRC staff determined that a limitation is necessary to ensure that this practice is carried forward into plant-specific analysis. Excluding uncertainty as determined using an NRC-approved instrument setpoint methodology, variability in initial conditions and plant parameters not specifically addressed in Chapter 6 of the LTR, as supplemented by RAI responses, shall not be analyzed using the statistical analysis, but rather shall be treated deterministically as set forth in revised Chapter 6 of the LTR. Significant plant parameters must be assumed to be in their most pessimistic condition with regard to the results; Technical Specification Analytic Limits for which the ECCS evaluation is an Applicable Safety Analysis shall be used; and parameters shown to have an insignificant effect on the results may be assumed to be in a nominal condition. This is Limitation 2.4, as discussed in Chapter 10 of this SE.

5.6. GENERAL DESIGN CRITERION 35 COMPLIANCE

The analyzed break spectrum is required to address compliance with GDC 35, which requires:

> A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite power system operation (assuming onsite power is not

[K] Plant scram can be initiated by low downcomer water level (L3), or by high drywell pressure. [[

]]

available) the system safety function can be accomplished, assuming a single failure.

Based on the above, the NRC staff review of GDC 35 compliance assesses limiting single failures and the treatment of offsite power availability.

Limiting single failures are addressed within the deterministic break spectrum analysis. Consideration of the limiting single failures specific to a break size range is provided, which includes both a system-level evaluation of hardware availability, and analysis of the consequences of the equipment failure. Examples of this evaluation are provided in the demonstration analyses provided in Chapter 8.

The demonstration analyses presented in Chapter 8 of the LTR use a conservative approach to modeling power system availability. Attributes of offsite power availability, such as a delayed scram in comparison to a loss-of-offsite power, which exacerbate the consequences of the event, are modeled. Otherwise, a loss-of-offsite power is assumed. Using this hybrid approach leads to a conservative modeling approach. Additional information clarifying the modeling approach is provided in the response to RAI 12. The response to RAI 12 also indicates **[[**

]]

In response to RAI 76, GEH clarified further that realistic assumptions, to be consistent with either OPA or LOOP, will be used to show compliance with GDC 35. The vendor also revised text in Table 2.5-1 of the LTR to state, "Consistent with the requirement of General Design Criteri[on] 35, both loss of onsite power and loss of offsite power are assumed individually. System availability to loss of either onsite or offsite power is modeled consistently with realistic plant configurations." The revised table entry indicates that the deterministic analysis will evaluate both scenarios to identify the limiting, which assures that GDC 35 is satisfied.

The response to RAI 12 also indicated that, in the TRACG LOCA application methodology, there is no consideration of any credit gained from non-safety grade systems. This approach is consistent with NRC staff expectations regarding the use of safety-grade systems to mitigate design basis events. The use of only safety-grade systems ensures that there is a high probability that the systems will function as assumed in the analysis.

Based on the review described above, the NRC staff determined that GEH has addressed the requirements of GDC 35 within TRACG-LOCA, and from this perspective, the EM is acceptable. Limitation 2.5, "General Design Criterion 35 Compliance," requires each application of TRACG-LOCA to evaluate whether loss-of-offsite power or offsite power available conditions are limiting.

6.0 REVIEW OF KEY MODELS

Chapter 6 presents the NRC staff review of key TRACG models with regard to ECCS evaluation. This chapter follows the guidance set forth in Regulatory Positions 3.2 through 3.14 of RG 1.157, to the extent that such guidance applies to modeling BWR LOCAs.

6.1. <u>SOURCES OF HEAT DURING A LOSS-OF-COOLANT ACCIDENT</u>

Sources of energy during a postulated LOCA include mainly the initial stored energy in the fuel and the decay heat. The use of the Cathcart-Pawel (C-P) oxidation correlation is also reviewed, as are thermal parameters for swelling and rupture of the fuel rods. Table 6 provides a list of RAIs relevant to this portion of the review, in the order first discussed.

Batch	RAI	Topic	Ref.
1	24	Use of PRIME as opposed to GESTR-LOCA	3
1	42	Dynamic gap conductance uncertainty	3
5	33	Rod internal pressure and fuel pin rupture	7
4	101	Decay heat modeling	6
1	54	Request for correction of Figure 5.1-18	3
3	100	Cathcart-Pawel oxidation limit	5
2	97	Cladding plastic deformation	4

Table 6. RAI Responses Related to Sources of Heat During a LOCA.

6.1.1. Initial Stored Energy of the Fuel

Regulatory Position 3.2.1 of RG 1.157 states, in part (Reference 8):

> *The steady-state temperature distribution and stored energy in the fuel before the postulated accident should be calculated in a best-estimate manner for the assumed initial conditions, fuel conditions, and operating history. To accomplish this, the thermal conductivity of the fuel pellets and the thermal conductance of the gap between the fuel pellet and the cladding should be evaluated. Thermal conductivity of the fuel is a function of temperature and is degraded by the presence of gases in crack voids between fuel fragments.*[L] *An acceptable model for thermal conductivity should be developed from the in-pile test results for fuel centerline and off-center temperatures, taking into account the conductivity of gases in crack voids.*
>
> *Thermal conductance of the fuel-cladding gap is a strong function of hot gap size and of the composition and pressure of the gases in the fuel rod. The calculation of hot gap size should take into account UO_2 or mixed-oxide fuel swelling, densification, creep, thermal expansion and fragment relocation, and cladding creep. […] An acceptable model for the above fuel parameters should be based on in-pile and out-of-pile test data…*
>
> *Best-estimate fuel models will be considered acceptable provided the models include essential phenomena identified above and provided their technical basis is demonstrated with appropriate data and analyses.*

The TRACG-LOCA EM uses fuel parameter inputs supplied by the NRC-approved PRIME code (Reference 25). These PRIME inputs are used by TRACG to calculate the steady-state temperature distribution and stored energy in the fuel. The PRIME inputs also include the composition and pressure of the fuel rod gases, which are used to calculate the gap gas

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[L] Since publication of RG 1.157, additional mechanisms affecting nuclear fuel thermal conductivity as a function of fuel pellet exposure have been identified, which have been shown to affect ECCS performance, as predicted using a realistic evaluation model, significantly. Refer to NRC Information Notice (IN) 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," and IN 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting From Nuclear Fuel Thermal Conductivity Degradation," for further discussion (References 26 and 27).

conductivity. The inputs are generated assuming a realistic operating power history. The PRIME code has been found acceptable for use in simulating BWR fuel performance. During the course of the NRC staff review, GEH updated its analytic platform to replace GESTR-M-based fuel performance analysis with PRIME-based analysis. Thus, the TR as submitted indicated that either of these methods would be used, but this approach has since been modified to phase out the use of GESTR. This is a difference both in the TR as originally submitted, and in the TRACG-AOO application as documented in Reference 22.[M]

In the response to RAI 24, GEH stated that only PRIME would be used for modeling in accordance with the TRACG-LOCA EM (Reference 3). Since the code is NRC-approved and accounts for the phenomena described above, the NRC staff determined that this approach is acceptable. Note, however, that the limitations applicable to the PRIME code, delineated in Section 4.0 of the NRC staff approving SE, are also applicable to TRACG-LOCA analysis. This is Limitation 3.2, “Limitations on the Use of Upstream/Concurrent Methods,” as discussed in Chapter 10 of this SE. Additional justification must be reviewed and approved by the NRC staff in order to apply TRACG-LOCA using a different set of limitations or a different fuel performance model.

Uncertainties associated with the fuel rod stored energy and gap gas conductivity are derived in accordance with the PRIME methods qualification and ported to the TRACG-LOCA evaluation model accordingly (Reference 1, Section 5.1.3.6). Relative to previous TRACG applications, which used GESTR-based uncertainties, the TRACG-LOCA uncertainties include not only pellet heat transfer uncertainty, but also chamfer stored energy and additional power uncertainty. The response to RAI 42 provides additional discussion regarding the inclusion of these terms, explaining that combining them produces a root mean square value of **[[ ]]**. This uncertainty value is evaluated in Section 8.5 of the technical evaluation report enclosed in Reference 25 and found to be acceptable.

The uncertainty in dynamic gap conductance is lumped with the pellet stored energy uncertainty. Thus, the NRC staff reviewed the basis for this **[[ ]]** value to confirm that it included sufficient margin to account for uncertainty associated with the ability of TRACG to model pellet contraction following reactor scram. In the response to RAI 42, the vendor noted that one of the uncertainty parameters combined to reach the **[[ ]]** value was a **[[ ]]** “additional power uncertainty.” A review of Reference 25 (specifically, Page 3-7 of NEDC-33258P-A) indicates that this value is provided for monitoring uncertainty and future concerns, confirming that it is a margin term. This is an acceptable treatment; however, the NRC staff notes that the Limitations provided in Chapter 10.3 of this SE apply; the allocation of this margin term to other items (i.e., “future concerns” as noted in NEDC-33258P-A) within the PRIME approval basis would require revisiting the treatment of pellet stored energy within the TRACG-LOCA approval basis. Sensitivity studies associated with the PRIME implementation into downstream safety analyses demonstrated that the effect of changing the gap gas conductance **[[ ]]**. In a revision to the response to RAI 33, GEH used this **[[**

]] Refer to Section 6.1.5 of this SE for additional discussion.

Based on its review, the NRC staff determined that GEH obtains the initial stored energy using an NRC-approved fuel performance code, and the incorporation of the PRIME-based inputs is

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[M] TRACG-AOO has been updated to eliminate the reliance on GESTR-based models and inputs in production safety analysis.

consistent with other NRC-approved applications of TRACG. In addition, the PRIME-based fuel performance model incorporates models for the phenomena listed in Regulatory Position 3.2.1 of RG 1.157. Based on these considerations, the NRC staff determined that TRACG-LOCA is acceptable with respect to initial stored energy in the fuel.

6.1.2. Fission Heat

Regulatory Position 3.2.2 of RG 1.157 states:

> *Fission heat should be included in the calculation and should be calculated using best-estimate reactivity and reactor kinetics calculations. Shutdown reactivities resulting from temperatures and voids should also be calculated in a best-estimate manner. The point kinetics formulation is considered an acceptable best-estimate method for determining fission heat in safety calculations for loss-of-coolant accidents. Other best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. Control rod assembly insertion may be assumed if it is expected to occur.*

Fission heat is calculated using a point kinetics model, which has been validated against a more detailed, three-dimensional, nodal kinetics model. The 3D model has been approved by the NRC staff. In addition, fission heat contributes a negligible amount of energy following scram. Void reactivity feedback contributes a significant amount of negative reactivity prior to that point. GEH presents its basis for the void coefficient in Section 5.1.3.1 of Reference 1. **[[**

]]

Section 5.1.3.2 of Reference 1 provides the basis for the point kinetics model. A figure compares the point kinetics model to the 3-D kinetics model in TRACG-AOO for a small-break LOCA event, showing reasonable agreement for the power transient.

The vendor uses a point kinetics model, which is consistent with RG 1.157 recommendations. Comparisons of the void reactivity curve and a small-break LOCA power transient between the point kinetics model and the TRACG-AOO-based 3-D nodal kinetics model show reasonable agreement. Based on these considerations, and in light of the fact that kinetics and void reactivity have a highest PIRT ranking of medium, the NRC staff determined the GEH approach for fission heat is acceptable.

6.1.3. Decay of Actinides and Fission Product Decay Heat

Regulatory Position 3.2.3 of RG 1.157 states:

> *The heat from radioactive decay of actinides, including neptunium and plutonium generated during operation as well as isotopes of uranium, should be calculated in accordance with fuel cycle history and known radioactive properties. The actinide decay heat chosen should be appropriate for the facility's operating*

> *history. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.*

Regulatory Position 3.2.4 of RG 1.157 states:

> *The heat generation rates from radioactive decay of fission products, including the effects of neutron capture, should be included in the calculation and should be calculated in a best-estimate manner. The energy release per fission (Q value) should also be calculated in a best-estimate manner. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. The [1979 ANSI/ANS-5.1 Decay Heat Standard] is considered acceptable for calculating fission product decay heat.*

The decay heat model included in TRACG04 is enhanced over what was previously provided in TRACG02. Whereas the TRACG02 code included the May-Witt decay heat model, TRACG04 adds both the 1979 and 1994 ANS decay heat models (Reference 22). The SE approving Reference 22 notes that the 1979 or 1994 standards are used in AOO, ATWS, and ASME overpressure analysis; however, Reference 1 indicates that the 1979 ANS decay heat model is used for ECCS evaluation. The NRC staff notes that this approach is in line with the guidance contained in RG 1.157.

The original LTR contained discrepancies and typographical errors in the description of the TRACG-LOCA decay heat model (Reference 1). The NRC staff review is based on the corrected information, which was provided in response to RAI 101 (Reference 6). The review also considers information provided in response to RAI 54 (Reference 3).

The TRACG decay heat model is carried forward from SAFER/GESTR-LOCA, as described in Appendix B to NEDE-30996P-A (References 28 and 29). TRACG implements decay heat models based on both the 1979 and 1994 ANS standards, like an auxiliary code, DECAY. According to the response to RAI 101, the DECAY computer code was developed to address issues identified in IN 96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly" (Reference 30).

The TRACG decay heat model is simplified in that it assumes **[[**

]] As time from shutdown increases, this assumption drives the TRACG decay heat conservatively higher than the DECAY code which calculates these contributions to the decay heat in a realistic manner.

Additional assumptions in the TRACG decay heat model also contribute conservatism, or enable its use in a bounding manner, when compared to DECAY and the SAFER model. **[[**

]]

Figures R101-1 and R101-2 of the RAI response show that these conservatisms cause the TRACG04 decay power curve to sit slightly above the SAFER and DECAY curves. **[[**

]] Since the SAFER model, from which the TRACG model was derived, is based on the ANS/ANSI 5.1 1979 standard, the NRC staff determined that it is consistent with the recommendations in RG 1.157. The model has already been accepted by the NRC staff, and the DECAY model was implemented to address concerns in IN 96-39. The TRACG model also introduces conservative, simplifying assumptions relative to the previous, legacy models. Based on these considerations, the NRC staff determined that the TRACG04 decay heat model is acceptable for ECCS evaluation.

[[

]] The NRC staff confirmed that the irradiation time is slightly conservative based on conventional uranium loadings in the GNF2 bundle design and typical BWR/4 power levels and core inventory. However, GEH must confirm that this conversion is bounding, based on plant-specific attributes for each application of TRACG-LOCA.

The uncertainty for the model is also based on the standard. In the uncertainty analysis, a value is randomly sampled from the uncertainty distribution, and applied to the decay heat curve **[[**

]] The time-variation in uncertainty for a bundle average exposure of **[[** **]]** is shown in Figure 5.1-16 of the LTR, which was updated in response to RAI 54. The uncertainty in decay heat is applied independently of core power; initial core power uncertainty is treated deterministically **[[** **]]**. Since the decay heat uncertainty approach prevents reducing the uncertainty by averaging effects, and accounts for the time-dependent changes in the decay heat uncertainty in accordance with the ANS 1979 decay heat standard, the NRC staff determined that the treatment of decay heat uncertainty is acceptable for TRACG-LOCA.

6.1.4. <u>Metal-Water Reaction Rate</u>

Regulatory Position 3.2.5 of RG 1.157 states:

> *The rate of energy release, hydrogen generation, and cladding oxidation from the reaction of the zircaloy cladding with steam should be calculated in a best-estimate manner. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses. For rods calculated to rupture their cladding during the loss-of-coolant accident, the oxidation of the inside of the cladding should be calculated in a best estimate manner.*

Further, RG 1.157 indicates that use of the Cathcart-Pawel oxidation kinetics correlation is considered acceptable for calculating the effect of metal-water reaction.

The requirements in § 50.46(b) impose a limit on cladding oxidation of 0.17 times the total cladding thickness before oxidation, or, as stated previously in this SE, 17-percent ECR. The Atomic Energy Commission (AEC) deliberation over the 17-percent ECR acceptance criterion is discussed in detail in the 1973 Opinion of the Commission regarding ECCS acceptance criteria for light-water-cooled nuclear power reactors (Reference 31, pp. 1085 - 1138).

In its proceedings, the AEC noted that the "limits specified in these criteria will assure that some ductility would remain in the zircaloy cladding as it goes through the quenching process" (Reference 31, Page 1096). The values were selected because experimental data indicated that cladding ductility is influenced not only by oxidation alone, but also by the temperature at which the oxidation occurs. The AEC received recommendations from fuel vendors, the AEC staff, and the public regarding the selection of an appropriate oxidation limit. The AEC's consideration included not only the total oxidation but also the thickness of brittle oxidation and zirconium layers in the cladding and the ratio of the thickness of the brittle layers to the remaining ductile layers. Noting wide agreement on the value of 17 percent ECR as a threshold above which cladding generally exhibited brittle behavior, the AEC settled on this value as the cladding oxidation limit.

The experimental studies supporting this limit evaluated cladding ductile performance and correlated it to the thicknesses of the differing layers (i.e., oxide, brittle zirconium, ductile zirconium) rather than to a measured ECR. The percentage values were calculated, based on the test conditions, using the Baker-Just (B-J) correlation. Thus, the AEC also noted that "the Regulatory Staff in their concluding statement compared various measures of oxidation (page 90) and concluded that a 17 percent total oxidation limit is satisfactory, [emphasis added] *if calculated by the Baker-Just equation*" (Reference 31, Page 1097).

Upon revision to § 50.46 in 1988 to allow more realistic emergency core cooling performance calculations, the state of the art for cladding oxidation calculations had evolved. In addition to Baker-Just, Chapter 6.13 of NUREG-1230 reviews Cathcart-Pawel alongside two additional oxidation rate equations (Reference 12). The NUREG-1230, as well as RG 1.157, recommend the use of Cathcart-Pawel based on its superior accuracy when compared to Baker-Just.

However, as noted in Research Information Letter (RIL) 02-02, Attachment 2, the original and confirmatory ring compression tests on which the 17 percent ECR criterion was based relied on an ECR value calculated using Baker-Just (Reference 32). As noted on Page 9 of Reference 32, Attachment 2, "had the Cathcart-Pawel correlation – which did not exist at that time – been used, the cladding oxidation limit would have been about 13%. Therefore, the Baker-Just correlation must be used when comparing results with the old 17% limit."

The use of a 17 percent limit on ECR when applied to cladding oxidation values calculated using the Cathcart-Pawel correlation does not provide the same level of assurance of cladding ductility as the same limit when applied to a result calculated using the Baker-Just correlation. In its present reviews of ECCS evaluation models, the NRC staff is imposing a limitation specifying that the ECR results calculated using the Cathcart-Pawel correlation are considered acceptable in conformance with § 50.46(b)(2) if the ECR value is less than 13 percent. If the ECR value is determined to be less than 13-percent, then there is reasonable assurance that an analogous value, if calculated using the Baker-Just correlation, would remain below 17-percent and still in compliance with the 10 CFR 50.46(b)(2) acceptance criterion.

In response to RAI 100, GEH indicated that several approaches to adhere to this interim limitation would be possible. First, calculations could be performed, using the Cathcart-Pawel correlation, to compare to the 13-percent limit. Second, GEH indicated the Baker-Just correlation could be used for comparison to the 17-percent limit. A third approach would incorporate a more performance-based limit that recognizes the role of various embrittlement mechanisms in establishing acceptance criteria that ensure that the cladding remains ductile, consistent with the intent of § 50.46(b)(2).

The NRC staff determined that a 13 percent limit on the ECR predicted using Cathcart-Pawel is appropriate; this determination is based on a comparison of the test conditions used in the ring compression tests that were used to support the original basis for the 1973 rulemaking. These tests involved oxidizing the cladding at temperatures near the regulatory limit of 2200 °F. At lower oxidation temperatures, the difference between Cathcart-Pawel and Baker-Just diminishes, i.e., a higher, Cathcart-Pawel-calculated ECR value would equate to 17 percent Baker-Just. Therefore, if GEH pursues Approach 1, limiting the Cathcart-Pawel calculated ECR below 13 percent, then there is reasonable assurance that the analogous ECR, calculated using Baker-Just, would remain below 17 percent, provided that the calculated PCT remains below 2200 °F. On this basis, the NRC staff determined that the GEH-proposed Option 1 is acceptable because it would comply with the regulatory acceptance criterion of 17 percent, established in 10 CFR 50.46(b)(2), using the Baker-Just kinetics equation. Since this approach relies on the Cathcart-Pawel equation to determine the ECR critical safety parameter, the NRC staff determined that Limitation 5, “Interim Limitation on Cathcart-Pawel Results,” as discussed in Section 10.5 of this SE.

The use of the Baker-Just correlation, as GEH proposed in Option 2, would also be acceptable. This determination is based on the fact that the acceptance criterion, as originally established, was based on the use of the Baker-Just equation. A Baker-Just equivalent to the Cathcart-Pawel-calculated value would need to be determined for a sufficient number of limiting analytic cases, and nodal locations, to provide assurance that the limiting Baker-Just oxidation ECR for the plant being analyzed has been calculated. Since this approach would use a Baker-Just equivalent of the ECR based on the transient time at temperature from the ECCS evaluation, the NRC staff also determined that this approach would be acceptable. As this approach uses the Baker-Just equation to determine a critical safety parameter to compare against the 17 percent acceptance criterion, the NRC staff did not determine that a limitation applies to this approach.

If GEH were to implement Option 3, i.e., apply a more performance based alternative to the existing, 17 percent, B-J-based acceptance criterion, the licensing process required to implement such an approach in a plant-specific analysis has not been established. Thus, the NRC staff makes no determination whether (1) such an approach would be acceptable on a plant-specific basis, or (2) such an approach would require a specific exemption to the acceptance criterion published in § 50.46(b)(2), pursuant to the requirements of § 50.12. Thus, the NRC agrees with GEH that, “The licensing process needed for an early adoption of the third approach has not been completely defined by the NRC so some risk by the licensee is encountered should they choose this third option” (Reference 5, Enclosure 1, Page 16).

In view of the interim 13-percent acceptance criterion when using the C-P reaction model, should the NRC staff position in this matter change, the NRC will notify GEH with a letter providing a basis for the change and revising the limitation as necessary. This is Limitation 5, discussed in Chapter 10 of this SE.

6.1.5. Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods

Regulatory Position 3.3.1 of RG 1.157 states:

> *A calculation of the swelling and rupture of the cladding resulting from the temperature distribution in the cladding and from the pressure difference between the inside and outside of the cladding, both as a function of time, should be included in the analysis and should be performed in a best-estimate manner. The degree of swelling and rupture should be taken into account in the calculation of gap conductance, cladding oxidation and embrittlement, hydrogen generation, and heat transfer and fluid flow outside of the cladding. The calculation of fuel and cladding temperatures as a functions of time should use values of gap conductance and other thermal parameters as functions of temperature and time. Best-estimate methods to calculate the swelling of the cladding should take into account spatially varying cladding temperatures, heating rates, anisotropic material properties, asymmetric deformation of cladding, and fuel rod thermal and mechanical parameters. Best-estimate methods will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.*

The TRACG gap conductance model is evaluated with regard to ECCS-LOCA evaluation in Section 6.1.1 of this SE. The TRACG fuel rod swelling and rupture models are “based on an empirical fit to experimental data for BWR size fuel rods. The cladding strain is a function of the cladding temperature and the hoop stress” (Reference 1, Table 2.5-1). The gap conductance calculations include allowance for fuel rod and pellet geometry, as well as for transient variations in fuel rod parameters such as plenum temperature and rod internal pressure. Some heat transfer effects outside the cladding, such as droplet shattering upon impingement on ruptured fuel rod segments, are neglected.

The models important for predicting fuel rod rupture include the rod internal pressure and a cladding perforation model.

The reference initial rod pressure is passed into TRACG from the power- and exposure-dependent fuel files obtained from PRIME. In the revised response to RAI 33, GEH provided a plot of PRIME predictions of rod internal pressure as compared to data (a reprint of a figure that was included in Reference 25) **[[**

]] The NRC staff reviewed the referenced RAI response, RAI 46, as contained in Reference 25, i.e., NEDE-33256P-A, Revision 1, and agrees with GEH’s assessment **[[**

]] The study varied the gap gas pressure, **[[**

]] On this basis, the staff accepts GEH’s **[[**

]]

The cladding rupture model correlates the rupture hoop stress to the postulated rupture temperature. This is consistent with the practice established in NUREG-0630; however, **[[**

]] In the revised response to RAI 33, GEH provided curves showing the model, and comparing it to rupture stress and temperature data, including proprietary GEH data and data from NUREG-0630. The upper and lower bounds reasonably envelope the data. [[

]] This approach not only covers the data set, but, based on the way GEH established the upper and lower bounds for the curve, the approach also introduces more spread toward lower rupture temperatures. This, in turn, results in a tendency for early rupture to be predicted. This is somewhat conservative, because an earlier rupture drives higher two-sided cladding oxidation.

In RAI 97, the NRC staff requested that GEH provide additional detail regarding models used to calculate cladding plastic deformation, and the attendant effects that would result from potential in-stack fuel relocation. For example, pellet relocation with a sufficiently high rubble packing factor could increase the heat load to the cladding in the balloon region. In response, the vendor clarified [[

]]
The vendor also described the model for plastic deformation. It reflects the cladding strain behavior that is described in NUREG-0630. Furthermore, GEH clarified that the model has been implemented in the SAFER evaluation model and accepted by the NRC. Finally, [[

]] Based on this study, and on the prior approval of the cladding plastic deformation model and its consistency with NUREG-0630, the NRC staff determined that the GEH disposition for cladding plastic deformation is acceptable.

6.2. BLOWDOWN PHENOMENA

The NRC staff review of blowdown phenomena, including break characteristics and flow, noding near the break, and critical heat flux (CHF) is reviewed in the following sub-sections. The NRC staff review resulted in issuance of 3 RAIs, which are listed in Table 7, below.

Batch	RAI	Topic	Ref.
1	57	Critical Flow Uncertainty	3
1	3	Nodalization Sensitivity	3
1	44	Treatment of GEXL Correlation Uncertainty	3

Table 7. RAI Responses Related to Blowdown Phenomena.

6.2.1. <u>Break Characteristics and Flow</u>

Regulatory Position 3.4.1 of RG 1.157 states:

> *In analyses of hypothetical loss-of-coolant accidents, a spectrum of possible break sizes should be considered, as indicated in Regulatory Position 3.1. The discharge flow rate should be calculated with a critical flow rate model that considers the fluid conditions at the break location, upstream and downstream pressures, and break geometry. The critical flow model should be justified by comparison to applicable experimental data over a range of conditions for which the model is applied. The model should be a best-estimate calculation, with uncertainty in the critical flow rate included as part of the uncertainty evaluation. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.*

The TRACG critical flow model is described in Section 6.3 of Reference 16. It uses separate models for each of liquid-only, two-phase, and vapor-only conditions. For transitions between these conditions, TRACG interpolates between the two applicable correlations.

The discharge model is qualified as documented in both Reference 17, Section 3.4, and in NEDC-32725P, Revision 1, "TRACG Qualification for SBWR [Simplified Boiling Water Reactor]" (Reference 33, Section 3.3.4). The qualification is based on Marviken tests, as well as to data from the Pressure Suppression Test Facility (PSTF),[N] Edwards blowdown experiments, and the GIRAFFE systems interactions test (SIT). In Reference 16, GEH noted that the critical flow model was evaluated also using comparisons to the blowdowns from PSTF, the Two-Loop Test Apparatus (TLTA), Full Integral Scale Test (FIST), and FIX tests. **[[**

]]

In its review, the NRC staff identified concerns with the model treatment of critical flow uncertainty, as documented in RAI 57. The RAI addressed the following topics:

a) The uncertainty assessment is based on nine tests used to define critical flow behavior over a span of liquid to vapor flow regimes
b) The selection of nine tests to the exclusion of other possible, relevant tests
c) The method used to compare TRACG model performance to test data to define an uncertainty distribution
d) The selection of critical flow discharge coefficients for the statistical evaluation
e) The influence of non-condensibles on the predicted behavior

[N] PSTF is described in Section 3.1.5 of Reference 17. The test rig was a vessel with a blowdown pipe at the bottom. To replicate top-break scenarios, the rig could be equipped with a vertical standpipe inside the vessel. The test provided data for comparison to both pressure and level predictions.

In response to Portion a), GEH reviewed open literature for several contemporary realistic ECCS evaluation models and concluded that the uncertainty used in the GEH critical flow model is quantitatively consistent with other methods. The vendor noted that **[[**

]] While the NRC staff accepts that the model uncertainty ranges appear consistent, this argument alone is not particularly compelling. Such a consideration also warrants considering the sampled probability density functions, and the underlying density functions are generally proprietary and thus incomparable.

Notwithstanding the above, GEH also noted that the uncertainty database largely includes similar assessment tests among the methods (i.e., Marviken). In addition, GEH stated that the break spectrum analysis includes analysis of break sizes with such granularity as to overlap the critical flow uncertainty, a feature of the analytic method that reduces the significance of the critical flow model relative to the results.
To demonstrate the combined effect of critical flow uncertainty, or discharge coefficient, and break area[O], GEH simulated a BWR/4 break spectrum. At each break area, GEH performed **[[ ]]** sensitivity studies by ranging the **[[**

]]. These results were plotted alongside a nominal break spectrum analysis, with PCT illustrated as a function of the effective break area (Reference 3, Figure R57-1). **[[**

]]

The above reasoning works for an intermediate-break-limited plant. However, if the double-ended guillotine, or the maximum split break, is limiting, the rationale is invalid. As can be seen in Figure R57-1, the trend of increasing PCT with effective break area continues for both the double-ended guillotine break and the maximum split break. In other words, there is a shallow increase in PCT associated with increases in the critical flow multiplier. However, in the case of the BWR/4, the consequences of these breaks are eclipsed by the limiting intermediate break. Moreover, the uncertainty evaluation at the limiting large break size will explicitly incorporate the effects of critical flow uncertainty via the critical flow multiplier.

The BWR/2 spectrum is generally limited by the double-ended guillotine break. Thus, in Figure R57-2, GEH provided a similar effective break area study for the BWR/2 break spectrum. As noted in the RAI response, **[[**

]] Thus, in a fashion similar to the large break region for the BWR/4 spectrum, the uncertainty analysis will account acceptably for the minor effect caused by critical flow uncertainty for the double-ended break.

[O] In LOCA simulation, the discharge coefficient or critical flow multiplier has a scaling effect on the break area. The product of break area and discharge coefficient is commonly known as the effective break area.

In response to Portions b) and c) of RAI 57, GEH deferred to the relative insensitivity of the PCT to the critical flow uncertainty. In response to Portion d), GEH stated that discharge coefficients are not used, and that the critical flow uncertainty is sampled from its uncertainty distribution.

In response to Portion e) of the RAI, GEH stated that, in BWR LOCA analysis, the ingress and generation of a significant amount of non-condensibles does not occur until the later phases of the event, at which time the flow through the break is no longer choked.

In consideration of all of the above information, the NRC staff accepts the GEH disposition for critical flow uncertainty. Primarily, GEH has provided sensitivity studies **[[**

]] The range studied by GEH exceeds the uncertainty range permitted by its sampling, but that uncertainty range is based on a variety of tests that span conditions in the multiple flow regimes considered in the critical flow model. The sensitivity studies indicated that further widening of the sampling distribution, to values that testing showed were unlikely, would not reasonably be expected to drive a significant increase in the predicted PCT.

Based on the review described above, the NRC staff determined that GEH is consistent with the guidance provided in Regulatory Position 3.4.1 of RG 1.157. The vendor has justified its critical flow model by comparison to relevant experimental data, and considers the uncertainty in the uncertainty analysis. The deterministic break spectrum is performed with sufficient resolution as to permit overlap, in terms of effective break area, with the discharge coefficient uncertainty. GEH has also studied the sensitivity of its analyses to critical flow multiplier, as described above. While the overall transient is somewhat insensitive to the multiplier, the uncertainty treatment ensures that, even in consideration of the double-ended guillotine break event, the effects of critical flow uncertainty are accounted for in the analysis with bounds representative of the experimental database. Based on these considerations, the NRC staff determined that the discharge flow model in TRACG-LOCA is acceptable; the resolution of the break spectrum is addressed as Limitation 2.3 in Chapter 10 of this SE.

6.2.2. <u>Noding near the Break and ECCS Injection Point</u>

Regulatory Position 3.5 of RG 1.157 states:

> *The break location and ECCS injection point are areas of high fluid velocity and complex fluid flow and contain phenomena that are often difficult to calculate. The results of these calculations are often highly dependent on the noding. Sufficient sensitivity studies should be performed on the noding and other important parameters to ensure that the calculations provide realistic results.*

GEH assessed nodalization as discussed in Section 5.2 of the LTR. Nodalization sensitivity studies included increasing the noding detail at the break location for a BWR/4. For the limiting intermediate break, Table 5.2-1 of the LTR indicates that increasing the break resolution effects a PCT reduction slightly less than **[[ ]]**

Although this may seem like a significant amount, an additional consideration explains the magnitude, which GEH discussed in the response to RAI 3. The sensitivity studies were performed using the LTR Revision 0, less-detailed, core model. The statistical trials associated with this model introduced a significant amount of non-phenomenological variability (about [[ ]]) that was subsequently reduced (to about [[ ]]) by using a more detailed core model. GEH estimates that the use of the more detailed core model would reduce the PCT variation associated with break noding [[ ]]. In any case, the [[ ]] difference in PCT associated with the change in break noding is within the analytic resolution of the model used to perform the evaluation. Since GEH studied the break nodalization and determined that associated changes in PCT were within the analysis resolution of the model, the NRC determined that GEH was consistent with this element of RG 1.157. GEH will update its nodalization sensitivity studies based on the revised modeling approaches, consistent with Limitation 7, "BWR/3-6 First-of-a-Kind Application," as discussed in Section 10.7 of this SE.

With specific regard to noding near the break point, it should also be noted that discharge flow tests simulated as part of the TRACG qualification, such as the Marviken tests (Reference 17, Section 3.4.1.6) and the PSTF blowdown experiments (Section 3.1.5.4), also included sensitivity studies of the TRACG nodalization of those tests. These tests are significantly simpler than a reactor analysis; however, these nodalization sensitivity studies indicated that the chosen TRACG nodalization was adequate, as varying noding detail did not significantly affect the predicted outcomes of the tests, in terms of predicted break flow rates and system pressures. These sensitivity studies provide additional evidence regarding the sufficient predictive capability of TRACG at the chosen level of break noding detail.[P]

6.2.3. Critical Heat Flux

Regulatory Position 3.8 of RG 1.157 states:

> *Best-estimate models developed from appropriate steady-state or transient experimental data should be used in calculating critical heat flux (CHF) during loss-of-coolant accidents. The codes in which these models are used should contain suitable checks to ensure that the range of conditions over which these correlations are used are within those intended. Research has shown that CHF is highly dependent on the fuel rod geometry, local heat flux, and fluid conditions. After CHF is predicted at an axial fuel rod location, the calculation may use nucleate boiling heat transfer correlations if the calculated local fluid and surface conditions justify the reestablishment of nucleate boiling. Best-estimate models will be considered acceptable provided their technical basis is demonstrated with appropriate data and analyses.*

The determination of CHF is used to transition TRACG from using nucleate boiling heat transfer correlations to using transition and film boiling, and vapor convection correlations. Within its range of applicability, the General Electric Critical Quality Boiling Length (GEXL) correlation is used. Outside this range, either the modified Zuber or modified Biasi correlations are used. GEXL is NRC-approved, and the modified Zuber and Biasi correlations are commonly used in Appendix K-conformant and realistic ECCS evaluation models.

[P] This additional consideration is presented after the conclusions for two reasons. First, it is not apparent that either of the referenced sensitivity studies was intended to investigate, specifically, the effect of noding near the break relative to predicted discharge flow and system pressure. Second, these tests are significantly simpler than the system analysis, and the comparison is directly to system pressure and flowrate, rather than to a critical safety parameter (as in the system analysis).

Critical power uncertainty is incorporated into the uncertainty analysis. In the response to RAI 44, GEH stated that fuel-type specific values for the GEXL bias and uncertainty will be applied, or a more conservative GEXL bias and bounding uncertainty that bounds all fuel types in the core will be applied. The bounding approach would be used, according to GEH, for competitor fuel designs where determining the precise values is difficult.

Since GEH is using an NRC-approved, fuel design-specific correlation to determine the transition to film boiling, the NRC staff determined that the vendor is consistent with Regulatory Position 3.8, as the correlation reflects the fuel-design specific considerations discussed therein. The more general CHF correlations are acceptable when GEXL does not apply. The bounding uncertainty approach is acceptable for average channels in the core, but hot channel modeling of competitor fuel is restricted by Limitation 1.3. Co-resident fuel in the core would be addressed by the evaluation model furnished by that fuel's vendor. Limitations 3.1 and 3.2, regarding upstream methods, also apply.

6.3. POST-CHF PHENOMENA

Accurate prediction of post-CHF heat transfer over a range of thermal-hydraulic conditions is of particular importance in ECCS evaluation. Acceptable estimation of the PCT requires that the EM can predict both the cooling phenomena present once the core uncovers, and the rate at which liquid coolant in the core recovers. In a BWR, which depends heavily on core spray flow injected above the core, this also requires the code to simulate counter-current flow and predict the core spray distribution. All of these items must be simulated accurately so that both the magnitude and duration of the PCT transient can be predicted. The cladding oxidation phenomena are highly dependent on both the duration and magnitude of the cladding temperature transient.

Like many other system analysis codes, TRACG models the core using a system of one-dimensional channels. The field equations average the state parameters over the cross section of the channel. Additional constitutive correlations are needed to simulate and accurately represent the more detailed, sub-channel flow variations, including wall-to-fluid interactions and phasic interactions in the fluid.

Such correlations are typically empirically derived using separate effects tests.[Q] With particular regard to TRACG04, the relations have been reviewed and accepted for use in simulating AOOs and the ATWS and ASME overpressure transients. Thus, while the models have been previously reviewed, their extension to post-CHF flow regimes requires consideration within the TRACG-LOCA review. This includes extending void fraction predictions to low pressures associated with LOCA, considering dispersed droplet flow and heat transfer, liquid droplet or film rewet of heated surfaces. The review also necessitates consideration of CCFL and core spray flow distribution.

Regulatory Position 3.9 of RG 1.157 states:

> *Models of heat transfer from the fuel to the surrounding fluid in the post-CHF regimes of transition and film boiling should be best-estimate models based on comparison to applicable steady-state or transient data. Any model should be evaluated to demonstrate that it provides acceptable results over the applicable*

[Q] As will be discussed later in this SE, the Sun-Gonzalez-Tien correlation, used in TRACG for dispersed flow film boiling heat transfer, is theoretically derived.

ranges. Best-estimate models will be considered acceptable provided their technical basis is demonstrate with appropriate data and analyses.

The evaluation presented below establishes that GEH is consistent with Regulatory Position 3.9, but also follows the graded approach suggested by SRP 15.0.2, acknowledging TRACG's prior approval precedent. RAIs associated with this review segment are listed in Table 8, below.

Batch	RAI	Topic	Ref.
2	78	Flow Regime Uncertainties	4
2	98	Disperse Droplet Flow Modeling	4
2	69	Adherence to Correlation Limitations	4
1	45	Sun-Gonzalez-Tien Uncertainty Treatment	3
1	47	Interior Rod Heat Transfer Coefficient Bias	3

Table 8. RAI Responses Related to Post-CHF Phenomena.

6.3.1. Flow Regime Map

The TRACG flow regime map is presented in Chapter 5 of Reference 16. The flow regime map is fairly simple, based primarily on void fraction. Flow regime boundaries are provided to define the transition between bubbly/churn and annular flow. In the higher void regime, the entrainment model is used to assist in identifying the transition between annular flow and dispersed droplet flow. In its review, the NRC staff requested that GEH explain how uncertainty in the flow regime map is addressed in TRACG-LOCA. In the response to RAI 78, the vendor stated that, because the flow regime selection logic is dependent on void fraction, and the void fraction includes an explicit uncertainty treatment, there is no explicit treatment of uncertainty associated with the flow regime selection. The NRC staff also notes that the flow regime is used to transition among heat transfer correlations, and each heat transfer correlation is accompanied by an uncertainty range. In consideration of the supplemental information furnished by GEH, and the NRC staff observation, the NRC staff determined that the vendor addressed the issue regarding flow regime uncertainty acceptably.

6.3.2. Entrainment, Interfacial Shear, and Wall Shear Models

These models, which govern the prediction of phasic interactions, are considered together in the evaluation of TRACG, primarily because their qualification is based on a similar set of data. The models are indirectly qualified using void fraction, pressure drop, and two-phase level data in various geometries, system pressures, hydraulic diameters, and mass fluxes. Model applicability is then confirmed using integral effects tests.

As noted above, the TRACG entrainment model is used to identify the transition from purely annular flow to annular flow with dispersed droplets. It is based on a correlation developed by Mishima and Ishii, and is described in Section 5.1.2 of Reference 16. GEH modified the implementation of the Mishima and Ishii correlation in TRACG in order to provide a more accurate prediction of conditions in which the entrainment fraction approaches unity. The original correlation, its modified form, and two-sigma uncertainty bands of the modified form are shown in Figure 5-3 of Reference 16. The figure also shows the data used to define the correlation, indicating that the modified correlation agrees well with the data. It should be noted that the data represent low pressures, i.e., 0.1 – 0.4 MPa. However, the model is indirectly qualified, alongside the interfacial shear model, using void fraction data that extend from

0.5 MPa to the full BWR operating pressure. The response to RAI 69 indicated that qualification of the entrainment and interfacial shear models at intermediate pressures has been added by comparisons to Toshiba void fraction data.

The development of these TRAC models is described in earlier references, e.g., NUREG/CR-2573, "BWR Refill-Reflood Program Task 4.7 – Model Development Basic Models for the BWR Version of TRAC" (Reference 34). In Reference 34, it was noted that a model for interfacial drag for inverted annular flow had not yet been developed; this capability was added later. At that time, the models were developed based on void fraction data. The original assessment included comparisons to level predictions of FRIGG 36-rod bundle tests, the PSTF 5801-15 level swell test, and the Oak Ridge National Laboratory (ORNL) Thermal Hydraulic Test Facility (THTF) Film Boiling Test, Run 3.06.6B. The THTF test was representative of PWR conditions, and hence used PWR-representative rod bundle geometry and higher pressure conditions. The early assessment showed that the initial TRACB models performed well when assessed against these data sets.

The modern TRACG interfacial shear model is assessed by evaluating how well TRACG can predict void fraction, in Section 3.1 of Reference 17. The separate effects qualification includes the following steady-state experiments:

- A full-size 8x8 BWR fuel bundle from the FRIGG facility
- Single heated tube and adiabatic tests
- Large hydraulic diameter facilities
- Toshiba 16-rod bundle tests at low pressure
- PSTF level swell tests

In each of the steady-state tests listed above, GEH compared TRACG predictions of void fraction to those measured, and concluded that TRACG was capable of predicting the measured void fraction to within **[[ ]]**. The tests used various means to infer void fraction; some used gamma or x ray densitometry, while others used pressure drop.

The applicability of the interfacial shear model for modern, i.e., 10x10 fuel designs was confirmed by comparison to pressure drop measurements from critical heat flux tests (Reference 21, SE Section 3.20.1). It should be noted, however, that this evaluation was limited to pre-CHF conditions, and found acceptable, in part, because that application did not require modeling of post-CHF conditions. Of the tests listed above, only one of the tube tests, from the CISE facility in Italy, produced data that indicated void fractions approaching a value of 1.

The two-phase flow wall friction factors are qualified using rod bundle data from the ATLAS test facility, as described in Section 3.5 of Reference 16. However, Section 3.5 of Reference 16 also notes that the extension of the qualification to LOCA conditions is provided by LOCA tests, such as TLTA and FIST data comparisons, that feature thermal-hydraulic conditions that exceed dryout conditions.

In response to RAI 98, GEH provided additional justification for the applicability of its entrainment, interfacial shear, and wall shear models to high-void, low-pressure conditions (Reference 4). In the RAI response, GEH stated that such conditions are "observed in the late phase of a typical BWR LOCA, where the RPV has been depressurized, fuel heatup is close to the end and ECCS is cooling the bundle from the top of the bundle for non-jet pump plants (BWR/2s) or from both the top and the bottom of the bundle for jet pump BWRs (BWR/3-6s)."

This condition includes counter-current flow at the top of the bundle, and because the bundle power is decaying, CCFL is unlikely. Thus, the liquid downflow rate is calculated by the wall shear and interfacial shear acting on the liquid.

In order to justify the applicability of these models to LOCA conditions, GEH relies on comparisons to tests more specifically tailored to those conditions. Specifically, the response to RAI 98 lists the Core Spray Heat Transfer (CSHT), FIST, TLTA, and Rig of Safety Assessment (ROSA)-III tests as the testing that provides confirmation of adequate inter-working of the entrainment, interfacial shear, and wall shear models, in addition to the post-CHF heat transfer models.

For the uncertainty treatment associated with interfacial shear, **[[**

]] Specifically with regard to LOCA applications, the uncertainty is assessed using the Toshiba data. Figure 5.1-8 of Reference 1 shows the effect on mean void deviation when a multiplier is applied to the interfacial shear parameter. **[[**

]] Since the void deviation is more sensitive to multipliers that are less than one than it is to multipliers that are greater than one, the NRC staff determined that this approach is acceptable. The multiplier is also shown to produce a range of vo`id deviations that are consistent with the **[[**
]] agreement shown in the body of void fraction comparisons.

In its review, the NRC staff requested that GEH provide additional, clarifying information regarding the statistical treatment of uncertainty associated with PIRT Item A3, the lower plenum two-phase level and side entry orifice uncovery time. In the response to RAI 23, GEH stated that this PIRT phenomenon is affected by other physical phenomena that are explicitly addressed in the uncertainty analysis, **[[**
]]

Based on its review, the NRC staff determined that TRACG employs correlations for entrainment, wall shear, and interfacial shear that are specific and appropriate to span the two-phase fluid conditions anticipated in a BWR LOCA. The models are assessed using a variety of data sets that include multiple void conditions, geometries, and system pressures applicable to BWR LOCA conditions. The uncertainty treatment for core interfacial shear is consistent with the broader range of GEH void fraction comparisons, but is based more specifically on comparison to Toshiba void fraction tests at the lower system pressures expected in a LOCA. Finally, reasonable agreement to separate effects and integral LOCA tests indicates that the TRACG two-phase fluid flow models perform acceptably. Based on these considerations, the NRC staff determined that the TRACG interfacial shear, wall shear and entrainment models are acceptable for ECCS evaluation.

6.3.3. <u>Heat Transfer</u>

The heat transfer selection logic for TRACG is discussed in Section 6.6.2 of Reference 16. The response to RAI 69 also includes a succinct table of the heat transfer correlations used by

TRACG, along with the flow conditions that cause the code to invoke each correlation (Reference 4).

In film boiling conditions, the modified Bromley correlation is used for inverted annular, low void, conditions. The correlations are described briefly in Section 6.6.9 of Reference 16. The Bromley correlation has been previously applied in the SAFER evaluation model. As noted in Section 5.1.3.20 of Reference 1, its uncertainty treatment is based on an assessment of quench tests. Since the correlation was approved for use in SAFER, the NRC staff determined that its application in TRACG was acceptable.

The Sun-Gonzalez-Tien (SGT) correlation (Reference 35) is used for dispersed flow, high void conditions, and is described in Section 6.6.10 of Reference 16. SGT has been used in the CORECOOL code to justify the application of spray heat transfer coefficients, rather than to calculate them directly.

The SGT correlation is theoretically derived (Reference 345). As such, it technically has no defining data set. However, it is assessed against both THTF and CSHT separate effects tests,[R] as discussed in Section 3.2 of the TRACG Qualification Report. The detailed assessment includes **[[**

]]

It should be noted that, in order to compensate for the channel-averaging effect discussed in the introduction to Section 6.3 of this SE, TRACG **[[**

]] is defined in Figure 7-21 of Reference 16. This approach was devised based on comparisons of a 4x4 test bundle, modeled using the Dittus-Boelter correlation, using both an extended, sub-channel model and a channel average model. The comparisons showed that the channel average model **[[**
]]. Thus, when simulating the CSHT experiments, GEH used the modeling approach described above. The qualification of SGT against **[[** **]]** CSHT experiments shows similar agreement to the extended, sub-channel model.

In the uncertainty analysis, GEH applies the same uncertainty to SGT as for the Dittus-Boelter correlation, which is used for single-phase convection for both liquid and vapor. In the response to RAI 45, GEH stated **[[**

[R] These tests examined the two-phase heating behavior in rod bundle configurations. In certain contexts, they would more appropriately be considered as integral tests. For example, Chapter 7 of NEDE-33005 refers to them as such. However, in the qualification report, these are located in the section describing separate effects tests, presumably because they are being used to qualify heat transfer correlations that account for multiple heat transfer mechanisms and are applied to rod bundle geometries. Note also that the SGT model "accounts for radiation and convection to the vapor-droplet medium" individually (See Page 2 of the SGT paper).

]] (Reference 3). In the RAI response, GEH also indicated that the demonstration statistical analyses provided in Chapter 7 of Reference 1, wherein the experimental data generally fall within the TRACG-calculated statistical results, provide an indication that the uncertainty range is appropriate.

Single-phase steam cooling is modeled using the Dittus-Boelter correlation, which is modified to account the impact of variations in fluid physical properties due to high temperature difference between the wall and vapor, as stated in the GEH response to RAI 47 (Reference 3). [[

]] Although the uncertainty parameters are derived from 4x4 bundle tests, GEH indicated in the response to RAI 47 that favorable comparisons to full-bundle test data indicate that the correlation performs reasonably well at multiple geometries.

These heat transfer models are qualified using a number of separate effects and integral effects tests. In addition to the THTF and CSHT tests discussed above, GEH used TRACG to simulate three FIST tests, and several ROSA-III tests. The tests generally showed that TRACG predicted the data well. In addition, as discussed in Section 8.1 of this SE, GEH repeated several CSHT, FIST, and SSTF tests using its statistical methods. The results of these analyses – in particular, good agreement regarding the slope of the heatup transient – showed that the TRACG statistical analysis enveloped the experimental data, providing confirmation that the statistical analysis produces reasonable results.

Based on the review summarized above, the NRC staff determined that GEH has adopted realistic models for post-CHF heat transfer for use in TRACG-LOCA. The models have been compared against relevant experimental data, in both steady-state (i.e., CSHT) and transient tests. Thus, GEH is consistent with the guidance in RG 1.157, and on that basis, the NRC staff determined that the TRACG-LOCA post-CHF heat transfer models are acceptable.

6.4. SPRAY COOLING PHENOMENA

The NRC staff review of spray cooling phenomena is described in the following subsections. These phenomena include counter-current flow limitation (CCFL), the distribution of the core spray flow, the minimum temperature for stable film boiling (T_{min}), and the quench front model. RAIs associated with this segment of the review are listed in Table 9, below.

Batch	RAI	Topic	Ref.
1	16	Guide Tube-Bypass CCFL Parameters	3
1	64	CCFL Constant Uncertainty Bands and Data	3
1	20	VSSL Ring 1/Ring 2 PCT Characteristics	3
2	94	Adequacy of Core Spray Flow Distribution Model	4
2	70	LPCS Performance Benchmarking	4
1	50	T_{min} Uncertainty Treatment	3
5	104	Applicability of Rewet Models to ECCS Analysis	7

Table 9. RAI Responses Related to Spray Cooling Phenomena.

6.4.1. CCFL

The Kutateladze correlation is used in TRACG to compute the onset of CCFL. At various places, TRACG checks the flow rates to determine whether CCFL exists. For the core, this includes the upper tie plate and the side entry orifice. The derivation of the CCFL correlation, along with testing data to support its application to the side entry orifice and upper tie plate is provided in Reference 36. Noting the 1975 vintage of this report, GEH also confirms the validity and uncertainty of the correlation for newer fuel designs, and investigates whether other locations in the assembly can exhibit CCFL behavior. Such exercises were undertaken, for example, in the development of Reference 19, and more recently for GNF-2 fuel in Reference 37. Based on the experimental data, specific biases and uncertainties are applied for each individual CCFL location.

[[ ]] In the response to RAI 16, [[ ]] to the one specified in Section 5.1.2.4 of Reference 1. The sensitivity study showed [[ ]] In addition to this sensitivity study, a further verification of the TRACG CCFL modeling capability is provided in the comparison to SSTF test EA3-1, which is discussed in Chapter 7 of the LTR. In particular, Figure 7.4-16 indicated that TRACG generally [[ ]] This comparison is relevant to the guide tube-bypass CCFL behavior, because the behavior affects the ability to establish or retain liquid in the bypass region of the core. Based on the study and the SSTF comparison, the NRC staff concluded that the [[ ]] and determined that the CCFL model used at this location was acceptable.

The response to RAI 64 provided additional clarification regarding how the uncertainty distribution for the CCFL parameters are determined. The response indicated that the uncertainty characterization for the side entry orifice was based on the distribution of differences between the TRACG CCFL constant, and the values measured from the tests. GEH also stated that the upper tie plate CCFL constant is a function of the fuel product line. [[ ]] The RAI response indicates that GEH bases its uncertainty characterization for CCFL on experimental data obtained for the specific geometry of the CCFL location in question.

Since GEH bases its uncertainty treatment for CCFL on empirical data that are representative of the component designs at the CCFL locations of interest, the NRC staff determined that TRACG-LOCA is acceptable with respect to CCFL modeling.

6.4.2. <u>Core Spray Flow Distribution</u>

The modeling of the core spray flow distribution is addressed in Section 5.1.6.4 of Reference 1. For BWR/3-6 applications, [[ ]] This approach provides a conservative lower bound on spray cooling to reach channels in the core. [[ ]] For the BWR/2 plant design, [[ ]], but still conservatively biased low.

The core spray modeling approach is supported by a series of experiments that reflect different plant classes and nozzle designs. The testing was also performed in varying scales using both air-water and steam-water environments. The testing basis is concisely summarized in Reference 38.

In reviewing the analyses based on this approach, the NRC staff observed that [[ ]] in a number of demonstration analyses did not exhibit limiting PCTs. In the response to RAI 20, GEH explained [[ ]]

In the response to RAI 94, [[ ]]

In the response to RAI 70, GEH provided information to address variations in core spray nozzle designs, and testing applicability to various reactor and fuel bundle designs. GEH stated that testing from SSTF, among other facilities, was used to confirm the predictive capability of spray distribution in a steam environment. Specific testing included a range of system pressures and spray flow rates, various nozzle designs, and different steam updraft flow rates. Also, GEH indicated that SSTF tests were initially performed for a BWR/6 design, but later included nozzle and sparger designs representative of BWR/4 and 5 designs as well. Additional analysis and testing of specific BWR/2 nozzle design was also performed in order to confirm the applicability of the methodology to Nine Mile Point Unit 1. GEH also noted that the purpose of these tests was to estimate the amount of spray flow that would reach specific bundles, but that the facility did not include detailed fuel assembly components.

For the BWR/2 application, however, the core spray distribution requires a plant-specific hardware capability analysis. Core spray distribution data are analyzed to determine a minimum value of core spray flow that could reach an interior bundle. This evaluation considers variability in reactor pressure, and in the availability of a core spray topping pump. An example of the core spray flow application is shown in Figure 5.1-23 of Reference 1, [[ ]] As is the case with the BWR/3-6 application, hot bundles in an interior ring of the core receive the flow minimum, and an analogous set of hot bundles in a separate ring receive core spray flow based on upper plenum availability and counter-current flow limitations.

In evaluating the core spray flow distribution, the NRC staff confirmed that the distribution assumptions employed in the BWR/3-6 analyses are conservative by comparing the results of two sensitivity studies employing alternative core spray flow distribution assumptions. The NRC staff also obtained additional information describing the extensive experimental basis supporting GEH's core spray flow modeling, which particularly supports the modeling approach used for the BWR/2. Since the core spray flow distribution is modeled conservatively for BWR/3-6, and since the BWR/2 modeling approach retains conservatism, but is based on numerous core spray flow distribution experiments, the NRC staff determined that the core spray flow distribution modeling for TRACG-LOCA is acceptable.

6.4.3. Rewet and Quench Behavior

At the time that the second-peak heatup begins to slow and transition to cooldown, the PCT location tends to be in a vapor-cooled or dispersed flow condition (Reference 39). Droplets may be in the channel, but are unable to contact the heated rods. Eventually, a quench front can rise, or in the case of spray cooling from above, a liquid film can fall along the surface of the fuel rod. A significant number of parameters influence the point at which disperse droplets in the core may contact the fuel rod, or at which point the liquid film may contact the rod at the point of the PCT location. These include fuel rod material properties, and thermal-hydraulic properties of the coolant in the channel.

The phenomena discussed above are addressed in TRACG by models that calculate the minimum temperature for stable film boiling (T_{min}), and for the axial advancement of a liquid quench front on a fuel rod or channel surface. The TRACG code uses the Shumway correlation (Reference 40) to predict the minimum temperature below which the code will calculate transition boiling, thereby increasing the heat flux from the fuel rod (Reference 41). The quench front model is a simplified correlation that approximates the solution employed in the SAFER method (References 28 and 29). It is described in Section 6.6.13 of Reference 16.

Minimum Temperature for Stable Film Boiling

In the response to RAI 50, GEH provided a clarification that, for LOCA applications, TRACG uses the Shumway correlation to predict T_{min}. The vendor also provided justification for the associated uncertainty treatment in this RAI response.

The Shumway report provides a succinct summary of the challenge associated with predicting the minimum temperature for stable film boiling accurately (Reference 40, Page 10):

> The reason there are so many correlations seems to be that the shape of the boiling curve is influenced by many phenomena which change dominance with experimental conditions. The minimum heat flux or temperature is not exclusively dependent on hydrodynamic or thermodynamic properties of the fluid. It varies with surface conditions such as roughness and surface thermodynamic properties. Factors such as velocity, pressure, subcooling, drop size, liquid contact angle, wetting agents, and gravitational field all influence minimum conditions.

The Shumway correlation includes "some effect of flowrate, pressure, void fraction, fluid properties and wall properties" (Reference 40, Page 39). In its implementation, GEH eliminates the void correction, because it has a tendency to increase the predicted rewet temperature, but as Shumway noted, "the accuracy of the void effect [was] untested" (Reference 40, Page 39; Reference 41, Page 7). GEH also assumes that the fuel rods are comprised of unoxidized zircaloy; formation of an oxide layer has been shown to increase the rewet temperature significantly due to the reduced heat capacity and increased surface roughness of zirconium oxide as compared to unoxidized zircaloy (Reference 41, Page 7). With these modifications, the vendor claims that the implementation of the correlation is conservative, as it predicts a lower value of T_{min} than would be expected for oxidizing fuel rods in a steam environment.

The Shumway correlation is based on data collected for stainless steel. The author included a dimensionless term for wall and fluid material properties based on other, contemporary correlations (e.g., Reference 42); however, Shumway did not provide a validation of the dimensionless term.

In Reference 41, GEH provides an evaluation of the Shumway correlation, comparing it to other, similar correlations and various sources of quench data. Perhaps most notably, GEH provides a comparison of Shumway for various materials, including zirconium, zirconium oxide, and the Groeneveld-Stewart correlation that is implemented in the NRC's TRACE code. The data sources include fresh and oxidized zircaloy, as well as Inconel 600. From Figure 6 of Reference 41, it can be seen that the Shumway correlation trends well with Peterson and Bajorek data (Reference 43) for unoxidized zircaloy, and that the data for oxidized zircaloy fall significantly above the Shumway correlation prediction for unoxidized zircaloy.

These data extend to 0.4 MPa, however, and the correlation is used in demonstration LOCA analyses, in Chapter 8 of the LTR, to predict quenching up to about 3 MPa. While Peterson and Bajorek did not obtain data for unoxidized zircaloy quench behavior above 0.4 MPa, they noted that the predicted rewet temperature becomes asymptotic with increasing pressure (Reference 43). They were also able to show this behavior for stainless and carbon steel, for pressures up to 3 MPa. Other authors corroborate these findings. The Shumway correlation, which GEH illustrated as a function of pressure in Figures 5 and 6 of Reference 41, indicates similar behavior for unoxidized zircaloy.[S] This is expected, because as Peterson and Bajorek note, "Gas properties, such as heat capacity and thermal conductivity, have a strong correlation with pressure at low pressure." In other words, the phenomenology is associated with the coolant properties, rather than the material.

In Reference 1, Section 5.1.3.26, GEH indicates that the correlation is applied with a **[[** **]]** uncertainty value about the calculated difference between T_{min} and T_{sat}. The vendor further notes that the upper end of this uncertainty range approximates the uncertainty applied in SAFER applications.

In the response to RAI 104, GEH documented the results of sensitivity studies that were performed using the BWR/2 large break LOCA event, in which a base case of the EM, using 181 cases, was compared to a run with the same sample, but with the Shumway correlation T_{min} biased to a low value. The BWR/2 event is studied because it exhibits the most severe results, in terms of both degree and duration of cladding heatup. In the biased evaluation, the licensing parameter for PCT **[[** **]]**, while the licensing parameter for ECR **[[** **]]**. The information provided in this study indicates that significant changes to the predicted T_{min} value do not have a significant effect on the overall results, as they may in other applications like anticipated transients without scram (ATWS) with instability, which apply the correlation at higher pressures and potentially over multiple dryout and rewet cycles. Furthermore, the apparent insensitivity of the model to changes in the predicted T_{min} can be attributed to the general behavior of the transient exhibited in both testing and in plant analytic results, in which the PCT is achieved, and the cladding begins a modest cooldown, prior to T_{min} being reached.

[S] This comparison was made by evaluating the data shown in Figure 6 of MFN 13-073, and the data shown in Figure 7 of the Peterson and Bajorek paper. When corrected for units and saturation conditions at the given pressure, it can readily be seen that the Shumway correlation exhibits asymptotic behavior, with the pressure effect leveling off at approximately 1 MPa, just as the data for stainless and carbon steel suggest.

The NRC staff review led to several determinations. First, at the pressures postulated for LOCA conditions, more zirconium-specific data exists to justify the application of the modified Shumway correlation. Second, disabling the void term and neglecting the thermal effect of the zirconium oxide layer that forms during the transient results in a conservative application of the correlation. Finally, the predicted rewet temperature is not a significant contributor to the overall results, as evidenced by sensitivity studies provided in the response to RAI 104. Based on these considerations, the NRC staff determined that the use of the Shumway correlation is acceptable for TRACG-LOCA.

The NRC staff review findings are based on consideration of LOCA-specific phenomenology and sensitivity studies using operating BWRs. Therefore, the conclusions are limited specifically to LOCA applications. This limitation is reflected as Limitation 2.7, "Use of the Shumway Correlation," in Chapter 10 of this SE.

Quench Model

The updated TRACG quench front model is described in Reference 44. The second half of Reference 44 evaluates the quench front model against several LOCA tests, which include test runs from THTF, TLTA, and ROSA-III. These comparisons cover a range of pressures at time of quench from **[[ ]]**, and the tests are simulated using TRACG runs with the quench front model on and off. Because of the extended heatup and precursory cooling associated with LOCA cladding heatup events, the comparison between TRACG runs reveals little difference in PCT. For the run with the greatest difference, for ROSA-III test 912, **[[**

]]

In the response to RAI 104, the vendor provided additional information to justify the TRACG models with respect to application in ECCS evaluation. The response included a review of the information described above, in addition to some additional IETs relevant to BWR/2 (i.e., CSHT tests that reflected falling film quench fronts) and sensitivity studies using a BWR/2, which involves extended heatups and falling film quench behavior. In the BWR/2 transient particularly, these models are significantly more important in predicting the overall sequence of events than in other classes of BWR designs.

The evaluation of the IET comparisons provided by GEH in response to RAI 104 indicated that, although in some cases the peak rod temperatures were slightly under-predicted by TRACG, quench times tended to be delayed relative to test data that included a quench. This tendency results in a longer duration of cladding heatup, preventing a non-conservative under-estimation of the ECR.

A sensitivity study, similar to that performed for the Shumway correlation, was performed for the quench front model. The BWR/2 base case was re-executed, biasing the quench model with a constant, low multiplier. The results showed that the biased quench front model produced significantly higher ECR values, as would be expected. Because the PCT occurs at different elevations, and in different cases, than the maximum ECR, an opposite effect on PCT was observed. The RAI response explained this behavior.

The NRC staff acknowledges that applying a bias to the quench model causes the predicted ECR to increase. However, based on the qualification studies using the LOCA IETs and on the competing influence of the quench model on the BWR/2 PCT and ECR, the NRC staff determined that GEH's quench model was acceptable. The proposed uncertainty treatment for the quench front model includes sampling to the biased value used in the sensitivity study, but also samples other values so that the competing effects shown in the studies are included in the analysis.

7.0 ESTIMATION OF OVERALL CALCULATIONAL UNCERTAINTY

The following section of the evaluation is based on Regulatory Position 4, "Estimation of Overall Calculational Uncertainty," of RG 1.157. It should be noted that RG 1.157 was written contemporarily with CSAU (Reference 11), and the guidance for estimating overall calculational uncertainty was based on a CSAU-like approach, in which uncertainties are estimated through the construction of a response surface, and thousands of perturbations of key uncertainty parameters were analyzed using the response surface. Since publication of RG 1.157, realistic ECCS evaluation models used for domestic licensing purposes have evolved. Appendix B of RG 1.203 recognizes this evolution, and in particular, its Reference A-7 discusses the development of more novel approaches (References 9, 45)

Newer methods, like TRACG-LOCA, rely on direct simulation of a more limited sample of cases. The sample size is determined using order statistics theory. In order to show conformance to the "high probability" language contained in § 50.46(a)(1)(i), upper tolerance limits for each of the critical safety parameters are obtained from the sample. Provided those upper tolerance limits remain below the acceptance criteria contained in § 50.46(b)(1) through (b)(3), the "high probability" requirements are satisfied.

The NRC staff review regarding the estimation of overall calculational uncertainty is discussed in this chapter. This review distinguishes between, and evaluates, the types of uncertainty that GEH chooses to analyze explicitly in the uncertainty analysis, and those types that are addressed via other means. The review also considers the statistical process, and underlying theory, used to evaluate the overall uncertainty and derive the "high probability" results for the critical safety parameters.

Table 10 summarizes the RAIs related to this portion of the review.

Batch	RAI	Topic	Ref.
1	1	General Justification for Uncertainty Treatment	3
2	92	CSHT Qualification Analyses	4
1	66	Sample Size and Joint Upper Tolerance Limits	3
2	89	Deterministic Break Spectrum Analysis	4
5	65	Continuity of Sample Characteristics	7
3	99	Statistical Meaning of Limiting Results	5
5	103	Application of Joint Upper Tolerance Limits	7
1	7	Analysis Resolution	3
1	6	Phenomenological Uncertainty, etc.	3
1	9	BWR/2 Noise-Driven Bifurcation	3

1	4	Core Detail	3

Table 10. RAI Responses Related to Estimation of Overall Uncertainty.

During the review process, GEH made two major modifications to its uncertainty approach. The final approach, which was accepted by the NRC staff, is summarized in the response to RAI 103. This review is primarily based on this ultimate approach.

7.1.   GENERAL

Regulatory Positon 4.1 of RG 1.157 distinguishes between code uncertainty and other sources, including uncertainty associated with the experimental data used in the code assessment process, input boundary and initial conditions, and fuel behavior.

To an extent, the TRACG code assessment accounts for both uncertainty associated with the individual models, and with the experimental data used in the code assessment process, as discussed in Section 5.0 of the LTR. GEH clarified its position on this matter in the response to RAI 1.d, stating "…the measurement uncertainty in the data is intrinsically accounted for when code comparison to experimental results is made."

The effects of scaling the models to BWR plant simulation are discussed in LTR Chapter 5.3. Scalability is demonstrated by comparing TRACG predictive capabilities at various levels, including component-specific tests, scaled integral system tests, and relevant plant data.

The vendor employs simplifying assumptions in a generally conservative fashion. For example, core spray flow variability is addressed in a simplifying assumption **[[**

]] In other cases, GEH provides sensitivity studies to demonstrate that a particular simplifying assumption is conservative. This is demonstrated in the review of the scram time logic discussed in Section 5.5 of this SE. Given the above, it is clear that GEH has evaluated its simplifying assumptions to determine that either the assumption is conservative, or that it has an insignificant effect on the figures of merit.

Many sources of uncertainty with importance relative to the figures of merit are addressed in the deterministic analysis, or are bounded by the use of conservative, simplifying assumptions. Thus, the deterministic break spectrum is already a conservative estimation of plant performance with regard to sources of uncertainty such as initial and boundary conditions, break locations and sizes, single failures, fuel behavior, and the like. When model uncertainty is analyzed for the limiting break of a given spectrum, it is evaluated using properties that bound such sources of uncertainty. Since this uncertainty is bounded, the NRC staff determined that the GEH approach is acceptable.

Regulatory Position 4.1 of RG 1.157 concludes as follows:

> *[…] Regulatory Position 3 provides a description of the features that should be included in the overall code uncertainty evaluation that is called for in paragraph 50.46(a)(1). This uncertainty evaluation should make use of probabilistic and statistical methods to determine the code uncertainty. For a calculation of this complexity, a completely rigorous mathematical treatment is neither practical nor*

> *required. In many cases, approximations and assumptions may be made to make the overall calculational uncertainty evaluation possible. A careful statement of these assumptions and approximations should be made so that the NRC staff may make a judgment as to the validity of the uncertainty evaluation. The purpose of the uncertainty evaluation is to provide assurance that for postulated loss-of-coolant accidents a given plant will not, with a probability of 95% or more, exceed the applicable limits specified in paragraph 50.46(b).*

The uncertainty evaluation furnished by GEH uses statistical methods to evaluate individual sources of model uncertainty. Probabilistic methods are then used to determine upper tolerance limits for each of the critical safety parameters, considering the PIRT-important sources of model uncertainty. In TRACG-LOCA, simplifying assumptions, supported by deterministic analysis and system evaluation, are used to address other important sources of uncertainty, such as for plant inputs and initial conditions. The simplifying assumptions tend to drive the plant operating state to a pessimistic condition, which yields conservative results. When the uncertainty analysis is applied to the conservative plant operating state, the NRC staff agrees that the results show, with probability 95-percent or more, that the acceptance criteria would not be exceeded, given that the results of the uncertainty analysis conform to the acceptance criteria. Thus, the NRC staff determined that GEH is consistent with the guidance in this Regulatory Position.

7.2. <u>CODE UNCERTAINTY</u>

Regulatory Position 4.2 of RG 1.157 states, in part:

> *Code uncertainty should be evaluated through direct data comparison with relevant integral systems and separate-effects experiments at different scales. In this manner, an estimate of the uncertainty attributable to the combined effect of the models and correlations within the code can be obtained for all scales and for different phenomena.*

The vendor addresses code uncertainty by assessing the uncertainty associated with individual models, correlations, and closure relations by evaluating model capabilities against relevant separate effects tests. The testing is described, largely, in Reference 17, and summarized with regard to ECCS/LOCA analysis in Chapter 5.1 of the LTR. The NRC staff evaluation considered the individual uncertainty applied to specific models as part of the review of those models. These considerations are addressed in Chapter 6 of this SE.

> *These comparisons should be performed for important key parameters to demonstrate the overall best-estimate capability of the code… In addition, a code uncertainty evaluation should be performed for other important parameters for the transient of interest to evaluate compensating errors.*

The qualification analysis presented in LTR Section 7.4 addresses this element of RG 1.157 guidance. In addition to making comparisons to peak rod temperatures for FIST large- and small-break LOCA tests, system pressure and temperature comparisons are provided for SSTF tests, and rod temperature comparisons are again provided for CSHT. These comparisons are made using the statistical methods, and they show that the TRACG-predicted results are generally consistent with the test data with no readily discernible bias or nonconservative tendencies.[T]

> *The experimental information used to determine code uncertainty will usually be obtained from facilities that are much smaller than nuclear power reactors. Applicability of these results should be justified for larger scales. The effects of scale can be assessed through comparisons to available large-scale separate effects tests and through comparison to integral tests from various sized facilities. If there are scaling problems, particularly if predictions are nonconservative, the code should be improved for large-scale plants (i.e., nuclear reactors). Codes not having scaling capability will not be acceptable if their predictions are non-conservative.*

The vendor assessed the effects of scale using integral effects tests from several facilities, as discussed in Chapter 5.3 of the LTR. These facilities varied in scale from small (1:624) to SSTF, which is a full-scale mockup of a sector of a BWR/6 core. Favorable comparisons to experimental data at all scaling levels demonstrates the scalability of the evaluation model. Based on its review of LTR Chapter 5.3 and supporting information contained in Reference 17, the NRC staff determined that GEH is consistent with this element of RG 1.157.

7.3. STATISTICAL TREATMENT OF OVERALL CALCULATIONAL UNCERTAINTY

7.3.1. Introduction: Precedential Perspective on the Use of Order Statistics

Initially, GEH proposed to determine the requisite upper quantile by applying a similar approach to that used in TRACG for AOO analysis. In this approach, GEH would select from two possible ways to obtain the desired statistical coverage, which are described in Section 7.1 of the LTR. A sample of 59 statistical trials is analyzed. Based on this sample size, GEH would examine the results for each figure of merit to determine whether the results conform to a normal distribution. If so, the uncertainty would be estimated as 2.024 times the standard deviation of the result.

If the results were not normally distributed, order statistics theory would be applied. Based on non-parametric order statistics, the highest-ranked result in a sample of 59 provides an estimator for the 95/95 one-sided upper tolerance limit with regard to a single figure of merit. This approach was employed in the demonstration analyses provided in Chapter 8 of the LTR.

While this approach was consistent with TRACG for AOO analysis,[U] it was not consistent with the NRC's long-standing practice of accepting order statistics-based ECCS evaluations on the basis that they provide tri-variate upper tolerance limits. That is, the NRC staff has accepted

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[T] It should be noted that, in some cases, the CSHT comparison using the uncertainty methodology, which is presented in LTR Chapter 7.4, indicated that TRACG nominally under-predicted the PCT associated with the particular CSHT run, and with uncertainty analysis included, the TRACG-predicted highest PCT came into the range of the CSHT-observed PCTs. This particular test was chosen because it produced the worst comparison. In response to RAIs 82 and 92, GEH produced additional CSHT test comparisons, including those with higher PCTs, which indicated much better agreement between TRACG and observed test results. GEH also pointed out that the nominal TRACG model in four other CSHT tests over-predicted the PCTs for all rod groups.

[U] Note, also, that this approach is consistent with the TRACG-LOCA application for the ESBWR; however, the ESBWR has a reactor and ECCS design that leads to a much more benign, and less phenomenologically complex, LOCA transient.

evaluations that use order statistics, provided that simultaneous upper tolerance limits are established for each of the critical safety parameters.

The first order statistics-based ECCS evaluation model was approved by the NRC in 2003 (Reference 46). In that method, the NRC staff accepted a univariate tolerance limit based on that vendor's successful demonstration that PCT was the most limiting of the three critical safety parameters. GEH has not attempted to make a similar assertion. Additionally, that method has been revised and now produces trivariate tolerance limits (Reference 47, Page 33). Aside from this example, contemporary, realistic ECCS evaluation models that have been approved by the NRC for operating plants and rely on order statistics produce trivariate, 95/95 tolerance limits for the critical safety parameters. See, for example, Page 7 of the NRC staff safety evaluation approving WCAP-16009-NP-A, "Realistic LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method" (Reference 48).

7.3.2. <u>Summary of GEH Approach</u>

In its response to RAI 103, GEH devised an approach that produces tri-variate, 95/95 upper tolerance limits for each of the critical safety parameters. The approach draws on the theory of Reference 49 to establish two sampling schemes. In the first scheme, Scheme A, GEH will use a sample size intended to provide tri-variate, 95/95 upper tolerance limits. These limits will be approximated by choosing the highest-ranked order statistics for each of the three critical safety parameters, i.e., the highest PCT, and the greatest amount of ECR and core-wide hydrogen generation. In the second approach, Scheme B, GEH will use a sample size that provides the requisite coverage, but allows for the rejection of two cases from within the sample set. For Scheme A, the sample size is 124; for Scheme B, the sample size is 181.

GEH indicated, in its response to RAI 99, and confirmed in its response to RAI 103, that the sampling scheme will be determined at the onset of the analysis. This precludes reducing the overall statistical confidence in the results by increasing sample size *a posteriori*. This concept of reduction in statistical confidence is discussed in further detail in Section 24.11, "What's Wrong With This Picture," of Reference 50.

Scheme B, with two rejected sample constituents, will be selected if GEH or a supported NRC licensee determines that one or more of the critical safety parameters will be challenged in a particular application. The example provided in the RAI response is for BWR/2 applications, wherein the extended transient results in oxidation levels that approach the 17-percent B-J oxidation criterion.

The vendor uses the theory developed in References 50 and 51 to establish the method of elimination in application of Scheme B. The strategy is illustrated in Figure R103-2 of the response to RAI 103. The vendor will determine the tolerance region by eliminating two cases that contribute the highest-ranked statistics for any of the critical safety parameters. The upper tolerance limits will then be chosen, independently, from the remaining 179 cases within the sample. The contributors for each of PCT, ECR, and hydrogen generation may come from different cases, provided each is the highest-ranked from the cases remaining in the sample.

7.3.3. <u>NRC Staff Evaluation of GEH Approach</u>

The NRC staff evaluation for the overall combination of uncertainty is based on Regulatory Position 4.4 of RG 1.157.

Regulatory Position 4.4 of RG 1.157 states as follows:

> *The methodology used to obtain an estimate of the overall calculational uncertainty at the 95% probability limit should be provided and justified. If linear independence is assumed, suitable justification should be provided. The influence of the individual parameters on code uncertainty should be examined by making comparisons to relevant experimental data. Justification should be provided for the assumed distribution of the parameter and the range considered.*

For the statistical analysis, GEH devises a sample of sufficient size to determine 95/95 upper tolerance limits for each of the three critical safety parameters. The theory used in determining the sample sizes assumes that the distributions for each of the critical safety parameters is dependent, without any underlying assumption as to their correlation. This assumption results in reliance on a larger sample size from which to obtain the upper tolerance limits (References 49 through 53), and hence the NRC staff determined that the approach is acceptable.[V]

> *In reality, the true statistical distribution for the key parameters (e.g., peak cladding temperature) is unknown. The choice of a statistical distribution should be verified using applicable engineering data and information. The statistical parameters appropriate for that distribution should be estimated using available data and results of engineering analyses. Supporting documentation should be provided for this selection process. These estimated values are assumed to be the true values of the statistical parameters of the distribution. With these assumptions, an upper one-sided probability limit can be calculated at the 95% level…*

Since GEH revised its approach to remove the ability to use normally-distributed, one-sided upper tolerance limits, and instead exclusively uses non-parametric statistics to determine the upper tolerance limits for the critical safety parameters, this element of guidance does not apply.

> *… As the probability limit approaches 2200 °F, more care must be taken in the selection and justification of the statistical distribution and in the estimation of its statistical parameters. If a normal distribution is selected and justified, the probability limit can be conservatively calculated using two standard deviations. The added conservatism of the two standard deviations compared to the 95th percentile is used to account for the uncertainty in the probability distribution…*

Certainly, the guidance provided above is relevant if an approach like that envisioned in CSAU is used, wherein a response surface is constructed, and thousands of statistical trials are simulated using the response surface. The upper quantile of such a large sample size can be inferred using far less than two times the standard deviation. However, contemporary approaches have abandoned the response surface approach in lieu of performing direct simulation. As is the case with TRACG-LOCA, the approach relies on distribution-free statistics, and the upper tolerance limits are set by the worst-case results from the sample, rather than by adding a multiple of the standard deviation to the mean.

[V] The general idea being, that under-estimation of the sample size leads to excessive variability in the estimation of the tolerance region, and as such, can lead to its under-prediction. This would be non-conservative.

… Other techniques that account for the uncertainty in a more detailed manner may be used. These techniques may require the use of confidence levels, which are not required by the above approach.

Ultimately, this element of the RG 1.157 Regulatory Position drove the successive revisions to the GEH sampling approach.

GEH initially proposed characterizing its results at a quantified upper tolerance level. The response to RAI 66 documented a shift in the methodology. The NRC staff acknowledges that the responses to RAIs 66 and 89 indicated that GEH's initially proposed approach was reproducible, with statistically insignificant differences among different samples applied to the same transient. In the response to RAI 66, GEH further studied the issue by running a 124-case sample, and comparing limiting results from the first 59 cases to the entire population. The difference in sample sizes did not yield statistically different results. However, this information was only provided using a single, two-part sample, and applied to a single plant analysis. It was not possible to infer that similar results should be more generally expected. The response to RAI 89 included a statistical analysis of several different breaks, and, in the region where the break sizes produced similar results because of overlap in the randomly sampled discharge coefficient, the limiting results, again, were not statistically different.

Drawing on the published work of Guba, Makai, and Pal (Reference 49), one might infer that a random sample of size 124 is required to provide a 95/95 joint upper tolerance region for a trivariate set of attributes (PCT, MLO, CWO). The tolerance region would be defined by the worst-ranked result for each figure of merit. The NRC staff issued RAI 66, in part, to address GEH's inconsistency with this approach. In response, however, the vendor merely eliminated its statement of confidence in the results. The topic was revisited in RAI 99, and in response, GEH proposed instead to use three independent, random samples of size 59 to evaluate each of three limiting break configurations. The NRC staff also did not accept this approach, because in the demonstration analyses, the deterministic break spectra for all product lines appear to show a conclusively limiting break size, meaning that the information obtained from analyzing the other break sizes does not contribute meaningfully to the overall limiting results.

In order to address the NRC staff concerns, GEH ultimately agreed to perform the two-scheme statistical analysis using a sample of either 124 cases with zero rejections, or 181 cases with two rejections, and estimating the upper tolerance limits by selecting the highest remaining order statistic for each of the three critical safety parameters. The NRC staff accepts this approach because it provides tri-variate, 95/95 upper tolerance limits for each of the critical safety parameters.

The evaluation of the peak cladding temperature at the 95% probability level need only be performed for the worst-case break identified by the break spectrum analysis in order to demonstrate conformance with paragraph 50.46(b). However, in order to use this approach, justification must be provided that demonstrates that the overall calculational uncertainty for the worst case bounds the uncertainty for other breaks within the spectrum. It may be necessary to perform separate uncertainty evaluations for large- and small-break loss-of-coolant accidents because of the substantial difference in system thermal-hydraulic behavior.

The vendor conforms to this element of guidance by evaluating a plant-specific break spectrum to identify the limiting break characteristics, and performing distinct uncertainty analyses for the limiting breaks. If the spectrum does not conclusively identify a single limiting break, then the multiple, possibly limiting breaks will be treated as sensitivity studies, meaning the statistical analysis is repeated for the different, possibly limiting, break characteristics.

> *The revised paragraph 50.46(a)(1)(i) requires that it be shown with a high probability that none of the criteria of paragraph 50.46(b) will be exceeded, and is not limited to the peak cladding temperature criterion. However, since the other criteria are strongly dependent on peak cladding temperature, explicit consideration of the probability of exceeding the other criteria may not be required if it can be demonstrated that meeting the temperature criterion at the 95% probability level ensures with equal or greater probability that the other criteria will not be exceeded.*

GEH considers each of PCT, ECR, and core-wide oxidation explicitly.

Based on the review and evaluation described above, the NRC staff determined that the TRACG-LOCA approach for estimating the overall uncertainty associated with the analytic results, incorporated by expressing the upper tolerance limits associated with such results, is acceptable. Several limitations to the approach apply, as set forth in Chapter 10.

7.4. PCT ANALYSIS RESOLUTION AND CORE DETAIL

The concept of PCT analysis resolution is introduced in Chapter 6.4 of the LTR. It characterizes "an inherent uncertainty that is common to analyses using system codes, which is not separately identified by physical phenomena" (LTR, Page 6-28). Small changes in the input cause variability in the results, and this effect is sometimes called computational noise. The analysis resolution quantifies this uncertainty. It can be estimated by running statistical analyses that perturb parameters by small fractions of their ordinary statistical distributions, and examining the spread of the results.

GEH attributed the magnitude of the analysis resolution to two key contributors: numerical techniques and channel plugging, or the parallel channel effect. The channel plugging effect arises because subtle changes in lower plenum behavior, or in bypass behavior, can cause the orifices at the channel bottom to either remain open with a vapor flow path, or to 'plug' with liquid. Whether or not the plugging occurs can be affected by minute changes in other system parameters, but the effect can drastically change the behavior of the hot channel heatup. Effectively, the plugging phenomenon can cause a bifurcation in PCT results over a statistical trial, with one grouping of curves reflecting the PCT transient of a plugged hot channel, and the other reflecting the transient of an unplugged hot channel. The plugged channel does not benefit from the vapor cooling associated with the steam updraft that occurs in a channel with a more open flow path at the bottom.

In its review, the NRC staff did not accept GEH's position that channel plugging should be considered a dominant contributor to the analysis resolution without further evaluation. Rather, the NRC staff determined that the plugging behavior, and resulting PCT bifurcation, suggested that the core model did not contain sufficient detail as to permit a realistic representation of the effects of the channel plugging phenomenon. The staff position was documented in RAI 7, among others including RAIs 6, 9 and 4.

As noted in the response to RAI 4, GEH implemented a TRACG core model with significantly greater detail, which minimized the effects of non-phenomenological variability on the code output. The responses to RAIs 6, 7, and 9 demonstrated that the non-phenomenological variability was significantly reduced, once the detailed core model was implemented.

[[

]] Based on these results, the NRC staff determined that the improved core detail adequately addressed the stochastic behavior associated with channel plugging.

In addition to the studies described above, GEH [[ ]] its small perturbation analysis to sample, [[ ]], rather than the originally proposed [[ ]]. These changes significantly reduced the PCT bifurcation associated with channel plugging, and provided a significantly reduced estimate of the analysis resolution. Additional studies using a [[ ]] provided results that were consistent with the [[ ]], suggesting that [[ ]] was a sufficient constraint upon which to determine the analysis resolution. As noted in the response to RAI 7.d, the use of [[ ]] also reduced the analysis resolution to a value that was "much less than the total uncertainty used in determining the upper tolerance limits" (MFN 14-064 (Reference 3), Page 27).

Based on the review described above, the NRC staff determined that GEH devised an acceptable means to estimate the numerical resolution associated with its analysis. The method relies on a model that uses increased core detail in order to reduce the heightened model sensitivity to the parallel channel effect, and the use of a small parameter perturbation analysis. Using the improved model, the vendor demonstrated that the analysis resolution was on the order of [[ ]].

8.0 EVALUATION MODEL QUALIFICATION AND DEMONSTRATION

8.1. STATISTICAL ANALYSIS FOR QUALIFICATION EVENTS

In Section 7.4 of Reference 1, GEH provided a series of statistical analyses for qualification events, with the stated purpose of validating the values used for the model uncertainties by showing that the test data fall within the resulting uncertainty band of the calculations. The vendor selected the large and small breaks from the FIST series, specifically, Tests 6DBA1B and 6SB2C. GEH also exercised its statistical analysis for two Core Spray Heat Transfer tests, as well as for Steam Sector Test Facility Test EA3-1.

Comparisons of pressures and rod temperatures to the FIST tests indicated excellent agreement, with most data points falling within the minimum and maximum results from the TRACG analyses. In particular, the limiting rod temperature transients were generally over-predicted, with maximum cladding temperature transients predicted by TRACG that tended to extend for longer than those exhibited by the tests.

GEH provided comparisons of system pressure, temperatures at the pool periphery, and fill fractions for the lower plenum, bypass, and upper plenum for the SSTF comparison. The

system pressure trend shows that the test data generally fall within the bounds of the calculation. The pool temperature predictions follow general trending of the data, but GEH explained that the TRACG predictions, like the test data, fluctuated once the lower plenum filled. The mass fraction in the lower plenum showed that the lower plenum refill time fell within the bounds of the TRACG calculations. The trending for the bypass and upper plenum fill fractions suggests an element of conservatism with regard to the TRACG predictive capabilities for upper plenum mixing and CCFL; [[
]]

In selecting the CSHT tests to simulate, GEH chose those tests where the TRACG nominal analysis tended to under-predict the rod temperature trends. These tests were Tests 111 and 112. The application of the statistical methods to these CSHT tests brought the TRACG limiting predicted peak temperatures either very close to, or within the uncertainty bounds of the experimentally measured temperatures. In the responses to RAIs 92 and 104, GEH explained that TRACG's nominal models tended to over-predict the PCTs for four other CSHT tests, and that increasing the sample sizes associated with these simulations to 124 (instead of 59), would add to the spread in temperature distribution associated with these tests.

8.2. <u>REVIEW OF STUDIES FOR BWR/2 PLANT</u>

The NRC staff reviewed the BWR/2 demonstration analysis in detail, in particular because the BWR/2 results have both high PCT and ECR values. The NRC staff review generated numerous RAIs that were specific to modeling approaches and phenomena of particular import to the BWR/2 analysis, as summarized in Table 11, below.

Batch	RAI	Topic	Ref.
1	13	BWR/2 Air Ingress and Break Location	3
1	38	Air and Two-Phase CCFL in BWR/2	3
1	59	PIRT Item F3 – Non-Condensibles	3
1	29	BWR/2 Axial Peaking	3
1	35	Significance of BWR/2 Doppler/Void Coefficients	3
1	32	BWR/2 Isolation Condenser	3
1	62	BWR/2 Isolation Condenser	3
2	96	Uncertainty Estimate for BWR/2 Results	4

Table 11. RAI Responses Related to BWR/2 Analyses.

The BWR/2 break spectrum is shown in Figure 8.3-23 of Reference 1. For the BWR/2, the limiting PCTs are produced by the larger breaks. In particular, the limiting PCT is produced by a double-ended guillotine break on the recirculation discharge. The results of the uncertainty analysis for this break are shown in Figures 8.3-33 and 8.3-34. [[
]]

The spectrum analysis [[

]] This behavior is the general subject of RAI 13, the focus of which is the effect that air ingress has on the PCT response.

In RAI 13.a, the NRC staff requested that GEH explain why opposite impacts from break geometry are obtained between the suction and discharge spectra. [[

]]

The studies that GEH performed confirmed that the limiting break geometry in the BWR/2 spectrum is [[

]]

Since the vendor provided a reasonable, phenomenological explanation for the break spectrum behavior, supported its findings with sensitivity studies, and ensures that model sensitivities attributable to break configuration are bounded by conservative modeling, the NRC staff accepted the vendor's explanation of the behavior. In the remainder of the response to RAI 13, GEH discussed additional sensitivity studies to evaluate the effect of varying amounts of air ingress due to different break locations, sizes, and geometries. These studies confirmed that, in consideration of varying levels of air infiltration for different break locations, sizes, and geometries, that the [[ ]] remained the limiting break in terms of PCT.

The studies described above provided information to inform GEH's response to RAI 38, which is evaluated in Section 3.3.1 of this SE. The NRC staff requested GEH reconsider its assignment of a medium rank for PIRT items M3, and in response, GEH indicated [[ ]] PCTs shown in the response to RAI 13, and in the larger breaks of the BWR/2 spectrum.

The NRC staff review of the effect of noncondensibles continued with RAI 59, in which the NRC staff requested that GEH provide normality test descriptive statistics for the uncertainty treatment associated with PIRT Item F3, "Noncondensible Return at Low Pressure." In its response, GEH indicated that the phenomenon itself is addressed [[

]] such that it is not a significant factor in PCT performance in BWR/2 accident scenarios.

The BWR/2 analyses presented in the LTR did not include a description of sensitivity studies to identify limiting axial power shapes. By comparison, the BWR/4 analysis provides this information, which ultimately supports the conclusion that an appropriately limiting axial power shape was modeled. This information, for the BWR/2 analysis, was provided in the response to RAI 29. To study the model sensitivity to axial peaking factor, GEH performed a two-step analysis. The first step evaluated the PCT and ECR for the different hot channels in the reference core. In the second step, GEH varied the elevation of the peak node in full-range uncertainty analyses, and then performed mean-to-mean comparisons among the different axial shapes. From this study, GEH concluded **[[**

]] This conclusion is supported by the results of the sensitivity studies, **[[**

]] The vendor further observed **[[**
]] The NRC staff accepted GEH's analysis, since it indicated that the analytic approach identified a reasonably limiting axial power shape; however, tables included for initial conditions for both the BWR/4 analysis and the BWR/2 analysis indicate that **[[**
]] While the NRC staff agrees, based on the studies provided for both BWR/4 and BWR/2, **[[**
]] sensitivity studies will be performed for each TRACG-LOCA application to provide the basis for a limiting initial condition.

In its review of the limiting transient for the BWR/2, the NRC staff observed that the boiling transition peak for the limiting channel is not fully quenched. In RAI 35, the NRC staff requested that GEH justify that parameters affecting the boiling transition temperature increase, such as void and Doppler coefficient, do not appreciably influence the PCT. In its response, GEH stated that the prompt voiding in the core due to rapid pressure reduction associated with a large break is sufficient to reduce fission power to almost zero even before the scram becomes effective. The NRC staff agrees with GEH's account of the large break event and concluded that the concerns discussed in the RAI are addressed.

The BWR/2 design includes an isolation condenser. The NRC staff requested additional information relative to modeling, or not modeling, the isolation condenser relative to the single failure analysis, and relative to the relevant model uncertainty. In the response to RAI 32, GEH stated that any isolation condenser single failures are bounded by never crediting the isolation condenser. GEH also explained the logic used to determine the limiting single failure, which relates to failure of other hardware, including diesel generators and the core spray system. In the response to RAI 62, GEH reiterated that the isolation condensers are not credited. Not crediting the isolation condenser in the break spectrum analysis produces a conservative effect in the range of small- and intermediate-break events, because the system provides additional cooling inventory, and helps to reduce the RCS pressure, thus enhancing the core spray effectiveness. GEH further committed to validate the isolation condenser heat transfer uncertainty treatment on a plant-specific basis. Based on the RAI responses, and in consideration of the fact that the large-break event is limiting for the BWR/2 plant design, the NRC staff did not review the models pertinent to isolation condenser heat transfer, and use of these models would require NRC review prior to implementation in a plant-specific analysis.

Finally, in its review of the BWR/2 statistical analysis, the NRC staff observed that the PCT distribution for the [[

]] and questioned the accuracy of the uncertainty assessment. [[

]] Since the NRC staff is basing its approval on GEH's agreement to use order statistics, and to use a larger sample size than employed in the demonstration analysis, the NRC staff confirmed that the questionable parametric statistics associated with the small break will be addressed by the switch to order statistic. Since the small break is non-limiting, the NRC staff concluded that the concern conveyed in the RAI was addressed.

Based on the NRC staff review of the BWR/2 analysis, the NRC staff determined that specific considerations and unique considerations for this plant design are adequately addressed in the TRACG-LOCA methodology.

9.0 LICENSING CONSIDERATIONS

GEH describes the methodology application in Chapter 9 of the LTR. The chapter summarizes the processes for initial TRACG-LOCA plant analysis, and for evaluating changes to the plant configuration, the analytic method, or the introduction of new fuel. The chapter also addresses limitations to other GEH LTRs, such as the Interim Methods Licensing Topical Report (IMLTR) and MELLLA+.

9.1. TRACG-LOCA APPLICATION

TRACG-LOCA is being incorporated into the GESTAR-II process via Amendment 37. As such, licensees using GESTAR-II may administratively implement TRACG-LOCA, provided that such implementation is consistent with the limitations delineated in this SE, and that implementing licensees determine that TRACG-LOCA is applicable to the facility design. It should be noted that reporting requirements of 10 CFR 50.46(a)(3) apply to TRACG-LOCA implementations.

From time to time, NRC licensees may seek license amendments that are based, at least in part, on evaluations of ECCS performance using TRACG-LOCA. For example, Section 9.3 of this SE reviews power uprates and operating domain expansions. Other amendments may seek changes to equipment out-of-service (EOOS) options. When such license amendments are submitted, the NRC staff expects that the requesting licensees will provide a summary of the ECCS performance evaluation on a level of detail consistent with the demonstration analyses discussed in LTR Chapter 8.

9.2. EVALUATING CHANGES

The LTR includes several sections on evaluating changes, which discuss the way that GEH may use TRACG-LOCA to assess items like code changes, plant configuration changes, and new fuel introductions.

Such changes are typically evaluated, and if the change is determined to have an effect on the predicted PCT, then the change must be treated in accordance with 10 CFR 50.46(a)(3) requirements. However, the regulations in 10 CFR 50.46(a)(3) simply require licensees to estimate the effect of each change to, or error in, an acceptable evaluation model, or in its application. The estimated effect then must be reported to the NRC. The regulations are not specific with regard to how the effect of such a change or error is estimated. As such, the NRC did not make any regulatory determinations with regard to the process used to evaluate changes.

9.3. CONCURRENT GEH LTRS AND RELATED LICENSING

In Section 9.9 of the LTR, GEH discusses compliance with the requirements of IMLTR and the MELLLA+ LTR. While the main purposes of TRACG-LOCA are to demonstrate compliance with 10 CFR 50.46 requirements and to support corresponding core operating limits, TRACG-LOCA analyses may also be used to support licensing requests, such as those associated with expanded operating domains like Maximum Extended Load Line Limit Analysis Plus (MELLLA+) and extended power uprate (EPU). Although GEH chose to address IMLTR and MELLLA+ in this respect, the NRC staff review considered a broader array of GEH LTRs.

This section of the SE discusses several licensing topical reports whose considerations regarding ECCS performance were based on the SAFER/GESTR-LOCA analytic method (References 28, 29). The SAFER/GESTR-LOCA method included several elements that influenced the wording contained in the passages that are excerpted below.

- [[

  ]]
- [[

  ]]
- Results: SAFER/GESTR-LOCA produced several different PCT results. These included a nominal PCT, in addition to: (1) an Appendix K PCT result, which was obtained using Appendix K-conformant model features, (2) an upper bound PCT, which was determined by adding an allowance for uncertainty to the nominal PCT, and (3) a licensing basis PCT, which was determined by including additional margin to the nominal PCT to account for differences between the nominal and the Appendix K PCT, and to account for plant-specific uncertainties not addressed specifically within Appendix K requirements.

9.3.1. Extended Power Uprates

GEH maintains a suite of licensing topical reports that are referenced by licensees in requests to implement extended power uprates (EPUs). These include:

- NEDC-32424P-A, "Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprates," commonly referred to as EPU Licensing Topical Report (ELTR) 1 (Reference 54).

- NEDE-32525P-A, "Generic Evaluations for General Electric Boiling Water Reactor Extended Power Uprates," commonly referred to as ELTR 2 (Reference 55).

- NEDE-33004P-A, "Licensing Topical Report: Constant Pressure Power Uprate," commonly referred to as the Constant Pressure Power Uprate (CPPU) LTR, or CLTR (Reference 56).

<u>ELTR1</u>

The NRC staff safety evaluation approving ELTR 1 noted the following with regard to ECCS performance:

(a) Reanalysis of the LOCA response utilizing the SAFER/GESTR computer model will improve the apparent safety margins, especially the margin between the calculated peak cladding temperature and the 2200-degree Fahrenheit limit contained in 10 CFR 50.46. ELTR 1 implies that this margin can be used in the relaxation of inputs to the ECCS codes, which corresponds to changes in operational requirements of ECCS equipment. It is the staff's position that this "additional" margin must be used judiciously. Therefore, the staff will not support changes that would relax equipment requirements, such as emergency diesel generator start times, pump flow requirements, and so on, as part of an amendment request for extended power uprate. The staff suggests that the relaxation of these parameters might be pursued as a BWR Owners Group initiative, but in any case must be treated separately from the generic BWR power uprate program.

(b) If a licensee updates a plant from a previous ECCS computer code to the SAFER/GESTR code, a baseline run using SAFER/GESTR at the present power level must be included so that the true effect of the power uprate can be assessed. Inclusion of SAFER/GESTR results obtained using relaxed input parameters (as previously discussed) may be included in the power uprate amendment request, but must be accompanied by corresponding results obtained by using the present input parameters and uprated power.

Appendix D to ELTR 1 provides the basis for EPU ECCS evaluations performed in accordance with this LTR. While the content of Appendix D is generally based on the use of SAFER methods, it is more broadly constrained by a requirement to use an NRC-approved evaluation model, and a requirement to evaluate the entire break spectrum and limiting single failure for uprated operating conditions. A TRACG-LOCA-based ECCS evaluation would be considered an acceptable evaluation model for such an analysis, on a plant-specific basis.

<u>ELTR2</u>

The NRC staff safety evaluation of ELTR2 states the following with regard to design basis accidents:

> Plant-specific analyses will continue to demonstrate the ability of each plant to cope with the full spectrum of hypothetical pipe break sizes from breaks as small as instrument lines to breaks in the largest recirculation, steam, feedwater and ECCS lines… Challenges to the fuel and containment, as well as potential radiological releases to the environment, will be assessed on a plant-specific basis using NRC-approved methods.

The evaluations presented in ELTR2, including those discussed in ELTR2, Supplement 1, Volume 1, conclude that the ECCS equipment performs acceptably relative to EPU operation, given acceptable results from an NRC-approved ECCS evaluation model like SAFER/GESTR-

LOCA. In performing plant-specific evaluations, TRACG-LOCA-based analyses would be considered similarly acceptable.

CLTR

The CLTR builds on the guidelines and evaluations provided in ELTR 1 and 2, and on the experience from EPU applications that had been reviewed and approved between the approval of ELTR 1 and 2 and the approval of the CLTR in 2003. Based on this experience, the NRC staff approved a limited set of analyses that would be used to support conclusions about ECCS performance at EPU plants.

Key considerations included that (1) the implementation of an EPU tended not to affect the LOCA break spectrum and limiting single failure, and (2) the implementation of an EPU tended to increase the predicted PCT on the order of 20 °F or less. Based on these considerations, the NRC staff approved a disposition requiring the evaluation of the limiting break and a sub-set of smaller break sizes for the ECCS performance evaluation at uprated conditions.

The CLTR is also limited, to some extent, to GE14 fuel designs. The NRC understands that some recent NRC licensees using more advanced fuel designs have used a hybrid of CLTR and ELTR1/ELTR2 dispositions. The TRACG-LOCA evaluation model specifically delineates its applicability to the EPU operating domains, such that an EPU application could be supported by a TRACG-LOCA break spectrum ECCS evaluation. However, it should be noted that the generic disposition discussed above is based on (1) SAFER/GESTR-LOCA analyses, and (2) a conclusion that the limiting break does not change for EPU operation.

Years of NRC staff experience reviewing BWR EPUs has revealed that, in fact, an EPU can change a plant-specific break spectrum, making a smaller break event more limiting. In addition, the difference in analytic assumptions employed between SAFER/GESTR-LOCA and TRACG-LOCA can, itself, introduce changes in a plant-specific break spectrum. Therefore, the NRC staff concluded that the ECCS-LOCA analytic dispositions set forth in the CLTR should not be used in justifying a plant-specific BWR EPU when the plant licensing basis method is TRACG-LOCA. Rather, the analytic summary discussed in Section 9.1 of this SE should be furnished with the EPU application. This determination is reasonably consistent with the practice discussed regarding Interim Methods and MELLLA+, in the following sections of this SE, and is reflected in Limitation 6, discussed in Chapter 10 of this SE.

9.3.2. Interim Methods

Some plants incorporate NEDC-33173P, “Applicability of GE Methods to Expanded Operating Domains,” or the so-called IMLTR in the licensing bases (Reference 57). The IMLTR imposes limitations based on the database against which GEH methods are qualified. Several of the limitations, identified below, warrant additional discussion within the framework of TRACG-LOCA.

ECCS Evaluations

For applications requesting implementation of EPU or expanded operating domains, including MELLLA+, the small and large break ECCS-LOCA analyses will include top-peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

The ECCS-LOCA will be performed for all statepoints in the upper boundary of the expanded operating domain, including the minimum core flow statepoints, the transition statepoint as defined in Reference 2 and the 55 percent core flow statepoint. The plant-specific application will report the limiting ECCS-LOCA results as well as the rated power and flow results. The SRLR will include both the limiting statepoint ECCS-LOCA results and the rated conditions ECCS-LOCA results.

This limitation is based on the SAFER/GESTR-LOCA analytic methods, reflecting both the SAFER/GESTR-LOCA power shapes and the licensing basis and upper bound PCTs. Plant-specific TRACG-LOCA analysis will evaluate all necessary statepoints in the power-to-flow operating domain. In addition, the method considers both bottom-/mid- and top-peaked power shapes. Plant-specific analyses also include a spectrum of break sizes to identify limiting break sizes for each of a small-, intermediate- and large-break. Once the uncertainty analysis is complete, TRACG provides upper quantile results for the limiting initial condition, such that there is no longer a distinction between licensing basis and upper bound PCT. Noting these differences, a TRACG-LOCA analysis can be used to satisfy this limitation. As discussed in Section 9.1 of this SE, the submittal of a TRACG-LOCA analytic summary will permit sufficient NRC staff review to determine whether this IMLTR limitation is satisfied. This review determination is reflected in Limitation 6, in Chapter 10 of this SE.

9.3.3. Maximum Extended Load Line Limit Analysis Plus

Several limitations related to concurrent changes, set forth in Section 12.4 of the staff SE approving MELLLA+, require ECCS performance evaluation (Reference 58). TRACG-LOCA is an acceptable method for performing such evaluations, provided that the conditions being evaluated remain within the TRACG-LOCA qualification range.

Several limitations are specific to ECCS-LOCA evaluations, specifically those discussed in Sections 12.10 through 12.14 of the staff SE approving MELLLA+. These include limitations related to the ECCS-LOCA off-rated multiplier (Section 12.10), the axial power distributions required for analysis in the MELLLA+ operating domain (Section 12.11), reporting of ECCS performance evaluation results (Section 12.12), requirements for small break LOCA analysis (Section 12.13), and requirements for break spectrum analysis (Section 12.14).

ECCS-LOCA Off-Rated Multiplier Limitations

The following indented paragraphs refer to applications of the SAFER/GESTR-LOCA methods:

> The plant-specific application will provide the 10 CFR Part 50, Appendix K, and the nominal PCTs calculated at the rated EPU power/rated CF, rated EPU power/minimum CF, and the low-flow MELLLA+ boundary (Transition Statepoint). For the limiting statepoint, both the upper bound and the licensing PCT will be reported. The M+SAR will justify why the transition statepoint ECCS-LOCA response bounds the 55 percent CF statepoint. The M+SAR will provide discussion on what power/flow combination scoping calculations were performed to identify the limiting statepoints in terms of DBA-LOCA PCT response for the operation within the MELLLA+ boundary. The M+ SAR will justify that the upper bound and licensing basis PCT provided is in fact the limiting PCT considering uncertainty applications to the non-limiting statepoints.

LOCA analysis is not performed on cycle-specific basis; therefore, the thermal limits applied in the M+SAR LOCA analysis for the 55 percent CF MELLLA+ statepoint and/or the transition statepoint must be either bounding or consistent with cycle-specific off-rated limits. The COLR and the SRLR will contain confirmation that the off-rated limits assumed in the ECCS-LOCA analyses bound the cycle-specific off-rated limits calculated for the MELLLA+ operation. Every future cycle reload shall confirm that the cycle-specific off-rated thermal limits applied at the 55 percent CF and/or the transition statepoints are consistent with those assumed in the plant-specific ECCS-LOCA analyses.

Off-rated limits will not be applied to the minimum CF statepoint.

If credit is taken for these off-rated limits, the plant will be required to apply these limits during core monitoring.

Each statepoint identified above is analyzed in the TRACG-LOCA analysis for MELLLA+ plants. However, TRACG-LOCA does not produce Appendix K, licensing basis, or upper bound results. Instead, the TRACG-LOCA analysis will produce nominal results from the system analysis, and upper quantile results from applying the uncertainty analysis at each state point. Noting these differences, the inclusion of a detailed description of the TRACG-LOCA analysis and its results will provide sufficient information for the staff reviewing a plant-specific MELLLA+ application to determine whether this condition is satisfied. This conclusion is reflected in Limitation 6, discussed in Chapter 10 of this SE.

Axial Power Distribution

For MELLLA+ applications, the small and large break ECCS-LOCA analyses will include top peaked and mid-peaked power shape in establishing the MAPLHGR and determining the PCT. This limitation is applicable to both the licensing bases PCT and the upper bound PCT. The plant-specific applications will report the limiting small and large break licensing basis and upper bound PCTs.

This limitation was written with regard to elements of the SAFER/GESTR-LOCA methodology, which relied on a mid-peaked axial power shape, and produced a number of different kinds of results, including licensing basis, Appendix K, and upper limit PCTs. The TRACG-LOCA methodology includes various power shapes, and evaluates a spectrum of break sizes to determine limiting small-, intermediate- and large-break results. Thus, in the application of the TRACG-LOCA methodology, a specific licensee can analyze various power shapes and determine nominal and upper quantile PCTs for the limiting break sizes. This information will satisfy the MELLLA+ limitation regarding axial power distribution.

ECCS Reporting

For SAFER/GESTR-LOCA applications:

Both the nominal and Appendix K PCTs should be reported for all of the calculated statepoints, and

The plant-variable and uncertainties currently applied will be used, unless the NRC staff specifically approves a different plant variable uncertainty method for application to the non-rated statepoints.

When relying on TRACG-LOCA results, NRC licensees will instead report nominal and upper quantile PCT results. Uncertainties and plant variables will be determined in accordance with the methodology.

Small Break LOCA Analysis

For SAFER/GESTR-LOCA applications:

> Small break LOCA analysis will be performed at the MELLLA+ minimum CF and the transition statepoints for those plants that: (1) are small break LOCA limited based on small break LOCA analysis performed at the rated EPU conditions; or (2) have margins of less than or equal to **[[ ]]** relative to the Appendix K or the licensing basis PCT.

Since TRACG-LOCA analyses require consideration of the entire spectrum of break sizes, this limitation is inherently satisfied by any application of the TRACG-LOCA methodology.

LOCA Break Spectrum

> The scope of small break LOCA analysis for MELLLA+ operation relies upon the EPU small break LOCA analysis results. Therefore, the NRC staff concludes that for plants that will implement MELLLA+, sufficient small break sizes should be analyzed at the rated EPU power level to ensure that the peak PCT break size is identified.

Since TRACG-LOCA analyses require consideration of the entire spectrum of break sizes, this limitation is inherently satisfied by any application of the TRACG-LOCA methodology.

10.0 LIMITATIONS

This section of the SE summarizes the limitations[W] that apply to the NRC staff review of NEDE-33005. The limitations are organized into seven categories, and are summarized in the table below. A brief discussion of each limitation follows the table, including cross-references to SE sections and RAI responses.

Limitation		Description
1		Applicability
	1.1	*Nuclear Power Plant Specification*
	1.2	*Native Fuel System Design Applicability*
	1.3	*Competitor and Co-Resident Fuel System Applicability*
	1.4	*First-of-Kind Applications*
	1.5	*Regulatory Compliance*
	1.6	*Promulgation of 10 CFR 50.46c*
2		Deterministic Analysis
	2.1	*Core Detail*
	2.2	*Hot Channels*
	2.3	*Break Spectrum Analysis*
	2.4	*Initial Conditions and Plant Parameters*
	2.5	*General Design Criterion 35 Compliance*
	2.6	*Calorimetric Power Uncertainty*
	2.7	*Use of the Shumway Correlation*
3		Upstream and Concurrent Methods
	3.1	*Reporting Requirements for Upstream/Concurrent Methods*
	3.2	*Limitations on the Use of Upstream/Concurrent Methods*
4		Statistical Analysis
	4.1	*Limiting Break Sizes*
	4.2	*Sample Size*
	4.3	*Rejection of Results*
	4.4	*Successive Elimination*
	4.5	*Dispositioning Unacceptable Statistical Results*
	4.6	*Resampling*
5		Interim Limitation on Cathcart-Pawel
6		Applicability of TRACG-LOCA to Expanded Operating Domains
7		BWR/3-6 First-of-a-Kind Application

Table 12. Summary of TRACG-LOCA Limitations.

10.1. APPLICABILITY

The following limitations delineate the applicability of TRACG-LOCA.

[W] This SE has favored the term “limitations,” however, no distinction is made between “limitations” and “conditions” as used in Office of Nuclear Reactor Regulation Office Instruction LIC-500, “Topical Report Process.”

10.1.1. Limitation 1.1: Nuclear Power Plant Specification

The methods described in NEDE-33005 are considered applicable to General Electric-designed nuclear steam supply systems of the BWR/2 through BWR/6 vintage. The model is not considered applicable for any other classes of plant designs. An extension of such applicability would require the submittal, review, and approval of a separate LTR. Refer to SE Section 3.1.

10.1.2. Limitation 1.2: Fuel System Design Applicability

The review considered Global Nuclear Fuel (GNF) fuel designs through GNF-2. Extension to evolutionary GNF fuel product lines may be completed in accordance with the NRC-approved General Electric Standard Application for Reactor Fuel (GESTAR-II). If a new fuel design feature cannot be readily accommodated via the TRACG-LOCA methodology, or is not supported by its qualification, GEH shall submit the modeling or methodology changes, or updated qualification, needed to accommodate the design to the NRC for review and approval prior to implementing such changes in licensing applications of the TRACG-LOCA methodology. Refer to SE Section 3.1.

10.1.3. Limitation 1.3: Competitor and Co-Resident Fuel System Applicability

Passages within the LTR and the RAI responses discuss modeling approaches and uncertainty treatments that are fuel-design-specific. Such items include limiting power shapes, cladding material properties, and critical quality-boiling length correlations. The analytic treatment of competitor and co-resident fuel is acceptable to the extent that it is supported by modeling approaches appropriate for the fuel design. However, the co-resident fuel must be operated within constraints that are at least as restrictive, jointly and severally, as those established by, or employed in, the ECCS evaluation furnished by the competitor. Alternatively, a licensee may propose an RLA justifying the applicability and acceptability of using TRACG-LOCA to set less limiting operating characteristics. Refer to SE Section 3.1.

10.1.4. Limitation 1.4: First-of-Kind Applications

If an NRC licensee chooses to apply this method in a fashion inconsistent with the existing TRACG-LOCA approval basis and requiring prior NRC staff approval, it is recommended that such a request be submitted as a standalone request. Embedding such a request in a separate requested licensing action (RLA), such as a request for an expanded operating domain, could result in the rejection of the RLA for use of unapproved methods, or in complicating or delaying the review of the RLA.

10.1.5. Limitation 1.5: Regulatory Compliance

This model is considered approved for use by the NRC staff for the purpose of evaluating emergency core cooling system performance under the requirements of § 50.46(a)(1)(i). More specifically, the model is considered acceptable for demonstrating compliance with the acceptance criteria set forth in §50.46(b), paragraphs (1) through (3). GEH addresses compliance with criteria (b)(4) and (b)(5) in Section 2.5 of the LTR. Compliance with Criterion (b)(4) is addressed by compliance with Criteria (b)(1) and (b)(2). GEH stated that the "bases and demonstration of compliance with Criterion [(b)(5), long-term cooling]… do not need to be evaluated as part of the TRACG ECCS/LOCA analysis."

Based on the statements by GEH, the NRC staff did not consider the application of TRACG-LOCA for evaluating long-term core cooling scenarios pursuant to Criterion (b)(5) of § 50.46. This limitation should not be construed to preclude the use of TRACG for compliance with § 50.46, Criterion (b)(5), but rather to clarify that such use has not been considered in the present review effort, and would require a separate application review, should a licensee wish to apply long-term cooling licensing calculations using the TRACG-LOCA methodology. Refer to SE Section 3.2.

10.1.6. Limitation 1.6: Promulgation of 10 CFR 50.46c

The approval of TRACG-LOCA applies to the revision of 10 CFR § 50.46 appearing in the January 1, 2016, version of Title 10 of the Code of Federal Regulations. A subsequent revision to the emergency core cooling requirements is presently under consideration by the Commission, which may result in the promulgation of new, performance-based requirements under § 50.46c. A revision or supplement to NEDE-33005 would be required to obtain NRC approval to use TRACG-LOCA for the purposes of compliance with § 50.46c.

10.2. DETERMINISTIC ANALYSIS AND MODELS

The following limitations apply to the use of TRACG-LOCA to evaluate a plant-specific break spectrum and determine the limiting breaks, upon which to perform the statistical analysis.

10.2.1. Limitation 2.1: Core Detail

Use of the detailed channel grouping described in the response to request for additional information (RAI) 6 constitutes a minimum required for analysis. Each core analyzed shall **[[**

]] Refer to SE Section 5.3 for additional detail.

10.2.2. Limitation 2.2: Hot Channels

As discussed in the response to RAI 102, **[[**
]] The RAI response provides additional detail regarding the initial conditions for these hot channels, their location in the core, and the process for adding additional, hot channels. GEH shall adhere to this process in performing plant-specific evaluations. Refer to SE Section 5.3 for additional detail.

10.2.3. Limitation 2.3: Break Spectrum Analysis

The break spectrum analysis shall be performed at the limiting location and ECCS configuration to identify the most severe event(s). In regions of limiting break sizes, the break spectrum analysis shall be performed at sufficient resolution as to permit overlap, in terms of effective break area, when the critical flow uncertainty is applied in the subsequent statistical analysis. Refer to SE Sections 6.2 and 5.4 for additional detail.

10.2.4. Limitation 2.4: Initial Conditions and Plant Parameters

Excluding uncertainty as determined using an NRC-approved instrument setpoint methodology, variability in initial conditions and plant parameters not specifically addressed in Chapter 6 of the LTR, as supplemented by the RAI responses, shall not be analyzed using the statistical analysis, but rather shall be treated deterministically as set forth in revised Chapter 6 of the LTR. If the deterministic approach is applied, significant plant parameters must be assumed to be in their most pessimistic condition with regard to the results; Technical Specification Analytic Limits for which the ECCS evaluation is an Applicable Safety Analysis shall be used; and parameters shown to have an insignificant effect on the results may be assumed to be in a nominal condition. Refer to SE Section 5.5 for additional detail.

10.2.5. Limitation 2.5: General Design Criterion 35 Compliance

The approach for plant-specific compliance with GDC 35 is provided in response to RAIs 12 and 76. Each plant implementing TRACG-LOCA shall evaluate whether the loss-of-offsite-power (LOOP) or offsite power available (OPA) condition is more limiting for each break scenario to establish that the GDC 35 requirement is met. An explicit analysis of both conditions for every scenario is not required if sufficient evidence exists to conclude whether OPA or LOOP is limiting otherwise. Refer to SE Section 5.6 for additional detail.

10.2.6. Limitation 2.6: Calorimetric Power Uncertainty

The vendor shall apply a 2-percent increase to thermal power to account for calorimetric power uncertainty. An alternative value may be used if appropriately justified. However, if an implementing licensee uses an ultrasonic flow meter for which the NRC has withdrawn its approval of the supporting licensing topical report, such licensee must revert to the generic, 2-percent value. Refer to NRC Regulatory Issue Summary 2007-24, “NRC Staff Position on use of the Westinghouse Crossflow Ultrasonic Flow Meter for Power Uprate or Power Recovery,” for additional details. Refer to SE Section 5.2 for additional discussion.

10.2.7. Limitation 2.7: Use of the Shumway Correlation

The basis set forth for acceptance of the Shumway Correlation, as described in Section 6.4.3 of this SE, is specific to ECCS evaluations, including consideration of the conservatism associated with application of the correlation, its significance in ECCS evaluation, and the thermal-hydraulic conditions associated with a BWR LOCA. Therefore, the Shumway Correlation shall not be construed as accepted by the NRC staff for any other application unless specifically determined in separate correspondence.

10.3. UPSTREAM AND CONCURRENT METHODS

Upstream methods include any evaluation model or computational device that provides input to the TRACG-LOCA evaluation model. Concurrent methods are codes, relations, or subroutines that are implemented within TRACG, but the approval basis is documented elsewhere, such as TGBLA06/PANAC11, or the GEXL correlation. Concurrent methods are technically considered part of the emergency core cooling system evaluation model, but their approval basis is documented in separate LTRs.

10.3.1. Limitation 3.1: Reporting Requirements to Upstream/Concurrent Methods

Errors in, and changes to, concurrent or upstream methods, and in their applications, are subject to the reporting requirements of 10 CFR § 50.46(a)(3), to the extent that they impact TRACG-LOCA evaluations. If errors in, or changes to, concurrent or upstream methods, or in their applications, are identified, they must be dispositioned in accordance with § 50.46(a)(3) and other applicable regulatory requirements. However, optional changes to concurrent or upstream methods, need only be dispositioned when adopted within a plant-specific application.

10.3.2. Limitation 3.2: Limitations on the Use of Upstream/Concurrent Methods

All upstream and concurrent methods must be used in TRACG-LOCA within their existing approval bases. Such methods must be applied in adherence to all conditions and limitations applied thereto. Refer, for example, to Section 6.1.1, for additional discussion.

10.4. STATISTICAL ANALYSIS

The following limitations apply to the statistical combination of uncertainty. Each of these limitations follows the discussion provided in Section 7.3 of this SE.

10.4.1. Limitation 4.1: Limiting Break Sizes

GEH shall perform a statistical analysis for the limiting hypothetical LOCA in each plant application. Each analysis shall rely on a random sample of 124 cases to provide tri-variate, 95/95 tolerance limits for the three critical safety parameters, without any rejected cases. Alternatively, 181 cases may be analyzed for the purpose of rejecting two analytic cases. Limiting results for each of peak cladding temperature, equivalent cladding reacted, and core-wide oxidation shall be obtained by identifying the most limiting result for each figure of merit from the sample, once the rejected cases have been eliminated. If the deterministic break spectrum identifies more than one potentially limiting LOCA scenario, then each potentially limiting scenario shall be statistically analyzed as a sensitivity.

10.4.2. Limitation 4.2: Sample Size

The methodology is approved under the consideration that GEH will perform one sample, consisting of either 124 or 181 cases. The sample size must be determined in advance of the statistical analysis, and no changes to the sample size are permitted.

10.4.3. Limitation 4.3: Rejection of Results

If a sample of 124 cases is selected, no statistical trials may be rejected from the sample. If a sample of 181 cases is selected, two statistical trials may be rejected from the sample.

10.4.4. Limitation 4.4: Successive Elimination

If using the sampling scheme with 181 cases and two rejections, GEH must determine the elimination strategy at the time the sample size is chosen. For example, the vendor must choose whether to eliminate the two cases with the highest ECR, or the two cases with the highest PCT, or one of each, prior to initiating the statistical analysis.

10.4.5. Limitation 4.5: Dispositioning Unacceptable Results

Once the limiting plant configuration is determined and the licensing statistical analysis is applied, GEH must document, in auditable format, the characteristics of any licensing statistical analyses that produced unacceptable results, and what changes were subsequently made to produce acceptable results. If a TRACG-LOCA analysis is used to support a RLA submitted to the NRC, such information shall be included in the request.

This limitation applies to the licensing statistical analysis only, and should not be construed to require such documentation of normal engineering and design analysis that would not otherwise be included in the documentation associated with a particular TRACG-LOCA analysis.

10.4.6. Limitation 4.6: Resampling

Unless a plant makes a major design change, such as the implementation of a power uprate or introduction of a new fuel type that would be expected to change predicted ECCS performance significantly, resampling is not permitted in concert with re-analysis.

10.5. INTERIM LIMITATION ON CATHCART-PAWEL RESULTS

The use of the Cathcart-Pawel oxidation correlation requires consideration of the fact that the 17 percent limit was based on the use of the Baker-Just equation. Local cladding oxidation results obtained using the Cathcart-Pawel equation will be considered acceptable, provided they are below 13 percent. If Cathcart-Pawel ECR remains below 13 percent and the PCT is below 2200 °F, there is reasonable assurance that the ECR would also remain below 17 percent, if calculated using Baker-Just. This limitation is an interim measure pending Commission adoption of a revision to the 10 CFR 50.46(b) acceptance criteria; should the staff position on this matter change, the NRC will notify GEH via a letter providing the revised position. This is discussed in Section 6.1.4 of this SE.

10.6. APPLICABILITY OF TRACG-LOCA TO EXPANDED OPERATING DOMAINS

In license amendment requests to implement expanded operating domains, for which the ECCS-LOCA analysis is based on TRACG-LOCA, the requesting licensee shall include documentation of the supporting ECCS-LOCA analysis in the license amendment request. Such inclusion will address the relevant conditions and limitations applicable to both NEDC-33173P-A, “Applicability of GE Methods to Expanded Operating Domains” (also known as the Interim Methods Licensing Topical Report, IMLTR), and NEDC-33006P-A, “Maximum Extended Load Line Limit Analysis Plus.”

Additional discussion addressing the prior limitations and conditions, specifically, is included in Chapter 9, “Licensing Considerations,” of this SE.

10.7. BWR/3-6 FIRST-OF-A-KIND APPLICATION

The NRC staff review effort included a detailed review of TRACG-LOCA as applied to a BWR/2, and as such, the application of TRACG-LOCA to Nine Mile Point Nuclear Station, Unit 1, is acceptable without further limitation. However, the NRC staff notes that the demonstration analyses and nodalization sensitivity studies supporting application of TRACG-LOCA to BWR/3-6 plants, were not updated to reflect the increased core detail and revised statistical approach that were revised as a result of the NRC staff review. As such, the NRC staff requires that GEH perform updated demonstration analyses for each of a BWR/4 and BWR/6, and an update to the BWR/4 nodalization sensitivity studies, and provide them for NRC staff review and acceptance, prior to first-of-a-kind application of TRACG-LOCA to a BWR/3-6. Specifically, the jet pump plant nodalization studies should be updated/reviewed/accepted prior to application to a jet pump plant. The BWR/4 demonstration studies should be updated/reviewed/accepted prior to application to a BWR/3-4, and similarly, the BWR/6 demonstration studies should be updated/reviewed/accepted prior to application to a BWR/5-6. This limitation can be satisfied by revising the jet pump plant nodalization studies documented in LTR Section 5.2, Table 5.2-1 and Figures 5.2-1 through 5.3-9 and the key summary demonstration analyses documented in LTR Chapter 8, Figure 8.1-29 for the BWR/4 and Figure 8.2-18 for the BWR/6.

11.0 CONCLUSION

Based on the review described in the preceding sections, and subject to the limitations delineated in Chapter 10 of this SE, the NRC staff determined that NEDE-33005P is acceptable for referencing in licensing actions. For the purpose of compliance with 10 CFR 50.46 requirements, TRACG-LOCA, as documented in Reference 1 and revised by the RAI responses, may be considered an acceptable evaluation model. With regard to referencing in licensing actions, Reference 1 as revised by the RAI responses may be considered approved for use.

12.0 REFERENCES

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8. U.S. NRC, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” Regulatory Guide 1.157, ADAMS Accession No. ML003739584.
9. U.S. NRC, “Transient and Accident Analysis Methods,” Regulatory Guide 1.203, ADAMS Accession No. ML053500170.
10. U.S. NRC, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition,” NUREG-0800, Chapter 15.0.2, “Review of Transient and Accident Analysis Methods,” Revision 0, March 2007, ADAMS Accession No. ML070820123.
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Attachment: Resolution of Comments

Principal Contributor: Ben Parks, SNPB/DSS

Date: February 22, 2017

Comment Summary for Draft Safety Evaluation for NEDE-33005P Revision 0, "TRACG Application for Emergency Core Cooling Systems/ Loss-of-Coolant-Accident Analyses for BWR/2-6"

Location	GE Comment	NRC Disposition
Executive Summary	GEH suggests the following change (Line 46): NEDE-33005P *Suggested change shown in the markup.*	The NRC finds the change acceptable. Change incorporated in final SE.
List of Acronyms	Four acronyms are in the SER but are not in the List of Acronyms. GEH suggests the following changes: 1. Add the following acronyms: B-J (Baker Just), C-P (Cathcart-Pawel), IET (Integral Effect Test, and TR (Topical Report) 2. CSAU: Delete 'Analysis' 3. GEH: Change "General Electric-Hitachi" to "GE Hitachi" 4. GNF: Change "Fuels" to "Fuel" 5. SRP: Change "Plant" to "Plan" *Suggested changes shown in the markup.*	The changes are editorial; however, the acronym "GE" embedded within "GEH" does not appear in the list of acronyms. Change accepted and incorporated in final SE; in addition, add to the list of acronyms, "GE General Electric"
Section 3.3.1 PIRT Process – NRC Staff Evaluation	Page 10: GEH recommends expanding the text. GEH suggests the following change to Footnote E: [[ ]] *Suggested changes shown in the markup.*	The change provides a more detailed description of the contributors to the effect described in the footnote, but does not change its meaning. NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 4.1.2 Provision of Complete Code Documentation	Page 12: The text incorrectly implies that Table 3 lists all recently approved TRACG applications. Those listed are only TRACG AOO applications. GEH suggests the following change (Lines 8-9): Table 3 summarizes the TRACG AOO applications for US ~~more recent, application-specific approvals received for TRACG for currently licensed and~~ operating BWRs. *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 4.2 Summary of Previous Review Findings	Page 13: To be consistent with the changes suggested in Section 4.1.2. GEH suggests the following change (Line 1): 4.2. SUMMARY OF PREVIOUS REVIEW FINDINGS RELATED TO TRACG FOR AOO APPLICATIONS *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 4.2 Summary of Previous Review Findings	Page 13: To be consistent with the changes suggested in Section 4.1.2. GEH suggests the following change (Line 2): Table 3. Summary of Previous Review Findings Related to TRACG for AOO Applications. *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 4.3.1 Nodalization Section 6.2.2 Noding near the Break and ECCS Injection Point Section 10.0 Limitations	GEH suggests that the use of the terminology 'later vintage' be revised to reflect the 'first-of-a-kind' terminology, consistent with the GEH suggested change to the title of Section 10.7. *The following suggested changes are shown in the markup:* 1. Section 4.3.1: Nodalization: • Page 15 (Line 44): TRACG-LOCA for ~~in later vintage~~ BWR/3-6s. • Page 16 (Lines 20-21): Limitation 7, "~~Later Vintage~~ BWR/3-6 First-of-a-Kind Application," as 2. Section 6.2.2: Noding near the Break and ECCS Injection Point • Page 41 (Line 24): Limitation 7, "~~Later Vintage~~ BWR/3-6 First-of-a-Kind Application," as 3. Section 10.0: Limitations • Page 74 (Line 7): Table 12: ~~Later Vintage~~ BWR/3-6 First-of-a-Kind Application	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 5.2 Operating Statepoints	Page 21: The figure should be cited from the MELLLA+ LTR and referred to as typical since some plant-specific maps are different. GEH suggests the following change (Line 5): A typical ~~The~~ MELLLA+ operating domain is illustrated in Figure 1-1 of Reference 58 ~~Figure 2~~. *Suggested change shown in the markup.*	The figure in question is a reproduction of Figure 6.2-1 of NEDE-33005P. Since the figure was obtained from the LTR that is the subject of the SE, the NRC staff retained the figure as-is. The proposed change is modified as follows: A typical ~~The~~ MELLLA+ operating domain is illustrated in Figure 2, which is a reproduction of Figure 6.2-1 of NEDE-33005P.

Location	GE Comment	NRC Disposition
Section 5.2 Operating Statepoints	Page 21: The figure should be cited from the MELLLA+ LTR and referred to as typical since some plant-specific maps are different. If the figure is retained, then a better figure is needed since the figure in the SE has Points D and E misaligned. GEH suggests the following change (Line 7): Replace Figure 2 with Figure 1-1 from Reference 58.	The figure in question is a reproduction of Figure 6.2-1 of NEDE-33005P. Since the figure was obtained from the LTR that is the subject of the SE, the NRC staff retained the figure as-is. Change not accepted.
Section 5.2 Operating Statepoints	Page 21: If Figure 1-1 from Reference 58 replaces the existing Figure 2, then the points analyzed for MELLLA+ plants needs to be revised. GEH suggests the following change (Line 11): points ~~A~~ B, D, and E in Figure 2. *Suggested change shown in the markup.* Note that if the existing Figure 2 is not replaced, then the points should be A, C, and E.	The figure contained in NEDE-33005P was retained in the SE (i.e., existing Figure 2 was not replaced); therefore, the GEH suggestion to identify Points A, C, and E was considered by the staff. As noted on Page 71 of the SE, the applicable requirement for MELLLA+ is to analyze, "the rated EPU [extended power uprate] power/rated CF [core flow], rated EPU power/minimum CF, and the low-flow MELLLA+ boundary (Transition Statepoint). These correspond to Points A, C, and E, on the figure. Therefore, the NRC staff agrees with this suggested change because it is consistent with the MELLLA+ analytic requirements. The proposed change is modified as follows: points A, C, and E in Figure 2.
Section 5.3 Power Distribution and Channel Groupings	Page 23: GEH suggests the following change (Line 29) the limiting heated nodes are *Suggested changes shown in the markup.*	The NRC staff agrees with the clarification. The NRC finds the change acceptable. Change incorporated in final SE.
Section 5.3 Power Distribution and Channel Groupings	Page 23: Standard and hot channel are the same. Clarify the footnote. GEH suggests the following change to Footnote H: Note that the standard channel nodalization~~, hot channel~~ includes **[[ ]]** axial nodes, of which 25 are heated. *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 5.4 Break Spectrum Analysis	Page 25: Typographical error. GEH suggests the following change (Line 9): rec~~e~~ommends *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 5.5.1 Initial Conditions	Page 26: Limitation 2.5 should be Limitation 2.4. GEH suggests the following change (Line 27): Limitation 2.~~5~~4 *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 5.5.2 Plant Parameters	Page 26: Limitation 2.5 should be Limitation 2.4. GEH suggests the following change (Line 46): Limitation 2.~~5~~4 *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1 Sources of Heat During a Loss-of-Coolant Accident	Page 29: GEH suggests the following change (Line 8): Cathcart-Pawel (C-P). *Suggested change shown in the markup.*	The NRC finds the change acceptable. Change incorporated in final SE..
Section 6.1.1 Initial Stored Energy of the Fuel	Page 30: The current statement is inaccurate. GEH suggests the following changes (Lines 9-10): These PRIME inputs ~~include~~ are used by TRACG to calculate the steady-state temperature distribution and stored energy in the fuel. The PRIME inputs also include the composition and pressure of the fuel rod gases, which are used to calculate the gap gas conductivity. *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 6.1.1 Initial Stored Energy of the Fuel	Page 30: GEH suggests the following changes (Line 25): ~~l~~Limitation 3.2, "Limitations on the Use of Upstream/Concurrent Methods," *Suggested changes shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.3 Decay of Actinides and Fission Product Decay Heat	Page 32: The current statement is inaccurate. GEH suggests the following changes (Line 44): TRACG implements decay heat ~~curves~~ models based on both the 1979 and 1994 ANS standards, ~~via~~ like an auxiliary code, DECAY. *Suggested changes shown in the markup.*	The NRC staff understands that specific curves are generated as output of the TRACG decay heat models, and the review was accomplished using some such curves as representations of decay heat modeling approaches. Furthermore, the response to RAI 101 makes the distinction that the TRACG decay heat models are based on DECAY. The NRC staff intent in the discussion is consistent with the vendor proposed revision, and with the RAI response as submitted. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.3 Decay of Actinides and Fission Product Decay Heat	Page 33: Clarify the sentence. GEH suggests the following change (Line 4): higher than ~~a comparable model~~ the DECAY code that calculates *Suggested change shown in the markup.*	The change is editorial in that it adds specificity; however, the NRC staff identified an additional change to the sentence structure that would be needed to accept the proposed change. The proposed change is modified and incorporated as follows: Higher than ~~a comparable model~~ the DECAY code ~~that~~, which calculates
Section 6.1.3 Decay of Actinides and Fission Product Decay Heat	Page 33: The current sentence is inaccurate. GEH suggests the following change (Line 39): The same ~~uncertainty~~ multiplicative factor is applied to each channel in the core, *Suggested change shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.4 Metal-Water Reaction Rate	Page 34: GEH suggests the following change (Line 37): Baker-Just (B-J) *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 6.1.4 Metal-Water Reaction Rate	Page 36: The SE language is confusing when compared to Limitation 5. The interim limitation is the 13% acceptance criterion for Cathcart-Pawel. GEH suggests the following changes (Lines 14-15): In view of the ~~fact that the use of a 17-percent~~ interim 13-percent acceptance criterion ~~solely using the B-J reaction 48 kinetics equation is an interim limitation~~ when using the C-P reaction model, should the NRC *Suggested changes shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.5 Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods	Page 36: The sentence incorrectly implies that most or all heat transfer effects outside the cladding are neglected, whereas the opposite is true. GEH suggests the following change (Lines 44-45): Some h~~H~~eat transfer effects outside the cladding, such as droplet shattering upon impingement on ruptured fuel rod segments, are neglected. *Suggested change shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.5 Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods	Page 37: The sentence is incorrect. TRACG calculates the internal rod pressure to account for local power. GEH suggests the following changes (Line 2): The reference initial rod ~~internal~~ pressure is passed *Suggested changes shown in the markup.*	As written, the sentence may have implied that coupled calculations are performed using both PRIME and TRACG, when in fact PRIME supplies initial conditions for the TRACG calculation. The NRC staff intent in the discussion is consistent with the vendor-proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.1.5 Thermal Parameters for Swelling and Rupture of the Cladding and Fuel Rods	Page 37: GEH suggests the following change (Line 20): [[ ]] *Suggested change shown in the markup*	The staff evaluation determined that data reflected ramp rates expected for BWR LOCA events. Such a model would only be viewed as conservative if the events typically exhibited higher ramp rates, contrary to the following sentence in the SE. Change not accepted.

Location	GE Comment	NRC Disposition
Section 6.2.1 Break Characteristics and Flow	Page 40: GEH suggests the following change (Line 44): Limitation ~~2.4~~2.3 *Suggested change shown in the markup*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.2.3 Critical Heat Flux	Page 42: Limitation 1.2 is different than Limitation 1.3. Revise the text to be consistent with Limitation 1.3. GEH suggests the following change (Lines 27-30): The bounding uncertainty approach is acceptable for average channels in the core, but ~~as per Limitation 1.2, TRACG-LOCA is restricted to analysis of GNF fuel designs, meaning that the analyzed hot channels must be designed by GNF~~ hot channel modeling of competitor fuel is restricted by Limitation 1.3. *Suggested change shown in the markup*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.3 Post-CHF Phenomena	Page 42: GEH suggests the following change (Line 36): Particular ~~import~~ importance *Suggested change shown in the markup*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 6.3.3 Heat Transfer	Page 46: Reference 35 is not explicitly cited in the SER. GEH suggests the following change (Line 33): correlation (Reference 35) is used *Suggested change shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. Furthermore, the citation of Reference 34 in the following paragraph is erroneous. The NRC finds the change acceptable. Change incorporated in final SE. Also, the citation to Reference 34 in the following paragraph is corrected to Reference 35.

Location	GE Comment	NRC Disposition
Section 6.4.3 Rewet and Quench Behavior	Page 51: The word "correction" is misleading because it implies modifying, after the fact, what the author understood (as did others in his day) the importance of material properties. The term was designed into the correlation from the beginning. GEH suggests the following change (Lines 30 and 32): ~~correction~~ dimensionless term *Suggested change shown in the markup*	The NRC staff was unable to confirm whether the author would agree with such inference, and therefore makes no conclusion regarding whether "correction" or "dimensionless" is more appropriate; however, adoption of the proposed revision does not alter the meaning of the text. The NRC finds the change acceptable. Change incorporated in final SE.
Section 7.3.2 Summary of GEH Approach	Page 58: It is unclear what GEH assertion in RAI 103 was not accepted by the NRC since SE Section 7.3.3 contains the NRC staff evaluation accepting the GEH approaches identified in RAI 103. GEH suggests the following change (Lines 35-38): ~~GEH asserted, in its response to RAI 103, that the order of selective elimination is unimportant, because the resulting, estimated tolerance region will still remain bounding of the true 95/95 joint tolerance region that would exist in reality. As discussed in the following section, the NRC staff did not accept this portion of the proposed approach.~~ *Suggested change shown in the markup.*	The staff concern discussed in this paragraph is addressed by Limitation 4.4. Based on this consideration, and on the vendor feedback, the NRC staff determined that this paragraph is moot, and hence agrees that it can be deleted. The NRC finds the change acceptable. Change incorporated in final SE.
Section 7.3.3 NRC Staff Evaluation of GEH Approach	Page 59: GEH suggests the following change (Line 11): References 49 ~~and~~ through 53 *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 7.3.3 NRC Staff Evaluation of GEH Approach	Page 60: GEH suggests the following change (Line 21): Guba, Makai, and Pal (Reference 49), *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 7.4 PCT Analysis Resolution and Core Detail	Page 62: GEH suggests the following change (Line 26): (MFN 14-064 <u>(Reference 3)</u>, Page 27). *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 8.2 Review of Studies for BWR/2 Plant	Page 65: GEH suggests the following change (Line 31): ~~generated~~<u>indicated</u> *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 8.2 Review of Studies for BWR/2 Plant	Page 66: Statement pertaining to GEH commitment is misleading. There is no need to validate the isolation condenser heat transfer if the isolation condenser is not being credited or even modeled. GEH suggests the following change (Lines 11-12): ~~GEH further committed to validate the isolation condenser heat transfer uncertainty treatment on a plant-specific basis.~~ *Suggested change shown in the markup.*	The NRC staff disagrees with this remark. Particularly, the final sentence of the vendor response to RAI 62 states, "If the isolation condenser is to be credited, we commit to validate the distributions that are appropriate on a plant-specific basis." This paragraph of the SE concludes by stating these models would require review prior to plant specific implementation. Change not accepted.
Section 9.1 TRACG-LOCA Application	Page 66: GEH suggests the following change (Line 39): Amendment ~~XX~~ <u>37</u>. *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 9.3.1 Extended Power Uprates	Page 68: When the PDF was created from the word file, the link to the reference for ELTR 1 was broken. GEH suggests the following change (Line 12): Fix the broken link to the reference for ELTR 1.	No link was intended. Appropriate correction has been made in the final SE.

Location	GE Comment	NRC Disposition
Section 9.3.1 Extended Power Uprates	Page 68: When the PDF was created from the word file, the link to the reference for ELTR 2 was broken. GEH suggests the following change (Line 15): Fix the broken link to the reference for ELTR 2.	No link was intended. Appropriate correction has been made in the final SE.
Section 9.3.2 Interim Methods	Page 70: GEH suggests the following change (Line 15): will include~~d~~ *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 9.3.2 Interim Methods	Page 70: GEH suggests the following change (Lines 38-39): Add a blank line to separate '9.3.3 Maximum Extended Load Line Limit Analysis Plus' from the paragraph immediately above it. *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 9.3.3 Maximum Extended Load Line Limit Analysis Plus	Page 71: Enhance readability and clarity. GEH suggests the following change (Line 9): Add an un-indented sentence before the indented paragraphs that states: The following indented paragraphs refer to applications of the SAFER/GESTR-LOCA methods: *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 9.3.3 Maximum Extended Load Line Limit Analysis Plus	Page 72: Enhance readability and clarity. GEH suggests the following change (Line 18): Add an un-indented sentence before the indented paragraphs that states: For SAFER/GESTR-LOCA applications: *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 9.3.3 Maximum Extended Load Line Limit Analysis Plus	Page 72: Enhance readability and clarity. GEH suggests the following change (Line 32): Add an un-indented sentence before the indented paragraphs that states: For SAFER/GESTR-LOCA applications: *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 9.3.3 Maximum Extended Load Line Limit Analysis Plus	Page 73: GEH suggests the following change (Line 3): Add a blank line to separate the LOCA Break Spectrum limitation from the paragraph immediately following it. *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.0 Limitations	Page 74: GEH suggests the following change (Line 3): organized into ~~six~~ seven categories *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.1.2 Limitation 1.2: Fuel System Design Applicability	Page 75: GEH suggests the following change (Line 11): Global Nuclear Fuel~~s~~ (GNF) fuel *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.2.3 Limitation 2.3: Break Spectrum Analysis	Page 76: GEH suggests the following change (Line 44): Sections 6.2 and ~~7.4~~5.4 *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.2.6 Limitation 2.6: Calorimetric Power Uncertainty	Page 77: Delete a comma. With the comma, the sentence reads as if the NRC has withdrawn approval of all ultrasonic flow meters. GEH suggests the following change (Line 23): licensee uses an ultrasonic flow meter~~,~~ for which the NRC has *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
Section 10.3.1 Reporting Requirements to Upstream/Concurrent Methods	Page 78: GEH suggests the following change (Line 1): <u>Limitation 3.1:</u> Reporting Requirements to Upstream/Concurrent Methods *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.3.2 Limitations on the Use of Upstream/Concurrent Methods	Page 78: GEH suggests the following change (Line 9): <u>Limitation 3.2:</u> Limitations on the Use of Upstream/Concurrent Methods *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.
Section 10.4.6 Limitation 4.6: Resampling	Page 79: Please clarify whether changing the fuel to a new product is a "major design change". GEH suggests the following change (Lines 13-14): power uprate <u>or introduction of a new fuel type</u>, *Suggested change shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision, subject to clarification. The proposed change is modified and incorporated as follows: power uprate or introduction of a new fuel type that would be expected to change predicted ECCS performance significantly,

Section 10.7 Later-Vintage BWR Application	Pages 79-80: Suggest that the NRC staff consider the more precise wording suggested below to clarify the requirements for removing this limitation. GEH suggests the following change (Page 79 Line 39 through Page 80 Line 13): 10.7. ~~LATER-VINTAGE~~ BWR/3-6 FIRST-OF-A-KIND APPLICATION The NRC staff review effort included a detailed review of TRACG-LOCA as applied to a BWR/2, and as such, the application of TRACG-LOCA to Nine Mile Point Nuclear Station, Unit 1, is acceptable without further limitation. However, the NRC staff notes that the demonstration analyses and nodalization sensitivity studies supporting application of TRACG-LOCA to ~~later vintage BWRs, specifically~~ BWR/3-6 ~~product lines~~plants, were not updated to reflect the increased core detail and revised statistical approach that were ~~devised~~ revised as a result of the NRC staff review. As such, the NRC staff requires that GEH perform updated demonstration analyses for each of a BWR/4 and BWR/6, and an update to the BWR/4 nodalization sensitivity studies, and provide them for NRC staff review and acceptance, prior to ~~implementation~~ first-of-a-kind application of TRACG-LOCA ~~at later-vintage~~to a BWR/3-6 ~~product lines~~. Specifically, the jet pump plant nodalization studies should be updated/reviewed/accepted prior to application to a jet pump plant. The BWR/4 demonstration studies should be updated/reviewed/accepted prior to application to a BWR/3-4, and similarly, the BWR/6 demonstration studies should be updated/reviewed/accepted prior to application to a BWR/5-6. This limitation can be satisfied ~~either~~ by revising the jet pump plant nodalization studies documented in LTR Section 5.2 Table 5.2-1 and Figures 5.2-1 through 5.3-9 and the key summary demonstration analyses documented~~provided~~ in LTR Chapter 8	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.

Location	GE Comment	NRC Disposition
	Figure 8.1-29 for the BWR/4 and Figure 8.2-18 for the BWR/6~~of the LTR, or by performing analyses of specific plants~~. *Suggested change shown in the markup.*	
Section 11.0 Conclusion	Page 80: GEH suggests the following change (Lines 19-21): For the purpose of compliance with 10 CFR 50.46 requirements, TRACG-LOCA, as documented in Reference 1 and revised by the RAI responses, may be considered an acceptable evaluation model. With regard to referencing in licensing actions, Reference 1 as revised by the RAI responses may be considered approved for use. *Suggested change shown in the markup.*	The NRC staff intent in the discussion is consistent with the vendor proposed revision. The NRC finds the change acceptable. Change incorporated in final SE.
Section 12.0 References	Pages 81-82: GEH suggests the following changes: • Page 81, Line 43: Reference 17: MFN 07-~~457~~452 • Page 82, Line 47: Reference 31: December 2~~0~~8 *Suggested change shown in the markup.*	The change is editorial. The NRC finds the change acceptable. Change incorporated in final SE.