NRC STAFF ASSESSMENT OF GE-HITACHI NUCLEAR ENERGY /

GLOBAL NUCLEAR FUEL - AMERICAS CODES AND METHODS WITH REGARD TO

THERMAL CONDUCTIVITY DEGRADATION

PART I: EXPERIMENTAL DATA

1 INTRODUCTION

NUREG/CR-6534, Volume 1, "FRAPCON-3: Modifications to Fuel Rod Material Properties and Performance Models for High-Burnup Application" provides a description of relevant experimental data indicating degradation of fuel thermal conductivity with exposure. NUREG/CR-6534, Volume 1 also provides a discussion of the revisions to the FRAPCON-3 fuel thermal conductivity model based on more recent test data collected for irradiated fuel. Relevant excerpts from NUREG/CR-6534 are provided in this section for completeness.

2 RELEVANT EXPERIMENTAL DATA

It has long been known that irradiation damage and the progressive buildup of fission products (rare earths, gases, and volatiles) with increasing fuel burnup in sintered urania and urania-plutonia fuel pellets will progressively reduce their thermal conductivity. The effect is stronger at temperatures less than 800K where the phonon-phonon form of heat transfer dominates. See, for example, the early work on this subject by Daniel and Cohen (1964) and the review by Lokken and Courtwright (1977) that suggested that irradiation damage was the primary mechanism at low temperatures. Until relatively recently, however, this reduction in conductivity with increasing burnup has not been included in fuel performance codes because evidence was inconclusive that the effect was significant. The effect was considered insignificant because previously typical end-of-life burnup levels were low for light water reactor (LWR) applications (less than 4 atom percent), and the pellet operating temperatures were relatively high (700K or higher at the pellet surface, and 1300K and higher at the pellet center).

Computer code predictions of pellet operating temperatures are typically benchmarked against steady-state fuel centerline thermocouple measurements from instrumented test fuel rods. These data are combined with the test coolant calorimetry and neutron detector data to yield the total thermal resistance from coolant to pellet center (i.e., the increase in center temperature per unit increase in the local LHGR). This resistance can then be correlated to fuel pellet type, fuel rod design and dimensions, and burnup. However, these data are not definitive regarding the partition of the total thermal resistance between the pellet and the fuel-cladding gap. Thus, an increase in the measured (total) thermal resistance with increasing burnup, which may in part be due to thermal conductivity degradation due to burnup and fuel cracking, has been typically explained as solely due to an increase in the thermal resistance of the pellet-cladding gap. The gap resistance can certainly increase because of pellet densification (which increases the gap size) and/or degradation of the helium-gap gas conductivity by the addition of noble fission gases (xenon and krypton) released from the fuel pellets. Fuel swelling and cladding creepdown decrease the thermal resistance with increasing burnup. Fuel center temperature data also fail to define the partition of thermal resistance within the fuel pellet (e.g., the low and high temperature regions).

Fuel performance codes have typically been benchmarked by retaining fuel thermal conductivity applicable to uncracked pellets and then tuning the gap closure mechanisms to achieve agreement with in-reactor measurements of pellet center temperature. This approach yields conservative (high) estimates for the pellet surface and average temperatures, and hence for the stored energy associated with a given combination of LHGR and center temperature (see Lanning 1982). One major use of code-calculated fuel temperatures is to estimate the fuel-stored energy and gap conductance as initial conditions for analyzing loss-of-coolant accidents (LOCAs). Furthermore, the limiting LOCA initial conditions generally occur at very low burnup for pressurized water reactors (PWRs) and within 10- to 20-GWd/MTU burnup for boiling water reactors (BWRs). Thus, for this important code application, the error produced by ignoring burnup-induced degradation of fuel thermal conductivity was deemed both small and acceptable.

In the past 20 years, however, requests to the U.S. Nuclear Regulatory Commission (NRC) have been made for commercial fuel operation to ever higher uranium burnup levels, exceeding 7 atom percent, and this has resulted in renewed interest in the degree and nature of burnup-induced degradation due to pellet thermal conductivity. At the same time, better experimental evidence, both in-reactor and ex-reactor, has been obtained to define the conductivity degradation as a function of burnup and temperature.

An instrumented assembly referred to as the Halden Ultra-High-Burnup Experiment has indicated a steady degradation in urania fuel thermal conductivity (averaged over the temperature range from approximately 750K to 1200K) of approximately 5 percent to 7 percent relative per 10 GWd/MTU, for burnups up to 88 GWd/MTU (Wiesenack 1997). These measurements are described in the open literature (see Kolstad 1992; Kolstad et al. 1991; Wiesenack 1995). These data are supplemented by results from other Halden instrumented tests involving small-gap rods operated to significant burnup with surviving centerline thermocouples. The degradation rate initially reported (Kolstad et al. 1991) is qualitatively consistent with the results of laser-flash diffusivity measurements on unirradiated simulated high-burnup fuel performed at Chalk River National Laboratory (CRNL), Ontario, Canada (Lucuta et al. 1991; Lucuta et al. 1992).

Following scrutiny of temperature data trends with burnup in small-gap xenon-filled rods, Halden now proposes that the thermal gap resistance becomes very small (smaller than previously thought) when fuel swelling and cladding creepdown closes the gap and that, therefore, the observed trend in the Ultra-High-Burnup fuel temperatures indicates even stronger pellet conductivity degradation.

3 REFERENCES

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PART II: SURVEY OF GE-HITACHI NUCLEAR ENERGY / GLOBAL NUCLEAR FUEL - AMERICAS REACTOR SAFETY ANALYSIS METHODS

4 INTRODUCTION

The NRC requires that the safety of nuclear power plants be evaluated in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of Applications; Technical Information";10 CFR 50.36, "Technical Specifications"; and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors." To demonstrate compliance with the NRC's regulations, licensees use a series of computer codes to perform safety analyses. Generally, either licensees or vendor organizations develop these computer codes, which are then generically approved by the NRC.

An essential element in the computational approach is the modeling of various physical processes that are important to the prediction of transient and accident events. These models are used in a concerted approach to simulate reactor conditions for postulated events and to assess particular figures of merit for comparison against applicable regulatory criteria.

Current licensing evaluations are performed using legacy fuel thermal performance models. These models simulate the heat transfer aspects of nuclear fuel elements under conditions of normal operation and accident conditions. The legacy fuel thermal performance models are specifically those models that the NRC staff developed and approved based on historical qualification data. These data were limited in terms of the range of exposure to levels far below those levels of exposure representative of modern day fuel designs and operating strategies.

Several vendor organizations have submitted improved fuel thermal models to the NRC for review and approval. These new models incorporate more recent test data which indicate that exposure has a significant impact on the thermal-physical properties of the fuel. In particular, recent experience and experimental data indicate significant degradation of fuel pellet thermal conductivity with exposure. The NRC analysis of these newer, modern fuel models and recent data have led the NRC staff to conclude that the use of the older legacy fuel models will result in predicted fuel pellet conductivities that are higher than the expected values.

Additionally, cycle fuel reloads for power reactors require safety analyses to be performed to ensure adequate protection by establishing operational and safety limits that cannot be exceeded during normal operation. The bases for these limits are cycle-specific safety analyses performed using a suite of NRC-approved computer methods. The NRC staff is concerned that the use of legacy fuel thermal performance models, at multiple points within the body of the safety analyses, may result in the downstream effect of calculated safety limit margins that are less conservative than previously understood.

The NRC staff has considered and documented the impact of fuel thermal conductivity errors on the body of safety analyses provided by GE-Hitachi Nuclear Energy / Global Nuclear Fuel - Americas (GEH/GNF) to BWR licensees. This document reports the results of the NRC staff's evaluation.

5 BACKGROUND

To perform safety analyses, all code systems must include a calculational methodology for evaluating fuel thermal and mechanical performance. In the GEH/GNF suite of codes, this methodology is GESTR-MECHANICAL (GSTRM). GSTRM is characterized by a system of physical and empirical models that are exercised in a coupled numerical framework to predict the fuel rod behavior under postulated operational and transient or accident conditions. Other GEH/GNF calculational methods, in which the prediction of either the fuel rod thermal or mechanical behavior or both is an essential part of the overall system analysis, replicate several of the fuel behavior models from GSTRM.

This section describes the standard reload safety analysis process, the GSTRM thermal-mechanical (T-M) methodology, its review and approval, and subsequent interactions with GEH/GNF regarding the application of the GSTRM methodology to expanded operating domain analyses, as well as the improved GEH/GNF modern fuel T-M code, PRIME.

5.1 Licensing Methodology (GESTAR II)

General Electric Standard Application for Reactor Fuel (GESTAR II) includes a supplement to the base report which provides the safety analysis methodology and information specific to the BWR plants in the United States designed by General Electric Nuclear Energy (GE) (now GEH/GNF) (Reference 1). GEH/GNF provides utility customers with cycle-specific information on a reload basis. This information is limited to those specific cycle-specific evaluations that are required to demonstrate safety and to establish plant operating and safety limits.

The reload analysis approach in the GESTAR II analysis methodology incorporates generic evaluations of plant designs and operational features. These generic evaluations establish the basis for those events that must be analyzed on a cycle-specific basis for each BWR. These analyses are performed to demonstrate that radioactive release from plants for normal operation, anticipated operational occurrences (AOOs), and postulated accidents meet applicable regulatory limits. The results of these evaluations are transmitted with the supplemental reload licensing report and form the basis for the cycle operating limits report.

The NRC staff has reviewed and approved the GESTAR II reload licensing process, and each amendment to the process is subject to NRC review and approval. The GESTAR II process provides the reload licensing basis and a list of cycle-specific evaluations that must be performed for the fleet of BWR plants, as well as the calculational tools that are used. In its survey of the GEH/GNF methods, the NRC staff considered both the tools used in the cycle reload safety analysis method and generic evaluations and assessments that have been performed as the basis for the GESTAR II approach to each BWR design and plant configuration.

5.2 GSTRM Development and Approval

GE submitted the GSTRM licensing topical report (LTR) for review and approval on December 30, 1977 (Reference 2). This LTR described the basic models and calculational framework that represent the GSTRM methodology. The NRC staff reviewed the GSTRM code consistent with the applicable regulatory guidance and issued its safety evaluation (SE) on November 2, 1983 (Reference 2). GE submitted the GSTRM method as an alternative method to the TEXICO fuel rod model approved in 1972 (Reference 3).

The GSTRM model incorporates a fuel rod thermal model, a fuel rod mechanical model, a fission gas release model, and a fuel rod internal pressure model. These models comprise several empirical relationships for material physical properties and equations that describe relevant physical processes, such as thermal conduction or cladding strain. The code treats the properties themselves as correlated parameters based on various test data. For example, the fuel pellet thermal conductivity is treated as an analytic expression in terms of the fuel temperature. The LTR describes the integral qualification of the GSTRM predictive capabilities relative to test data (Reference 2).

The predominance of GSTRM qualification data are derived from the Halden boiling-water reactor (HBWR) experiments. In several cases, the test data used to correlate models and qualify the methodology were limited to [ ]. Also, the tests conducted in this timeframe considered contemporary fuel rod designs (Reference 2). In modern day reactor core designs, the rod burnups are considerably higher and additional fissile and burnable poison loadings enable higher rod powers at higher burnup relative to those data considered during the development of GSTRM.

During its review of the GSTRM LTR, the NRC staff considered the possibility of imposing a limitation to the GSTRM code restricting its application to fuel exposures of [ ]. The NRC staff noted concerns regarding fission gas release as the basis for the limitation. In particular, the NRC staff noted that, for a limited sample of the qualification data, GSTRM appeared to under-predict the degree of fission gas release at high exposure.

The specific bases for the NRC staff restriction was (1) the GSTRM model for a loss-of-coolant accident (GESTR-LOCA) [ ] and (2) a comparison of the GESTR-LOCA fission gas release model to a modification of the American Nuclear Society (ANS) 5.4 fission gas release model developed by Pacific Northwest National Laboratory (PNNL) shows that the [ ].

GE provided arguments justifying the applicability of the GSTRM fission gas release model. The key arguments posed by GE are summarized as follows:

[

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The NRC staff accepted these arguments and approved the GSTRM methodology without imposing the [ ] restriction (Reference 2). Furthermore, the NRC staff approved GSTRM with the understanding that GE would revise the models within GSTRM to reflect more modern test data as they become available and to represent future fuel rod designs. The NRC staff accepted the use of a significance test to determine whether NRC review and approval of model revisions was warranted. In 1984, GE submitted the five significance test criteria for GSTRM model revisions (Reference 4). Similarly in 1984, GE submitted the changes that were made to GSTRM between the approval of the GESTR-LOCA mechanical model and the code version referenced in GESTAR II Amendment 7 (Reference 5). The NRC staff accepted these model revisions as being within the scope of allowed changes, as determined by the significance test criteria (Reference 6).

In its review of the GESTAR amendment incorporating GSTRM into the GEH/GNF standard reload safety analysis process, the NRC staff noted that, because GSTRM is so fundamental to the GEH/GNF fuel analysis work, a transition will be needed from the Amendment 6 TEXICO approach to the Amendment 7 GSTRM approach. In its SE approving Amendment 7 to GESTAR, the NRC staff required that GE submit an implementation schedule within 3 weeks of the issuance of the SE (Reference 6). The NRC staff maintains that adequacy of the T-M analysis methodology remains fundamental to fuel analysis work.

5.3 Notification of 10 CFR Part 21 Evaluation

During its review of the T-M analyses performed for the Economic Simplified Boiling Water Reactor (ESBWR) (documented in LTR NEDC-33242P, “GE14 for ESBWR Fuel Rod Thermal-Mechanical Design Report,” issued January 2006 (Reference 7)), the NRC staff identified a concern regarding the thermal conductivity model in GSTRM. In particular, the NRC staff identified that the code did not account for changes in pellet thermal conductivity with exposure. The NRC staff transmitted this concern in Request for Additional Information (RAI) 4.8-16, issued August 2006 (Reference 8). In parallel, the NRC staff pursued the concern by requesting that GEH perform an evaluation of the deficiency in the GSTRM fuel thermal conductivity model, in accordance with 10 CFR Part 21, "Reporting of Defects and Noncompliance." Concurrent with the request, the NRC staff was completing a review of the applicability of the GEH/GNF reactor analysis methods to expanded operating domains (Reference 9). At the time, the NRC staff, in the SE, imposed a condition on the GEH/GNF methods stating that the findings of the NRC staff's review of the Part 21 evaluation would be applicable to expanded operating domain applications of those methods (Reference 10).

GEH/GNF provided the NRC staff with an assessment of the fuel thermal conductivity error in GSTRM through a notification of a Part 21 evaluation (hereafter referred to as the Part 21 notification) in January 2007 (Reference 11). The Part 21 notification concluded that the error was not reportable under 10 CFR Part 21, "Reporting of Defects and Noncompliance." However, the NRC staff review identified [ ].

The Part 21 notification states that the GSTRM code under-predicts the fuel centerline temperature above approximately [ ] and that this may result in [ ]. According to the "Summary of General Electric Technical Position on GESTR Fission Gas Release Model and Gadolinia Concentration," provided in Reference 2, the "fission gas release model must represent a 'best fit' of the fission gas release data. 'Best fit' in this sense means no apparent bias relative to the data." Therefore, [ ].

The NRC staff documented this nonconservatism and imposed a penalty to the critical pressure for extended power uprate (EPU) and maximum extended load line limit analysis plus (MELLLA+) plants as part of the interim methods LTR (NEDC-33173P, "Application of GE Methods to Expanded Operating Domains," issued June 2006; hereafter referred to as the IMLTR) SE (Reference 12). This penalty required a reduction of the critical pressure criterion by 350 pounds per square inch (psi) to account for biases in the predicted rod internal pressure.

The critical pressure is the rod internal pressure at which the cladding creep rate exactly equals the fuel pellet expansion rate. For rod internal pressures above the critical pressure, the gap between the fuel pellet and the cladding may reopen. Should the gap reopen, the increased thermal resistance will result in higher fuel pellet temperatures, resulting in higher fission gas release. The increased fission gas release will degrade gap conductivity while increasing the rod internal pressure, thus increasing pellet temperature and widening the gas gap further. The onset of gap reopening results in a runaway process of increasing gap opening until cladding failure. This phenomenon is referred to as cladding liftoff. To account for nonconservatism in the GSTRM assessment of the pellet conductivity, the NRC staff imposed a penalty to the critical pressure acceptance criterion in GSTRM.

In January 2008, GEH issued a supplement to the Part 21 notification (hereafter referred to as Supplement 1) (Reference 13). Supplement 1 provided the results of sensitivity studies aimed at demonstrating that competing conservatisms in the overall GSTRM methodology were sufficient to compensate for the bias in the predicted critical pressure as a result of the error in the fuel thermal conductivity model. Based on these sensitivity studies, the NRC staff drafted a revision to the IMLTR SE (Reference 14). The NRC staff findings were limited to current fuel designs with a maximum linear heat generation rate (LHGR) of 13.4 kilowatts per foot (kW/ft). The NRC staff never formally issued the revised IMLTR SE.

In August 2008, GEH issued a second supplement to the Part 21 notification (hereafter referred to as Supplement 2) (Reference 15). In Supplement 2, GEH provided similar sensitivity studies for the GNF2 fuel design. At the time Supplement 2 was issued, the GNF2 fuel design had not been deployed at the full reload level—only as lead test assemblies (Reference 16). The NRC staff reviewed Supplement 2 and determined that the magnitude of the error was exacerbated for higher rod powers. The GNF2 LHGR limit of [ ] the value of 13.4 kW/ft for the GE14 fuel design. On the basis of these sensitivity studies and its findings, the NRC staff imposed a burnup limit on the GNF2 fuel design; a revision to the GESTAR II licensing methodology (Reference 17) captured this burnup limit.

In February 2009, the NRC staff issued a reassessment of Supplement 1 (Reference 18). This reassessment provides the basis for the NRC staff's withdrawal of the revised SE before its formal issuance. The withdrawal of the revised IMLTR SE leaves the 350-psi critical pressure penalty in place in the approving SE for the IMLTR. Therefore, current IMLTR plants and future EPU or MELLLA+ applicants must adjust the critical pressure according to the 350-psi penalty to account for the fuel thermal conductivity error when using GSTRM to perform T-M operating limit analyses.

5.4 Interim Methods for Expanded Operating Domains

The IMLTR SE imposes a penalty to the critical pressure acceptance criterion to account for errors in the predicted fission gas release stemming from biases in the fuel pellet conductivity. This penalty is applicable to all future EPU and MELLLA+ applicants. To address the NRC staff penalty on a generic basis, GEH/GNF revised the GE14 GESTAR II compliance report. The compliance report serves as the basis for the acceptance of the use of the fuel product within the framework of the GESTAR II reload licensing methodology. Section 5.1 of this document provides additional descriptive details of the GESTAR II reload licensing methodology.

On April 9, 2009, GNF submitted a revision to the GE14 GESTAR II compliance report (Reference 19). The report provides an appendix that includes revised thermal-mechanical operating limits (TMOLs). These limits were developed using the GSTRM methodology with a 350-psi penalty imposed on the critical pressure. The GESTAR II process allows GNF to make significant design changes without specific NRC staff review. However, this process does include an advance notification clause, which affords the NRC staff the opportunity to perform an audit. The NRC staff conducted an audit of the GE14 GESTAR II compliance report revision between May 18, 2009 and May 20, 2009, at the GNF facility in Wilmington, NC. The NRC staff issued an audit report documenting the results of the compliance audit (Reference 20). The NRC staff found that the revised TMOL curves were acceptable and adequately incorporated the 350-psi critical pressure penalty for IMLTR plants.

However, the NRC staff notes that the concern is generic and may be applicable to plants operating at lower power levels but reaching higher fuel exposure during normal cycle operation. Additionally, the NRC staff did not consider the impact of the thermal conductivity error on downstream transient analysis codes that utilize data from GSTRM or, in other cases, models that are equivalent to the GSTRM models.

In accordance with Limitation 12 in the IMLTR SE, future EPU and MELLLA+ license amendment requests must utilize updated T-M methods (Reference 10). By letter dated February 27, 2009, GNF committed to update the codes to incorporate updated T-M models in the complete code system once approved by the NRC staff (Reference 21).

5.5 PRIME

In January 2007, GEH submitted the PRIME T-M methodology for NRC review and approval (References 22, 23, and 24). The PRIME T-M methodology incorporates an exposure- and gadolinia-dependent fuel thermal conductivity model. GEH also presented this fuel thermal conductivity model to the NRC staff during its review of TRACG04 for AOO and anticipated transient without scram (ATWS) overpressure transient analyses in 2006 (NEDC-32906P, Supplement 3, “Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients,” issued May 2006; hereafter referred to as the Migration LTR) (Reference 25).

The PRIME model incorporates modern HBWR experimental data that encompasses higher fuel rod burnup data. PRIME also incorporates a model for fuel thermal conductivity that explicitly accounts for the impact of gadolinia concentration and fuel exposure. Additionally, the TRACG04 code is capable of accepting fuel rod parameters from PRIME, such as gas gap composition, and includes an internal option to utilize thermal conductivity models that are consistent with the PRIME model (Reference 25). Therefore, during its review of PRIME, the NRC staff requested that GEH/GNF evaluate the effect of using the legacy GSTRM fuel thermal performance models in transient and accident codes relative to using the modern PRIME models. The NRC staff requested this information in RAI 39 related to the PRIME review (hereafter referred to as PRIME RAI 39).

By letter dated February 27, 2009, GEH committed to supplement the IMLTR by describing the implementation of the PRIME code models in the downstream safety analysis codes. The supplement (1) identified the specific codes and internal models to be upgraded, (2) provided any adaptations of the PRIME models to support the interface with each affected code, and (3) described the software testing and acceptance criteria to determine acceptable implementation (Reference 21). This supplement was submitted by GEH in July 2009 and approved by the NRC staff by letter dated September 12, 2011.

6 CODE SYSTEM OVERVIEW AND INTERFACES

The NRC staff has reviewed the individual codes that comprise the GEH/GNF suite of safety analysis methods. This section provides an overview of the individual codes and the interfaces between codes in the overall system used to evaluate steady-state operation, transients, and accidents for BWRs. Figure 1 provides a process diagram that depicts the interfaces between different analysis codes in the overall GEH/GNF transient analysis methodology, specifically TGBLA, GSTRM, PANACEA, ODYN, ISCOR, and TASC (Reference 29). Additional codes utilized by GEH/GNF to perform safety analyses include TRACG, ODYSY, SAFER, LAMB, CORECOOL, and STEMP (Reference 1).

6.1 TGBLA

At the onset of core analysis, calculations are performed to generate cross-section data for use in the core simulator code, PANACEA. These cross sections are generated using the TGBLA lattice physics code. The current standard production method is TGBLA06 in the GEH/GNF reload safety analysis process. References 26 and 27 describe the TGBLA06 methodology.

TGBLA06 utilizes a collision probability method to calculate fuel pin flux distributions and couples this calculation to a two-dimensional neutron diffusion equation to determine the detailed flux distribution within a two-dimensional fuel lattice. These calculations are performed using a high energy resolution neutron cross-section library.

The detailed flux spectrum and distribution are used to condense the cross section to three-group nuclear parameters. Specifically, TGBLA06 calculates the infinite lattice eigenvalue, migration area, diffusion coefficient, fast fission infinite migration area correction factor, infinite pin peaking factors, and flux discontinuity factors.

The TGBLA06 code performs depletion analyses of lattices under various control and void conditions. Depletion and branch case analyses are performed to develop a matrix of nuclear parameter results for various nodal state conditions. These results are used to correlate the nuclear parameters as a function of various nodal conditions, such as relative water density, control state, and exposure. These correlated parametric fits are employed in downstream nuclear design calculations.

6.2 GSTRM

GSTRM is a fuel T-M performance code. The current standard production method uses GSTRM07 in the GEH/GNF reload safety analysis process. References 2 and 5 describe GSTRM.

The GSTRM code predicts fuel rod thermal and mechanical performance for variable operating power histories. GSTRM was developed to predict the thermal and mechanical performance of BWR fuel rods. The development of GSTRM began with the GEGAP-III thermal model as its foundation. In addition to the effects of fuel and cladding thermal expansion, as well as fuel irradiation swelling, densification, relocation, and FGR addressed by GEGAP-III, the GSTRM model includes the effects of fuel-cladding axial slip, cladding creepdown and axial growth, irradiation hardening and thermal annealing, pellet and cladding plasticity, work hardening, creep and relaxation, and pellet hot pressing. The GSTRM fuel irradiation swelling, relocation, and fission gas release models have been imposed over the respective models in GEGAP-III.

GSTRM performs several thermal and mechanical calculations by considering a series of calculational problems in which time and space are considered invariant. The code also considers a set of nested iterative calculations of loops in which the fuel and cladding temperature, hot gap size, and rod internal pressure are determined in sequence. In addition, GSTRM provides the capability of analyzing bimetallic cladding consisting of an outer layer of Zircaloy and an inner layer of zirconium or barrier cladding. The mechanical analysis of the cladding therefore may assume two cladding ring elements within each axial node.

GSTRM is used to perform steady-state safety analyses by calculating the fuel temperatures and cladding strain experienced by fuel rods operating at limited power histories. These are compared against applicable regulatory criteria to ensure that cladding failures are precluded as a result of normal operation.

GSTRM is also employed to provide inputs to downstream safety analyses. In particular, GSTRM generates files that are used to characterize the fuel thermal heat conduction properties of the fuel to transient analysis codes. GSTRM is also used in an integral sense in emergency core cooling system (ECCS) LOCA evaluations to evaluate the transient thermal and mechanical performance of the fuel under accident conditions.

6.3 PANACEA

PANACEA is the core simulator code in the GEH/GNF code system. This code is used to predict core power distribution and eigenvalues to perform nuclear design analyses. The current standard production method is PANAC11. References 27 and 28 describe the PANAC11 methodology.

The nuclear model in PANAC11 is a static, 1.5-group, coarse mesh, nodal diffusion model. The nuclear model begins with three-group theory. The three-group equation is collapsed to a 1.5-group equation by assuming that each group has the same buckling. For each node, the 1.5-group equation is integrated and solved. A piecewise linear approach is used to determine the nodal flux in terms of the six surface currents. Current continuity and nodal diffusion coefficients are used to eliminate the surface currents and solve for the nodal flux in terms of the neighboring nodal fluxes.

The integrated surface currents are incorporated into the nodal spectral collapsing to account for spectrum hardening or softening as a result of neutronic coupling between nodes. The ratio of the infinite thermal to fast flux is corrected according to the integrated neutron balance for each group.

The node size is selected to account for the nuclear coupling between nodes as it relates to neutron transport. In general, the mean free path for a thermal neutron is very short, so the nodal size is selected based on the mean free path for fast neutrons and is about 6 inches.

Aside from the solution for the flux, various feedback mechanisms must be accounted for within the nuclear model to determine nodal nuclear parameters. These include void effects, Doppler effects, exposure effects, control rod effects, xenon effects, and reflector effects.

The thermal-hydraulic model in PANAC11 is relatively simplistic compared to the more detailed thermal-hydraulic models in transient and accident analysis methods. The primary purpose of the thermal-hydraulic model capabilities in PANAC11 is to predict channel flow distribution and void distribution within the core. Based on this limited application, simplifications are acceptable to maintain sufficient accuracy in the predictive capability of the code commensurate with its application. The thermal-hydraulic model includes several empirical correlations to describe heat transfer and interphasic phenomena (such as slip).

Key outputs of the PANAC11 code include the core power distribution, eigenvalue, control blade worths, shutdown margin, bundle critical power ratio, local LHGRs, and collapsed nodal cross sections for downstream transient calculations.

6.4 ODYN/ISCOR/TASC and STEMP

Transient calculations are performed using the ODYN/ISCOR/TASC code system. Figure 1 depicts the relationship between these codes (Reference 29). ODYN is used to predict the transient system response to anticipated transients (Reference 28). The system response is coupled with the ISCOR code to determine hot channel flow distributions. TASC simulates the transient hot channel thermal-hydraulic conditions based on ODYN/ISCOR output to determine the transient critical power ratio. In this sense, ISCOR and TASC are supporting codes for the ODYN transient analysis methodology to translate the transient response into the hot channel critical power ratio during the transient evolution.

The overall system model in the ODYN code consists of a one-dimensional representation of the reactor core and the recirculation and control system models. These two models are coupled to each other. A steady-state initialization is made and the parameters for the transient are then calculated.

First, the recirculation and control system is solved for the steady-state conditions. Some of the initial conditions are input and may be plant unique. Other initial hydraulic values, such as core pressure drop and bypass flow fraction, which are also input to the steady-state recirculation and control model, are calculated elsewhere. These parameters are calculated in the steady-state, multichannel core code. Using all of these input values, the steady-state recirculation and control model calculates the remaining hydraulic parameters in the plant. The steady-state initialization in the recirculation and control model provides the loop pressure drop, core exit pressure, core inlet flow, and enthalpy to the one-dimensional reactor core model. The reactor core model uses these values to calculate the neutron kinetics, thermal hydraulics, and fuel parameters for the steady-state conditions.

The neutronics model calculates the steady-state axial power distribution. The model uses cross-section fits obtained from radially averaged PANACEA results. The fits are such that the axial power in the one-dimensional model is required to yield the same axial behavior as in the three-dimensional PANAC11 solution. The steady-state thermal hydraulic solution permits the calculation of the steady-state fuel temperature distribution.

During the transient, the recirculation and control system model calculates the time derivatives. At the end of the time step, the recirculation and control system model supplies the new external boundary conditions to the reactor core model. The reactor core model calculates the new neutron flux, thermal-hydraulic parameters, and fuel temperatures. It also provides reactor core exit quality, flow, and pressure as input to the recirculation and control system model. The recirculation and control system model calculates the loop pressure drop, and the reactor core model calculates the core pressure drop. These pressure drops are compared. If they are not equal within a certain limit, the recirculation and control system model derivatives are modified and the time step calculations are repeated.

The recirculation system is modeled by solving the mass, energy, and momentum conservation equations for the steamline, reactor vessel, and recirculation loop components, which include jet pumps and recirculation pumps and their associated piping. The control system is modeled as a series of connected gains, filters, integrators, and nonlinearities (limiters and function generators). The control system output is valve position and thus flow control. The one-dimensional core model comprises equations describing the neutron kinetics, thermal hydraulics, and heat transfer behavior of the core.

Major assumptions used in the modeling of the recirculation system are as follows:

(1) Pressure variations in the system are described with 10 nodes. One node is used for the reactor inlet, another node is used for the reactor vessel dome, and the remaining eight nodes are used to describe the behavior of the steamline.

(2) Liquid and vapor mass volume balances are used to predict the reactor vessel water level changes.

(3) The recirculation loop model can simulate any combination of multiloop systems. The entire recirculation loop is assumed to be subcooled and incompressible.

(4) Steam in the steamline is treated as single phase flow. Condensation of steam in the steamline is precluded during the transient.

Major assumptions used in the reactor core model are as follows:

(1) A one-dimensional neutron kinetics model is assumed. The neutron flux varies axially with time. One energy group diffusion theory and six delayed neutron groups are used. Decay heat is modeled using a simple exponential decay heat model. The one-dimensional neutron diffusion parameters are obtained by collapsing the parameters obtained from PANAC11.

(2) A single active, heated channel represents the core average conditions and another single channel represents the core bypass. A five-equation model representing mass and energy conservation for the liquid and vapor and the mixture momentum conservation is used to calculate the core thermal-hydraulic behavior.

(3) Heat transfer to the moderator and fuel temperatures are calculated using an average fuel and cladding model at each axial location of the core. The gap conductance is an input parameter which may vary axially in time. The conduction parameters are temperature dependent. A radially uniform (flat) power distribution is assumed in the fuel rods.

The primary output of the ODYN/ISCOR/TASC analysis is the transient change in critical power ratio. The critical power ratio outputs are used to establish operating limits that limit transition boiling in the reactor as a result of anticipated transients.

Ancillary to the critical power calculations, ODYN calculates the nodal power histories during the transient. These nodal power histories are used to determine the T-M performance of the fuel rods during the anticipated transient. Generally, though not in all cases, the transient LHGRs are compared against limits established by GSTRM to demonstrate acceptable performance relative to the thermal overpower and mechanical overpower specified acceptable fuel design limits (SAFDLs). Figure 1 depicts this step in the analysis procedure on the far right of the process diagram.

ODYN is also used to perform analyses of ATWS cases, including simulation of the phases of pressurization, natural circulation, and boration. In these cases, ODYN is coupled with STEMP, a simple code that is used to evaluate suppression pool temperature during ATWS events.

6.5 TRACG

In 2001, the NRC staff approved the use of TRACG02 as an alternative calculational method to ODYN/ISCOR/TASC for the evaluation of AOOs (References 30, 31, 32, and 59). More recently, the NRC staff has conducted a review of the TRACG04 methodology (References 25 and 33). The TRACG04 method is based on the TRACG02 method with improvements in the neutron kinetics model and various closure relationships.

TRACG is based on a multidimensional, two-fluid model for the reactor thermal hydraulics and a three-dimensional neutron kinetics model for the reactor core. The two-fluid model used for the thermal hydraulics in TRACG is fundamentally the same as the basic two-fluid model in TRAC-PF1 and TRAC-BF1. The two-fluid model solves the conservation equations for mass, momentum, and energy for the gas and the liquid phases. TRACG does not include any assumptions about thermal or mechanical equilibrium between the phases. The gas phase may consist of a mixture of steam and noncondensable gases, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multidimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum, and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas - liquid interface, as well as at the wall. The constitutive correlations are dependent on the flow regime and are determined based on a single flow regime map, which is used consistently throughout the code.

In addition to the basic thermal-hydraulic models, TRACG also contains a set of component models for BWR components, such as recirculation pumps, jet pumps, fuel channels, steam separators, and dryers. Furthermore, TRACG contains a control system model capable of simulating the major BWR control systems, such as the pressure, level, and recirculation flow control systems.

The three-dimensional kinetics model is consistent with PANACEA. It solves a modified one-group diffusion model with six delayed neutron precursor groups. Feedback is provided from the thermal-hydraulic model for moderator density, fuel temperature, boron concentration, and control rod position. A major difference between TRACG02 and TRACG04 is that the TRACG02 kinetics model is based on PANAC10, while the TRACG04 kinetics model is based on PANAC11.

When evaluating the fuel rod T-M performance during transient conditions, TRACG relies on data supplied by upstream GSTRM calculations. It should be noted that TRACG04 has the capability of utilizing either PRIME or GSTRM calculations with self-consistent models in the transient and accident analyses performed using TRACG. Upstream T-M analyses must be performed to provide TRACG with fuel files. These files provide details of the gas gap composition and other parameters that allow TRACG to simulate the transient thermal heat conduction through the fuel pellet, gap, and cladding.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, pump, valve, tee, channel, jet pump, steam separator, heat exchanger, and vessel components. System simulations are constructed using these components as building blocks. Any number of these components may be combined.

The code input specifies the number of components and their interaction, as well as the detail in each component. TRACG consequently has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete BWR plants. TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests, and full-scale BWR plant data.

The wrapup file is used to initialize TRACG analyses to the PANACEA calculated steady-state conditions. During the steady-state initializing calculation with TRACG, updates to the core power distribution are disabled such that TRACG converges on a thermal-hydraulic condition that matches the PANACEA wrapup file power distribution. The wrapup file contains nuclear parameters for each neutronic node. Each neutronic node is assigned to thermal-hydraulic channels through user specification and specific TRACG channel grouping. The TRACG three-dimensional kinetics model is based on the same neutronic nodalization as present in PANACEA.

In the initialization process, there are several differences between the TRACG thermal-hydraulic model and the PANAC11 model. In addition, the nodalization for the neutronic model is not the same as the TRACG thermal-hydraulic model. Because of these differences, the TRACG initialization process to develop the steady-state condition for stability evaluation employs a means to adjust the neutronic model to accommodate the steady-state thermal-hydraulic solution.

PANACEA calculations are performed such that the neutronic solution is for a predetermined hot critical eigenvalue that often differs from unity to account for modeling biases. The hot critical eigenvalue is taken into account by adjusting the TRACG-predicted eigenvalue with the predetermined hot critical eigenvalue for PANACEA. The static effective multiplication factor is the same as the hot critical eigenvalue employed in the cycle analyses using PANACEA. This allows the TRACG steady-state solution to converge to the same eigenvalue as PANACEA.

The nodal accommodation is performed by offsetting the nodal relative water densities and fuel temperatures on a nodal basis to match the TRACG results. The temperature offset is performed by using a difference in the TRACG-calculated transient temperature and the TRACG-calculated steady-state temperature with a Doppler reactivity coefficient predicted by PANACEA.

The primary outputs of the TRACG code are similar to outputs from the ODYN/ISCOR/TASC code system. In particular, TRACG is used to evaluate the transient critical power ratio and transient LHGRs throughout the core. These calculations are performed to establish operational limits to ensure that SAFDLs are met on a cycle-specific basis.

TRACG02 and TRACG04 have also been approved to perform various other calculations. For instance, TRACG02 has been approved to calculate the change in critical power ratio per initial critical power ratio (DCPR/ICPR) versus oscillation magnitude (DIVOM) slopes to support licensing activities for plants referencing the Boiling Water Reactor Owners Group (BWROG) long-term stability solution Option III (Reference 30). Other applications include American Society of Mechanical Engineers (ASME) overpressure analysis (Reference 33), ATWS overpressure analysis (Reference 31), and, in the case of new reactor applications, LOCA analysis (References 34 and 35).

6.6 ODYSY

Stability calculations other than DIVOM analyses are performed using the ODYSY code. The current standard production version of ODYSY is ODYSY05 (Reference 37). These stability calculations are performed to assess the margin to the onset of instability by assessing the reactor decay ratios for various instability modes against applicable criteria. References 36 and 37 provide the ODYSY application methodology.

ODYSY is a best-estimate code which incorporates a linearized, small perturbation, frequency domain model of the reactor core and associated coolant circulation system. The program may be used to predict hydrodynamic stability for both a single channel and a full reactor core. It will predict both corewide, mode-coupled thermal-hydraulic and reactor kinetic instabilities and single channel thermal-hydraulic instabilities.

The ODYSY code is based on the ODYN code which uses a five-equation hydraulic solver and a one-dimensional neutron kinetics model extended to multiple channels. ODYSY has axial varying void and Doppler reactivity feedback and has flexibility in the fuel rod modeling to accommodate axial variations in fuel bundle geometry. The axial variation capability makes the code ideal for evaluating the stability of advanced fuel designs which have axial varying geometry.

ODYSY calculates corewide and channel decay ratios by evaluating the open loop transfer function of the reactor system. This is accomplished by taking a Laplace transform of the governing equations and performing the evaluation in the frequency domain. ODYSY has models of both the vessel and the reactor recirculation loops and can handle the geometrical features of modern fuel.

6.7 SAFER-GESTR

The purpose of the SAFER code is to calculate long-term reactor vessel inventory and peak cladding temperature (PCT) for LOCA and loss-of-inventory events. SAFER is intended for use with the GSTRM code (SAFER-GESTR) (Reference 2). SAFER builds directly on the SAFE and REFLOOD codes and retains many of these models' features. Major modifications are in the areas of counter current flow limiting models, core heat transfer models, bypass leakage models, and the nodal representation of the reactor vessel. SAFER assumes thermodynamic equilibrium in each region. Hydraulic and heat transfer modeling is essentially one-dimensional, and fluid properties are calculated at a mean thermodynamic pressure.

For the purpose of inventory calculations, the reactor vessel is divided into eight regions: (1) the lower plenum, (2) the volume inside the control rod guide tubes, (3) the active core region, (4) the core bypass, (5) the upper plenum (including the separator standpipes), (6) the lower downcomer below the feedwater sparger and outside the jet pumps, (7) the upper downcomer (above the feedwater sparger), and (8) the steam dome. A single hot fuel assembly is also modeled as a separate region for the purpose of calculating PCT. Mass and energy balances are performed for each region and for the entire vessel. A pressure rate is derived from the constraint that the total vessel volume is constant. Vapor flow from a region is modeled by a drift flux correlation or a bubble rise model. The void fraction profile is assumed to be a linear function of the mean void fraction in a given saturated region.

A modified Wallis correlation is used to model counter current flow limits (CCFLs). Back flow leakage from the bypass region is modeled as a mechanism for filling the lower plenum when CCFLs restrict liquid downflow into the core. Reactor pressure boundary break flows are modeled as the minimum of critical flow or Bernoulli flow with a loss coefficient of unity. Overall loop momentum equations are solved for the two loops through the jet pumps and the core. Each loop is between the steam dome, through the downcomer, through a bank of jet pumps, into the lower plenum, and back up through the core and upper plenum to the steam dome. SAFER uses several special regional models. The upper plenum model incorporates features such as core spray distribution, non-uniform mixing, and liquid entrainment resulting from core exit vapor flow. The core bypass region is divided into two parts for CCFL calculations. The core region is divided into seven subregions for a more accurate estimate of the axial void distribution. Mass and energy balances are also performed on each subregion of the core. A hot fuel assembly model is included to calculate the fluid inventory and PCT in a high power bundle. Flows in and out of the hot channel are calculated by imposing on the hot channel the plenum-to-plenum pressure drop taken from the average core calculation.

The fuel rod and cladding transient temperature response is determined by solving the transient heat conduction equation in cylindrical coordinates assuming no axial or circumferential heat conduction. A maximum of 10 equally spaced radial nodes and 5 axial nodes are used. The reactor pressure vessel (RPV) thermal response is simulated by considering four lumped parameter heat slabs characterized by specified constant values for thermal capacitance and thermal resistance. The vessel internals are simulated with six lumped parameter heat slabs. Surface heat transfer coefficients for these slabs are specified as a function of void fraction for each hydraulic region contacted by the slab. The heat source is the sum of contributions from decay heat and metal-water reactions. Gamma smearing is also modeled. The metal-water reaction heat source assumes that a continuous supply of water or steam is available and that no energy is required to bring the reactants to the reaction temperature.

Heat transfer from the fuel rods to the coolant is calculated using realistic values for heat transfer coefficients based upon different heat transfer regimes. These include nucleate boiling, forced flow film boiling, pool film boiling, and transition boiling. Nucleate boiling heat transfer coefficients are modeled as a function of coolant void fraction. The applicable heat transfer regime is determined from a logic which examines the coolant quality, the cladding superheat, and the critical heat flux. The model also considers steam cooling, core spray heat transfer, and radiation heat transfer.

The heat transfer models are applied to different characteristic fuel rods to obtain the fuel rod temperature distribution. The temperature distribution for the average power rod in the average power bundle with core average water inventory is used in the system mass and energy balance calculations. The maximum cladding temperature is calculated for the hottest rod in the average power bundle in both the central and peripheral core regions. The corewide PCT is determined from the temperature response of the hottest rod in the hot channel.

The application methodology consists of three essential parts. First, a limiting case LOCA is determined by applying realistic analytical models across the entire break spectrum. Second, the limiting case LOCA is analyzed with a licensing model which incorporates all of the required features of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities." This determines the licensing-basis PCT. Third, a statistically derived upper bound PCT is calculated to demonstrate the conservatism of the licensing-basis PCT.

The NRC originally approved the application methodology for SAFER-GESTR in 1984 for BWR/3–6 designs or plants with jet pumps. For application to BWR/2 plants, GE submitted a revised SAFER model for NRC review and approval (Reference 38). The revised SAFER model incorporates the CORECOOL code to calculate PCT under spray cooling conditions during LOCAs. Although SAFER includes this capability, the simplified heat transfer model of the SAFER code is less rigorous than that used in CORECOOL. Therefore, certain applications which require more realistic PCT predictions use the CORECOOL code.

The revised SAFER code changes the hydraulic model nodalization, CCFL correlation, and the overall momentum equations. GE made changes to nodalization and momentum equations specifically to account for the effect of the external loops for the BWR/2 plant designs.

The revised SAFER model adds two regions to the reactor system model, an intact external recirculation loop and a broken external recirculation loop. These two regions are important for non jet-pump reactors for modeling LOCAs. The revised SAFER refines the core region from 7 subregions to 12 subregions for a more detailed calculation of the void fraction profile in the core.

GE modified the total momentum equations to reflect the addition of the new regions. The methodology used to calculate the effect of recirculation loops is similar to that used in the NRC-approved LAMB code.

Since BWR/2 plants include an isolation condenser, the revised SAFER code includes an external flow model to calculate the transient performance of an isolation condenser. An isolation condenser consists of a heat exchanger and inlet line from the steam dome and return line connected to the recirculation loop. This system removes steam from the steam dome, condenses it on the tube side of the heat exchanger, and returns the condensate back to the reactor. The system functions by natural convection. The isolation condenser model is simple and uses classical heat transfer models. The NRC staff approved the updates to the SAFER code for application to non-jet-pump plants in 1987 (Reference 38).

GEH/GNF submitted additional model improvements to the NRC in 2000 (Reference 39). These model improvements include refinements to the jet pump entrainment and two-phase leakage flow. SAFER04 captures these model refinements. The suite of codes used to perform the GEH/GNF ECCS-LOCA evaluations include LAMB for short-term system blowdown, TASC for short-term hot channel heat transfer, SAFER for long-term system inventory, CORECOOL for fuel rod heatup, and GSTRM for fuel rod modeling. Figure 2, taken from Reference 1, depicts the interfaces between these codes.

6.8 Special Analyses

Special analyses refer to the calculations of special events, primarily ATWS and station blackout (SBO), which are considered beyond the design basis. To perform these evaluations, GEH/GNF relies on computer codes to simulate the expected plant behavior in response to these events.

6.8.1 Anticipated Transients without Scram

An ATWS event is a postulated transient with a common-mode failure of the reactor protection system (RPS), such that the scram function of the RPS is not fulfilled. The NRC has previously concluded that the likelihood of an inability to scram was exceedingly low; therefore, the agency has classified ATWS as a beyond design-basis accident. In 1984, the NRC issued 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants” (known as the ATWS rule) requiring that plants incorporate mitigating measures, including plant modifications, to ensure that the consequences from a postulated ATWS event were acceptable.

GE submitted LTR NEDE-24222, “Assessment of BWR Mitigation of ATWS,” in December 1979 to demonstrate the effectiveness of the measures in mitigating the consequences of ATWS on BWR plants (Reference 40). GE performed these evaluations using the REDY code, and they form the basis for the generic acceptability of the mitigating measures provided in 10 CFR 50.62 to ensure safety for the fleet of BWR plants.

Consistent with these generic evaluations performed in the late 1970s, GE provided more recent guidance on ATWS analysis for NRC staff review in LTR NEDC-32424P-A, “Generic Guidelines for General Electric Boiling Water Reactor Extended Power Uprate,” issued February 1999 (Reference 41) (hereafter referred to as ELTR1) for EPU conditions. ELTR1 provides guidance for performing either plant-specific or generic ATWS licensing analysis. The intent of these EPU analyses is to demonstrate continued compliance with the requirements of 10 CFR 50.62 to ensure acceptable consequences.

The NEDE-24222 basis states that analyses were performed on a generic basis to demonstrate that the mitigation capabilities were sufficient to meet the following criteria:

(1) The reactor coolant pressure boundary shall remain below emergency pressure limits.

(2) The containment pressure shall remain below design limit. The suppression pool temperature shall remain below local saturation temperature limits.

(3) A coolable core geometry shall be maintained.

(4) Radiological releases shall be maintained within 10 CFR Part 100, "Reactor Site Criteria," allowable limits.

(5) Equipment necessary to mitigate the postulated ATWS event shall be evaluated to provide a high degree of assurance (assurance of function) that it will function in the environment (pressure, temperature, humidity, radiation) predicted to occur as a result of the ATWS event.

ELTR1 provides commensurate criteria to those provided in NEDE-24222. These criteria include the following:

(1) peak vessel bottom pressure less than the ASME service level C limit of 1,500 pounds per square inch gauge (psig)

(2) maximum containment pressure and temperature lower than the design pressure and temperature of the containment structure

(3) PCT below the 2200-degree-Fahrenheit requirement of 10 CFR 50.46

(4) clad oxidation below the requirements of 10 CFR 50.46

The common practice is to perform detailed calculations for representative plants from each product line for the limiting ATWS scenarios.

As an example, the NRC staff reviewed the applicability of generic ATWS evaluations in its review of the constant pressure power uprate (CPPU) LTR (Reference 43), also known as the CLTR. In its review, the NRC staff required that plant-specific analyses confirm that the CPPU conditions meet the ATWS mitigation requirements defined in 10 CFR 50.62 because (1) an alternate rod insertion system was installed, (2) the boron injection capability was equivalent to 86 gallons per minute, and (c) an automatic ATWS-recirculation pump trip (ATWS-RPT) had been installed. Section L.3 of ELTR1 discusses the ATWS analyses and provides a generic evaluation of the following limiting ATWS events in terms of overpressure and suppression pool cooling:

- main steam isolation valve (MSIV) closure
- pressure regulator failure to open (PRFO)
- loss of offsite power (LOOP)
- inadvertent opening of a relief valve (IORV)

The ATWS analyses will be performed for a representative core design at the CPPU operating condition for the MSIV closure and PRFO events. The LOOP event will be analyzed only if it produces a significant loss in suppression pool cooling. The plant-specific ATWS analysis for CPPU will demonstrate that the ATWS acceptance criteria for peak vessel bottom pressure and peak suppression pool temperature are met. The CLTR dispositioned the other events and acceptance criteria in a generic manner.

Another example is the generic ATWS evaluations provided in LTR NEDC-32523P-A, “Generic Evaluations of General Electric Boiling Water Reactor Extended Power Uprate,” (hereafter referred to as ELTR2) (Reference 42) for BWR/3 plants. The transients that were analyzed include (1) MSIV closure, (2) PRFO, (3) LOOP, and (4) IORV.

Because plant-specific ATWS analyses are required for EPU applications, and current plant licensing bases reference generic ATWS analyses previously performed, the NRC staff considered the computational methods used to perform these calculations as part of this survey. The GEH/GNF methods used to evaluate ATWS events include the REDY and ODYN codes. The NRC staff also notes that NEDE-24222 and ELTR1 define the analysis acceptance criteria. These acceptance criteria are consistent with the criteria specified in the Section 15.8, “Anticipated Transients Without Scram,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants” (Reference 45) (hereafter referred to as the SRP) for ATWS evaluations.

6.8.1.1 REDY

REDY is a point kinetics transient code with integral thermal hydraulics (Reference 49). The NRC approved REDY to perform ATWS evaluations in 1979, and the code formed the basis for the generic ATWS evaluations presented in NEDE-24222. The REDY model for ATWS and transient applications includes several major simplifying assumptions. These include (1) the point kinetics approximation, which does not allow for transient evolution of the reactor power shape, (2) the integral momentum fluid dynamics approximation, which does not allow for nonequilibrium of the phases, and (3) expansion of the liquid in the downcomer, lower plenum, and core bypass due to flashing is not allowed.

During its review of the REDY ATWS analyses presented in NEDE-24222, the NRC staff also considered qualification of the REDY predictions to calculations performed using the more sophisticated ODYN code. In 1995, GE proposed to expand the application of ODYN to ATWS scenarios and to perform future calculations using ODYN in place of REDY (Reference 1). REDY is no longer in active use. However, the generic analyses provided in NEDE-24222 may be referenced.

6.8.1.2 ODYN

The ODYN code confers many benefits in terms of simulation accuracy relative to the REDY code. These include one-dimensional kinetics, separated flow models, and enhanced boron mixing and reactivity models.

The NRC staff has reviewed the ODYN computer code for application to BWR ATWS analyses. The NRC staff considered as part of its review the benchmarking and model description provided by GE in NEDO-24154-A (Reference 28). The NRC staff also performed a series of audit calculations to confirm the GE boron reactivity model and the adequacy of ODYN predictions as part of its review of ELTR2. Based upon this review, the NRC staff concluded that ODYN is acceptable for application to BWR ATWS calculations provided that the code is used within the restrictions discussed in Section 5.1.3 of NEDO-24154-A (Reference 28), including at EPU conditions. ODYN is in active use.

6.8.1.3 MELLLA+ Conditions

MELLLA+ operation adversely impacts the plant's ATWS analysis response because of operation at the higher rod line (References 55 and 50). ATWS-RPT initiated from the reduced core flow (CF) at EPU power state point (e.g., 120 percent of the originally licensed thermal power (OLTP) and 80 percent of the rated CF is less effective relative to operation at the higher power and flow conditions as an initial condition. By implementing MELLLA+, the allowable CF at rated power is further reduced. With operation at the higher 140-percent rod line, the reactor will settle at higher core power at natural recirculation, following a dual recirculation pump trip (2RPT), as compared to MELLLA without scram. The short-term peak vessel overpressure becomes higher, as compared to MELLLA, due to the reduced power reduction capability afforded by the ATWS-RPT.

In addition, since the reactor is operating at a higher rod line, after 2RPT, the reactor core will settle at higher powers relative to operation at MELLLA+. The higher reactor power at natural circulation increases the long-term heat load to the containment and results in higher suppression pool temperature. In its review of NEDC-33006P, "General Electric Boiling Water Reactor Maximum Extended Load Line Limit Analysis Plus," issued January 2002 (Reference 55) (hereafter referred to as the MELLLA+ LTR), the NRC staff determined that, for ATWS, the heat capacity temperature limit (HCTL) may be exceeded, at which point the operators are instructed to depressurize the reactor in accordance with the plant's emergency operating procedures (EOPs). The HCTL value is plant specific, and it depends roughly on the ratio of suppression pool volume to the reactor power. A typical value is approximately 160 degrees Fahrenheit.

However, the licensing ODYN code cannot model the depressurization or any ATWS water-level strategies, other than top of active fuel plus 5 feet (TAF+5). Therefore, the ODYN licensing calculation cannot model or simulate the actual plant conditions or operator actions, as delineated by the EOP for ATWS. ODYN has been shown to be conservative for peak pressure relative to TRACG. The NRC staff concluded that the plant-specific applications will include ATWS sensitivity analyses simulating the ATWS scenario consistent with the plant-specific ATWS EOPs, including water-level strategies employed at the plant, depressurization if the HCTL is reached, and associated operator actions and systems actuations.

Predictions of the consequences associated with emergency depressurization are inconclusive. The sensitivity analyses show that the reactor can achieve hot shutdown conditions after the depressurization. However, both the NRC staff and GEH calculations indicate a potential for recriticality. Some of the TRACG simulations performed by GEH indicate that, following the emergency depressurization, sufficient boron has been mixed into the core volume to maintain the reactor shutdown at the reduced pressure (approximately 100 psi) by the combined effect of the mixed boron and the void fraction generated by decay heat. For these TRACG depressurization calculations, the containment limits are satisfied. Fuel suffers dryout overheat due to core uncovery, with a more severe transient for the lower-water-level control strategies (e.g., top of active fuel minus 2 feet) than that associated with the TAF+5 strategy. The NRC staff concludes that recriticality after depressurization is not certain. It depends on plant and event-specific parameters, as well as assumed operator actions, such as reclosing some safety/relief valves (SRVs) after the pressure reaches the 50-psi target.

The plant-specific MELLLA+ applications will incorporate TRACG simulation following the EOPs, including depressurization, if the HCTL is reached.

The NRC staff approved the application of TRACG to ATWS instability in a generic review of the effectiveness of the mitigation actions (References 53 and 54). The NRC also reviewed and approved TRACG for the ATWS scenario up to the calculation of the peak pressures (Reference 31). In addition, in MFN 07-034, GEH committed to submit TRACG for ATWS application as the sole ATWS licensing evaluation code and replace ODYN as the ATWS licensing code with TRACG upon NRC review and approval (Reference 51). In its review of the MELLLA+ LTR, the NRC staff had performed a limited evaluation of the boron-mixing correlations modeled in TRACG (Reference 50). Since TRACG is a best-estimate code that can model the ATWS scenario with more fidelity, including all the required operator actions and water-level strategies, the NRC staff accepted the use of TRACG for performing the sensitivity analyses, in addition to the currently licensed ODYN code. However, plant-specific applications will continue to use ODYN to demonstrate that the plants can meet the ATWS acceptance criteria, including the peak vessel pressure.

6.8.2 Anticipated Transient without Scram with Instability

As a result of large power oscillations experienced at LaSalle Unit 2 following an inadvertent ATWS-RPT on March 19, 1988, the NRC staff initiated a reexamination of the characteristics and consequences of BWR instability under ATWS conditions (Reference 53). Limiting ATWS instability events are characterized by pressurization, failure of the control rods to insert, and large bypass capacity. Under these conditions large power oscillations are expected to develop and the NRC staff was concerned that thermal-hydraulic instability may exacerbate the consequences of a postulated ATWS.

6.8.2.1 Generic Assessment

The BWROG submitted LTRs NEDO-32047, “ATWS Rule Issues Relative to BWR Core Thermal-Hydraulic Stability,” dated June 1995, and NEDO-32164,“Mitigation of BWR Core Thermal-Hydraulic Instabilities in ATWS,” dated December 1992 (References 53 and 54, respectively), to provide a basis for appropriate mitigating actions to eliminate ATWS instability as an operational or regulatory concern. The BWROG contracted with GE to evaluate the characteristics and consequences of large power oscillations under ATWS conditions using the TRACG code. The NRC staff reviewed the changes to the emergency procedure guidelines (EPGs) that were proposed in the LTRs based on the TRACG studies.

The NRC staff approved these LTRs in 1994 on the basis of its acceptance of the TRACG code as an adequate tool for estimating qualitatively the global behavior of operating reactors during transients that may result in large power oscillations. The NRC staff also based its conclusions regarding the applicability and acceptability of the revised EPGs on the analyses that were performed for limiting ATWS conditions. These conditions were determined to occur for pressurization ATWS events (such as turbine trip) at a high power-to-flow ratio (such as points in an expanded operating domain above the rated power rod control line) with a large bypass capacity (100 percent).

The NRC staff approval of these LTRs, therefore, constitutes its acceptance of TRACG as a means to qualitatively evaluate the consequences of ATWS instability and to assess the effectiveness of mitigating actions in accordance with the EPGs. The analyses provided in the LTRs formed the basis for the generic NRC staff conclusions regarding the revised EPGs, however, the NRC staff notes that, subsequent to the NRC staff approval of NEDO-32407 and NEDO-32164, several plants have adopted further expanded operating domains, such as EPU. During its review of license amendment requests to implement EPU, the NRC staff has asked that licensees demonstrate the continued applicability of the generic TRACG evaluations to their specific EPU core and plant configuration in order to assure the NRC staff of the continued effectiveness of the mitigating actions in the EPGs.

Generally speaking these mitigating actions remain appropriate and applicable because the highest rod line for EPU operation remains identical to the highest rod line for MELLLA operation. Therefore, TRACG ATWS instability analyses are rarely required. However, the NRC staff has raised concerns regarding ATWS instability for the proposed MELLLA+ expanded operating domain in which the core power-to-flow ratio at the onset of an ATWS event may be even higher.

6.8.2.2 MELLLA+ Assessment

MELLLA+ operation adversely impacts BWR plants' ATWS instability response because of the operation at the higher rod line (approximately 140 percent) and, to some degree, the core design associated with operation of EPU/MELLLA+ for a 24-month cycle length (References 55 and 50).

The NRC staff review of the MELLLA+ LTR (Reference 55) indicates that, in principle, MELLLA+ operation affects ATWS stability. Operation at the minimum CF at EPU power levels (120-percent OLTP, 80-percent CF) results in a significantly higher power following a 2RPT than when operating at MELLLA at OLTP or EPU. This higher power makes the final power even larger after the feedwater cooldown period; thus the unstable power oscillations are enhanced under MELLLA+, and their consequences would be expected to be more severe. However, TRACG simulations performed by GEH demonstrated that the EPG mitigation actions are still effective in suppressing the oscillations during these ATWS events. The unstable power oscillations are allowed to grow to greater than 1,000 percent. Following one of the power excursions analyzed by GEH, the fuel dries out and fails to rewet.

The resulting temperature excursion is sufficiently large to compromise the integrity of the fuel. The corresponding nonisolation ATWS analysis following the prescribed EPG mitigation actions (e.g., immediate water-level reduction and boron injection) show that, for the particular reactor modeled, the power oscillations are adequately managed and the fuel integrity is not challenged. However, the NRC staff finds that the results of the ATWS instability analysis have a large sensitivity to particular reactor conditions. Therefore, a condition in the SE requires the evaluation of the effectiveness of the ATWS instability mitigation actions on a plant-specific basis for MELLLA+ plants (Reference 50).

On the basis of the NRC staff's previous review of the ability of TRACG to perform ATWS instability simulations and the commitment provided by GEH in letter MFN-07-034, the NRC staff expects that the required ATWS instability analyses for hypothetical MELLLA+ applicants will be performed using TRACG.

6.8.3 Station Blackout

The term "station blackout" refers to the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant. SBO therefore involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system, but not the loss of available ac power to buses fed by station batteries through inverters or the loss of power from "alternate ac sources." Based on 10 CFR 50.63, "Loss of All Alternating Current Power," all licensees and applicants are required to assess the capability of their plants to maintain adequate core cooling and appropriate containment integrity during an SBO and to have procedures to cope with such an event (Reference 57).

The specific regulatory criterion in 10 CFR 50.63 states that the reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of an SBO for the specified duration (References 57 and 46).

The postulated SBO event for a BWR results in scram, feedwater trip, turbine trip, and MSIV closure. The progression of the event results in automatic initiation of the high-pressure coolant injection (HPCI) system to regulate the RPV water level. During the coping period, steam is relieved to the suppression pool through the SRVs. Analyses are performed to ensure adequate core cooling and acceptable containment pressure and temperature relative to design limits. These analyses are performed using the approved SHEX containment long-term pressure and temperature analysis code (Reference 55).

During its review of the CLTR, the NRC staff considered the potential for operation at EPU conditions to affect the plant response during an SBO. The higher licensed power level will result in a higher decay heat load relative to those present at lower pre-EPU power levels. Therefore, EPU applicants are required to reanalyze the SBO event on a plant-specific basis (References 43 and 44).

7 SAFETY ANALYSIS IMPLICATIONS

During its review of the PRIME T-M methodology, the NRC staff issued RAI 39. PRIME RAI 39 requested that GEH/GNF evaluate the sensitivity of safety analyses performed using the historical GSTRM-based fuel T-M models in downstream transient and accident analysis codes after the NRC approved the PRIME method.

In February 2009, GEH committed to update the legacy suite of codes to incorporate the PRIME thermal models (Reference 21). These models account for the fuel thermal conductivity degradation effect. However, in the interim, the licensing evaluations performed by GEH/GNF will include several analyses performed using the historical GSTRM-based models. GEH has supplemented the IMLTR by describing the implementation of the PRIME models into the downstream safety analysis codes. The NRC staff has reviewed and approved this implementation plan (Supplement 4 to the IMLTR). The NRC staff intends to conduct an audit once the legacy codes have been updated to ensure consistency with the approved implementation plan. The NRC staff refers to the period of time between PRIME approval and the eventual update of the legacy methods as the interim process.

During its review of TRACG04 to perform transient calculations (Migration LTR), the NRC staff identified a concern with utilizing the PRIME thermal conductivity model in TRACG04 with gas gap conductance files based on the GSTRM code. This concern arises because the fuel thermal time constant is a strong function of the pellet thermal conductivity and the gas gap conductance. Combining the GSTRM gas gap conductance file, noting deficiencies in the GSTRM fuel conductivity model, may have an adverse impact on the efficacy of the safety analysis codes.

Therefore, the NRC staff requested that GEH use the TRACG04 code with both consistent PRIME and GSTRM inputs to assess the sensitivity of the safety analysis figures of merit. The TRACG04 code was selected to perform this sensitivity analysis in part because the code already includes a capability for utilizing the PRIME thermal conductivity model and in part because the NRC staff has reviewed various capabilities of TRACG to perform a wide variety of transient and safety analyses.

The NRC staff accepts the use of TRACG04 for this purpose because the physical basis for the TRACG04 models are significantly similar to that of models included in the other legacy codes (ODYN, SAFER, and others). Therefore, the NRC staff expects that the TRACG04 code, given that it has more detailed modeling capabilities, such as three-dimensional kinetics, will yield the most accurate assessment of the physical sensitivity of the transient and accident plant response to differences in the fuel thermal model.

The NRC staff’s review considered each safety analysis, including AOOs, overpressure transients, ATWSs, stability evaluations, and ECCS LOCA design-basis accident (DBA) analyses. References 58 and 52 provide these analyses. For each type of analysis, the NRC staff reviewed the sensitivity of the figures of merit to determine if the interim process results in nonconservatism in the safety analysis results.

The NRC staff relies on these sensitivity studies to provide insights into the effects of fuel thermal conductivity biases on safety analyses performed for the operating fleet of BWR plants. The NRC staff also considered relevant information from NRC staff confirmatory calculations, sensitivity studies provided in RAI responses, and relevant audit findings in its survey.

7.1 Steady-State

The steady-state core analysis codes include GSTRM and TGBLA06/PANAC11. These codes are used to assess the performance of any given core design against applicable T-M and nuclear acceptance criteria established in the SRP. In this section, the NRC staff evaluated the impact of fuel thermal conductivity biases on the efficacy of these steady-state codes to evaluate reactor performance relative to the acceptance criteria.

7.1.1 Thermal-Mechanical Criteria

General Design Criterion (GDC) 10, "Reactor Design" (see Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50) requires that SAFDLs are not exceeded during any condition of normal operation. To demonstrate compliance with GDC 10, fuel rod T-M design limits are established to ensure fuel rod integrity throughout its core lifetime along the licensed power-to-flow domain during normal steady-state operation and in the event of an AOO. Amendment 22 to the NRC-approved GEH licensing methodology GESTAR II specifies the T-M acceptance criteria for new fuel product lines. The LHGR limit is an exposure-dependent limit placed on the rod peak pin nodal power that ensures the integrity of the fuel cladding during normal steady-state operation and limits the initial heat generation rate during transient thermal and mechanical overpower conditions. The internal rod pressures during steady-state operation, the maximum fuel temperature, and the cladding strain during transients (e.g., AOOs) all affect the fuel integrity. The fuel T-M design criteria requires, in part, that:

(1) Loss of fuel rod mechanical integrity will not occur as a result of excessive cladding pressure loading.

The fuel rod internal pressure is limited so that the cladding creep out rate resulting from internal gas pressure during normal operation will not exceed the instantaneous fuel pellet cladding irradiation swelling rate. In establishing the LHGR limit, at each point of the exposure-dependent envelope, the fuel rod internal rod pressure required to cause the cladding to creep outward at a rate equal to the pellet irradiation swelling is determined.

The calculated internal rod pressures along the LHGR envelope are statistically treated so that there is assurance, with 95-percent confidence, that the fuel rod cladding creep rate will not exceed the pellet irradiation swelling rate.

(2) Loss of fuel rod mechanical integrity will not occur due to fuel melting.

The fuel rod is evaluated to ensure that fuel melting will not occur during normal operation.

During the design certification review for the ESBWR, the NRC staff discovered an issue in the GSTRM T-M calculations which supported the GE14 fuel designs. Specifically, the NRC staff discovered that GSTRM under-predicted the urania fuel temperature calculations as compared to both the FRAPCON-3 and the PRIME calculations for high exposures. GEH/GNF used PRIME for sensitivity analysis in response to NRC staff RAIs associated with the ESBWR. From the review of the ESBWR, the NRC staff attributed the observed differences primarily to the GSTRM urania fuel thermal conductivity model, which does not model the exposure dependency as does the other two codes.

In the NRC SE of the IMLTR, the NRC staff evaluated the conservatism in the GSTRM T-M methodology. Specifically, the confirmatory analyses performed in this review and the data provided in the RAI responses revealed that GSTRM appeared to under-predict the internal rod pressure calculation. During the review of the GSTRM T-M methodology in the IMLTR, the NRC staff established that additional GSTRM benchmarking data were needed for the current fuel designs as operated in the current core designs and operating strategies. GEH/GNF had committed to perform fission gas release (internal rod pressure) and exposure gamma scans in a response to the IMLTR RAI 9 (Reference 59).

The NRC staff requested that GEH/GNF initiate an evaluation under 10 CFR Part 21 to demonstrate the adequacy of the GSTRM T-M methodology in terms of the observed GSTRM under-predictions in fuel temperature and internal rod pressure, including its impact on the T-M fuel performance. In a letter dated January 21, 2007, GEH submitted the Part 21 notification that evaluated the GSTRM T-M methodology.

The IMLTR SE contains limitations and conditions that pertain to the performance of the T-M methodology. Specifically, Limitation 14 in Section 9.0 of the IMLTR SE states that the conclusions of the NRC staff assessment of information provided in the Part 21 notification will apply to the use of the GSTRM T-M methodology.

The NRC staff evaluated the Part 21 notification and the data provided in the T-M-related RAI responses from its review of the IMLTR and found that they were inadequate for verifying the adequacy of GSTRM-calculated fuel rod temperatures and rod pressures.

7.1.1.1 Internal Rod Pressure

The analysis of rod pressure, P_{rod} (this term, which was not used by GEH/GNF, is provided for clarity), for a given GEH/GNF fuel design application involves the GSTRM calculation of rod internal pressure to demonstrate that the application remains below the calculated $P_{critical}$ required by the SAFDL for rod pressure (i.e., the no-clad-liftoff (NCLO) criterion). GSTRM calculation of rod pressures relies on two internal models: the fission gas release and the internal rod void volume models. The fidelity of the internal rod pressure calculation depends on the accuracy of these two models. The $P_{critical}$ limit prevents cladding creep from exceeding the fuel swelling limit at operating power and is calculated utilizing the GSTRM fuel swelling and cladding creep models. The GESTAR II rod pressure analysis methodology requires that the ratio of $P_{rod}/P_{critical}$ remain below unity at a 95-percent confidence level.

GSTRM also appears to under-predict fission gas release and fuel rod pressures at the current end-of-life high burnup levels. In the Part 21 notification, GEH/GNF states that the dated database covered the range of current operating conditions. However, considering the database available at the time of the GSTRM approval suggests that the LHGR range may have been above the LHGR limits for current fuel designs only at low burnup levels and may be significantly below current LHGR limits at high burnup levels characteristic of current fuel designs. At high burnup levels, the LHGRs that GEH/GNF used to validate GSTRM were most likely very low (much lower than current LHGR limits for the present fuel designs at high burnup) and the uncertainties in fission gas release were also very high. Thus, the corresponding rod pressure uncertainties were also high.

The most recent industry fission gas release data at high burnups are either near or above the GEH/GNF LHGR limit and would be expected to provide a much better benchmarking database than the GSTRM 1984 verification data. In addition, [ ], while currently available data from Halden exceed 80 GWD/MTU.

[ ]. The NRC staff FRAPCON-3 confirmatory calculation shows that GSTRM under-predicts the internal rod pressure by 600 psi for both the nominal and the 95-percent confidence level. The lower prediction demonstrates that, although the internal rod pressure prediction meets the acceptance criteria in terms of $P_{critical}$, the adequacy of the GSTRM uncertainty treatment needs to be confirmed through measurement qualification data and corrected for licensing applications.

The fission gas release will be sensitive to the modeling of the fuel thermal conductivity for high exposures. At any given rod power, the fuel temperature is a function of the thermal conductivity and gap conductance. Overpredicting the fuel thermal conductivity at high exposure results in lower fuel temperatures, which in turn, results in under-predicting the degree of fission gas release.

[ ].

The NRC staff FRAPCON-3 confirmatory calculation determined a $P_{critical}$ value of 3,200 psi for the rod pressure limit (best estimate). This pressure limit ($P_{critical}$) is used to determine the maximum rod pressure where cladding creep equals fuel swelling consistent with the GEH/GNF criterion for rod pressure. [

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Therefore, the GSTRM code needs to be verified against fission gas release data typical of current fuel designs at LHGRs near the GEH/GNF LHGR limit at rod average burnups between 40 and 62 GWD/MTU. The fission gas release data used for verification needs to be provided to demonstrate that it is near, or bounds, the LHGR limits at high burnup based on current fuel designs as operated.

Recently, a separate NRC staff audit preformed for the ESBWR followed up on the GSTRM T-M methodology. The NRC staff reviewing the ESBWR design revised FRAPCON-3.3 confirmatory analyses and removed small differences in the input (e.g., stack densification and void volumes). Comparisons of the new FRAPCON confirmatory analyses to the GSTRM rod pressure calculation demonstrated that the GSTRM under-prediction may be less than the under-prediction of 500–600 psi established in the earlier FRAPCON confirmatory calculations. Therefore, the NRC staff finds that a decrease in the $P_{critical}$ margin from 500 to 350 psi is warranted, considering the results of the corrected FRAPCON calculation. This shows the difficulty of relying on code-to-code comparisons in validating the accuracy of calculational methodologies. The best approach to establishing the degree of GSTRM under-prediction and uncertainties is by a direct comparison of GSTRM internal rod pressure calculations to measurement data (fission gas release and void volume) comparable to design and maximum operating conditions for commercial fuel designs.

During the ESBWR audit, GEH/GNF also indicated that accounting for the “operating history” assumption in the derivation of the LHGR and the calculation of the internal rod pressures compensates for some of the nonconservatisms in the GSTRM internal rod pressure calculation with exposure. The assumption that the rod operates with bottom-, mid-, and top-peaked power shapes throughout the rod resident time in the core, such that it places a node of the rod at the LHGR limit, is sometimes referred to as the “operating history” assumption. This assumption increases the fission gas release, thus adding conservatism to the internal rod pressure calculations. However, the internal rod pressure calculations provided in the IMLTR RAI 9 response were based on consistent GSTRM/FRAPCON comparison in which both codes utilized the same operating history assumption.

7.1.1.2 Fuel Melting

In the Part 21 notification, GEH/GNF suggested that [

]. However, this assessment was based on old data obtained and used to validate GSTRM before 1984. Based on comparisons to the NRC code, FRAPCON-3, the GSTRM best-estimate fuel temperatures appear to be under-predicted, even at relatively low burnups. However, examination of the uncertainties applied to GSTRM fuel temperature calculations provided in the IMLTR RAI 9 response shows that, although the GSTRM best-estimate calculation underestimates the fuel centerline temperature, the conservative uncertainty treatment compensates for the under-prediction, resulting in bounding

95/95 fuel centerline temperatures (Reference 59). Therefore, the NRC staff concludes that the GSTRM fuel temperature calculation, with the 95-percent uncertainty treatment, is acceptable. This judgment is based on the uncertainties applied in the IMLTR RAI 9 fuel temperature calculation, which is based on NRC-approved uncertainty treatment methodology.

7.1.2 Nuclear Criteria

SRP Section 4.3, “Nuclear Design,” (Reference 47) specifies the nuclear design acceptance criteria, which are based on meeting the following relevant requirements from Appendix A to 10 CFR Part 50:

(1) GDC 10, “Reactor Design,” requires the reactor design (reactor core and associated reactor coolant, control, and protection systems) to ensure that the SAFDLS are not exceeded during any condition of normal operation, including AOOs

(2) GDC 11, “Reactor Inherent Protection,” requires a net negative prompt feedback coefficient in the power operating range

(3) GDC 12, “Suppression of Reactor Power Oscillations,” requires that power oscillations that can result in conditions exceeding SAFDLs not be possible or be reliably and readily detected and suppressed

(4) GDC 13, “Instrumentation and Control,” requires a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions

(5) GDC 20, “Protection System Functions,” requires, in part, a protection system that automatically initiates a rapid control rod insertion to ensure that SAFDLs are not exceeded as a result of AOOs

(6) GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” requires protection systems designed to ensure that SAFDLs are not exceeded for any single malfunction of the reactivity control systems

(7) GDC 26, “Reactivity Control System Redundancy and Capability,” requires, in part, two independent reactivity control systems of different design principles that are capable of holding the reactor subcritical under cold conditions

(8) GDC 27, “Combined Reactivity Control Systems Capability,” requires, in part, a control system designed to control reactivity changes during accident conditions in conjunction with poison addition by the ECCS

(9) GDC 28, “Reactivity Limits,” requires, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding

GEH/GNF performs steady-state calculations using the nuclear design methods to demonstrate compliance with several of the GDC specified in SRP Section 4.3. The NRC staff reviewed the TGBLA06/PANAC11 methods used to perform these steady-state calculations for evaluating compliance with the relevant GDC.

This section of the survey deals specifically with steady-state calculations. Therefore, this section does not address all of the nuclear design GDC.

Compliance with GDC 10 and 20 is demonstrated by performing transient evaluations to ensure that SAFDLs are not exceeded as a result of AOOs. Steady-state calculations are also performed to demonstrate that normal operation does not result in the SAFDLs being exceeded; however, transient conditions are generally most limiting in terms of the relevant SAFDLs. Section 7.2 of this survey addresses these calculations.

Compliance with GDC 12 is demonstrated by performing stability analyses. Section 7.3 of this survey addresses these calculations.

Compliance with GDC 28 is demonstrated by analysis of the consequences of a postulated control rod drop accident (CRDA), which is also a reactivity initiated accident (RIA). Section 7.4.2 of this survey addresses these calculations.

7.1.2.1 Inherent Negative Reactivity Feedback

To demonstrate compliance with the requirements of GDC 11, calculations are performed to determine the magnitude and nature of the reactivity feedback coefficients. These include the Doppler, moderator temperature, and void reactivity coefficients. These calculations are performed to demonstrate that the reactivity coefficients ensure inherent negative feedback. The void reactivity coefficient is the strongest contributor to the power reactivity coefficient for BWRs. The void reactivity coefficient is generally orders of magnitude stronger than either the Doppler coefficient or moderator temperature coefficient. Additionally, the void reactivity coefficient is generally insensitive to the fuel temperature because of the strong influence of the water density on the neutron spectrum. Therefore, while the calculation of the fuel temperature may impact the calculated Doppler reactivity coefficient, compliance with this GDC is generally demonstrated by calculating the magnitude of the much stronger, negative void reactivity coefficient. Therefore, the NRC staff considers the use of legacy methods to demonstrate compliance with GDC 11 to be acceptable.

7.1.2.2 Shutdown Margin

GDC 25 requires that the reactivity control system be capable of maintaining the reactor in a subcritical condition, assuming a single malfunction of the system. GDC 26 requires a secondary control system capable of shutting down the reactor. GDC 27 requires that these combined systems have sufficient capability to shut down the core under accident conditions. Several PANAC11 eigenvalue calculations demonstrate compliance with GDC 25, 26, and 27.

The purpose of these calculations is to demonstrate sufficient capability of the reactivity control systems to ensure shutdown.

The calculations performed to demonstrate compliance with these GDC are cold shutdown margin calculations. TGBLA06/PANAC11 methods are used to determine the core eigenvalue under various control states (e.g., all rods in, all rods except the strongest rod in, or borated conditions). As the core reactivity increases with decreasing temperature and increased water density, these calculations are performed at limiting cold conditions. Therefore, the model used to determine the fuel temperature at power conditions is not necessary to assess the cold shutdown margin. Thus, the NRC staff finds that the fuel thermal conductivity model does not affect the calculations used to demonstrate compliance with these GDC. Consequently, the NRC staff finds the use of legacy methods to perform these calculations to be acceptable.

7.1.3 Conclusions Regarding Steady-State Calculations

The NRC staff considered those calculations that are performed in the steady-state mode to evaluate reactor performance relative to the relevant NRC acceptance criteria. As a result of its survey, the NRC staff identified a concern regarding the fuel rod internal pressure analysis. The NRC staff determined that the use of the historical GSTRM code to evaluate fuel rod performance against the NCLO acceptance criterion may be nonconservative. The NRC staff has previously imposed a penalty to GSTRM for use in the EPU or MELLLA+ licensing applications to conservatively reduce the critical pressure to ensure adequate margin.

As for the nuclear design calculations, the NRC staff determined that the fuel rod thermal conductivity model does not affect those steady-state calculations that are performed as part of the safety analysis to demonstrate compliance with the applicable nuclear design criteria. Therefore, the NRC staff finds that the continued use of TGBLA06 and PANAC11 to perform these safety analyses is acceptable.

7.2 Transients

Transients refer to those analyses performed to assess the impact of AOOs, as well as analyses performed to demonstrate compliance with overpressure criteria, particularly the ASME overpressure and ATWS overpressure criteria. ATWS overpressure refers to a specific transient analysis in which scram is not modeled; however, the transient analysis is performed for the period of time before boration.

These calculations are performed to determine the impact of transient conditions on thermal margin and are used to establish operating limits such that anticipated operational occurrences do not challenge fuel rod integrity. The analysis codes used to do these calculations include ODYN, TASC, ISCOR, and TRACG.

7.2.1 Reload Safety Analysis Insights

The NRC staff considered the relevancy of the sensitivity studies to the broad range of AOOs that may occur for operating BWR plants. Licensees analyze a host of transients each operating cycle to determine thermal operating limits. The potentially limiting AOO events are determined and analyzed. The potentially limiting transient events analyzed on a cycle-specific basis include generator load rejection or turbine trip without bypass, loss of feedwater heating or inadvertent HPCI, control rod withdrawal error, feedwater controller failure to maximum demand, and pressure regulator failure (for BWR/6 plants) (Reference 1).

For the operating fleet of BWR plants, these events are generally the limiting events. Of these, the generator load rejection without bypass, turbine trip without bypass, feedwater controller failure, and pressure regulator failure events are pressurization transients. The sensitivity studies provided in the PRIME RAI 39 response provide details of the sensitivity of the transient response to pressurization transients (Reference 58).

7.2.2 Critical Power Criterion

GDC 10 requires that SAFDLs not be exceeded during any condition of normal operation, including the effects of AOOs. To demonstrate compliance with GDC 10, critical power ratio safety and operating limits are established to preclude fuel cladding failure as a result of boiling transition.

Transient calculations are performed in the safety analysis to demonstrate margin to boiling transition. For these calculations the figure of merit is the relative change in critical power ratio (DCPR/ICPR). The direct comparison of the BWR/4 turbine trip without bypass (TTNB) AOO indicates that the DCPR/ICPRs predicted using GSTRM and PRIME models are essentially identical. The GSTRM result is slightly higher (conservative) relative to the PRIME result. This trend is consistent with the NRC staff's expectation based on its review of the Migration LTR (Reference 33).

Therefore, the NRC staff finds that the use of the GSTRM models in the legacy methods will not adversely affect licensing calculations to demonstrate margin to boiling transition.

7.2.3 Thermal-Mechanical Criteria

GDC 10 requires that SAFDLs not be exceeded during any condition of normal operation. To demonstrate compliance with GDC 10, fuel rod T-M design limits are established to ensure fuel rod integrity in its core lifetime along the licensed power-to-flow domain during normal steady-state operation and in the event of an AOO. The fuel T-M design criteria requires, in part, the following:

(1) Loss of fuel rod mechanical integrity will not occur due to fuel melting.

    The fuel rod is evaluated to ensure that fuel melting will not occur during normal operation and corewide AOOs. For every fuel product line, the thermal overpower (TOP) limit is established to preclude fuel centerline melting. The acceptable TOP limit during a transient event is established by assuming continuous operation at the applicable exposure-dependent LHGR envelope followed by instantaneous overpower at selected exposure points in the envelope. Meeting the TOP limit ensures that the incipient fuel centerline melt criterion is not exceeded. The TOP limit is determined for both the urania and gadolinia bearing pellet rod nodes.

(2) Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction.

The fuel rod is evaluated to ensure that the calculated cladding circumferential plastic strain due to pellet-cladding mechanical interaction does not exceed 1 percent during normal operation and during AOOs. For every fuel product line, the mechanical overpower (MOP) limit is established to preclude a 1-percent diametric strain during AOOs. The acceptable MOP limit during a transient event is established by assuming continuous operation at the applicable exposure-dependent LHGR envelope for each fuel design followed by instantaneous overpower at selected exposure points in the envelope. Meeting the MOP limit ensures that the plastic strain criterion is not exceeded. The MOP limit is determined for both the urania and urania/gadolinia bearing pellet rod nodes.

7.2.3.1 Thermal Overpower

The PRIME RAI 39 response indicates that the use of GSTRM models in the legacy codes may result in the under-prediction of the fuel centerline temperature. Generally, during reload licensing analyses, the fuel centerline temperature criterion is met by demonstrating margin to the TOP limit that is generically established for the fuel product. In cases in which the screening criterion is not met, more detailed calculations are performed using approved transient methods.

Generally speaking, for slow transients the PANACEA code is used to determine the change in LHGR as a result of the transient. For fast transients the ODYN code is used. The transient change in LHGR is combined with steady-state GSTRM calculations or transient conduction heat transfer calculations to determine the fuel centerline temperature.

On the basis of the PRIME RAI 39 response, the NRC staff cannot conclude with reasonable assurance that the detailed analyses using legacy methods are conservative.

When using the generic TOP limits, the figure of merit from the transient calculation is the transient change in LHGR predicted by the systems analysis code. This code may be either ODYN or TRACG. In its review of the TRACG04 methodology for transients (References 25 and 33), the NRC staff found that the use of GSTRM thermal conductivity is conservative for this purpose. The transient LHGR will be overpredicted because the higher GSTRM thermal conductivity will reduce the fuel thermal time constant and result in higher calculated transient cladding heat flux. Additionally, the GSTRM model will result in conservative Doppler worth calculations.

These trends are independent of the analytic code; therefore, the same arguments are applicable to ODYN and TRACG02. On this basis, the NRC staff finds that the use of generic PRIME TOP limits is acceptable when the transient LHGR is calculated using the legacy methods during the interim process.

To address its technical concern regarding the potential nonconservatism in the transient TOP analysis, the NRC staff imposed the following conditions on PRIME to address the interim process:

(1) Legacy fuel TOP limits for fuel products currently used in operating plants shall be confirmed to be conservative using the PRIME methodology or revised to be consistent with the PRIME results.

(2) Detailed cycle-specific calculations (if they are required) must be performed using transient fuel performance models that are fully consistent with the approved PRIME models.

In terms of meeting the condition for detailed cycle-specific calculations, the NRC staff understands that several approaches may be employed that are acceptable. For example, TRACG04 may be used as it is an approved transient analysis code that includes the PRIME thermal conductivity model and may accept gas gap conductance input from PRIME.

7.2.3.2 Mechanical Overpower

Transient calculations are performed to demonstrate margin to the 1-percent cladding plastic strain limit, thus ensuring MOP margin. TRACG04 does not directly output the plastic strain. [

]. The TRACG04 results using GSTRM and PRIME models are essentially identical. Again, the NRC staff notes that the use of the GSTRM models is slightly conservative relative to the use of the PRIME models.

Generally, compliance with the 1-percent plastic strain criterion is demonstrated by performing transient calculations and demonstrating margin to the generic MOP limit for a specific fuel design. The [

]. Therefore, the NRC staff finds that GSTRM MOP limits generated for legacy fuel products are conservative.

When the generic MOP limit is not met on a cycle-specific basis, detailed transient analyses are performed. When TRACG04 is used, [

].

On the basis of this insensitivity, the NRC staff finds that the legacy methods may be used during the interim process. For the legacy methods, the NRC staff finds that no specific thermal margin enhancement is required to address their use in demonstrating compliance with the 1-percent plastic strain criterion, if detailed cycle-specific analyses are required.

7.2.4 Overpressure Criteria

GDC 14, "Reactor Coolant Pressure Boundary," requires that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to ensure an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. To demonstrate compliance with GDC 14, transient calculations are performed to ensure that the vessel meets ASME pressure limits.

These analyses include ATWS and ASME overpressure analyses. The calculations are very similar to pressurization transient analyses. Therefore, the NRC staff considered the predicted pressurization for the BWR/4 TTNB AOO as representative of all pressurization transients (including overpressure) in terms of the pressure sensitivity to the fuel thermal conductivity models and gas gap conductance files.

In the BWR/4 TTNB AOO case, TRACG04 predictions using either model reflect essentially identical peak pressures. The NRC staff notes that the use of GSTRM appears to be slightly conservative. This is consistent with the NRC staff's expectations based on its review of the Migration LTR (References 25 and 33).

A subsequent section of this report further discusses the ESBWR ATWS event analyses. The NRC staff notes that the ESBWR ATWS event provides a comparison of the predicted peak pressures. These two predicted peak pressures are essentially identical. Therefore, when considered with the BWR/4 TTNB AOO, the NRC staff has reasonable assurance that the calculated peak pressure for transients and ATWS events are insensitive to the fuel thermal modeling.

Therefore, the NRC staff finds that the use of the GSTRM models in the legacy methods will not adversely affect licensing calculations to demonstrate overpressure margin.

7.2.5 Conclusions Regarding Transients

The NRC staff reviewed the sensitivity studies provided in response to PRIME RAI 39. The NRC staff found that the use of either modern or legacy fuel thermal performance models had an insignificant effect on the analysis of critical power or MOP transients. However, the NRC staff found that the predicted fuel centerline temperatures were under-predicted using the legacy fuel thermal performance models. Therefore, when evaluating the transient margin to fuel melting, the NRC staff finds that the use of legacy codes may be nonconservative.

7.3 Stability

GDC 12 requires that the reactor design ensures that power oscillations that may result in the fuel exceeding SAFDLs are either not possible or can be readily detected and suppressed. GDC 10 requires that the fuel does not exceed the SAFDLs. Demonstration of compliance with these GDC may require various analyses.

7.3.1 Decay Ratio Criterion

To demonstrate that power oscillations are not possible, calculations are performed to determine the reactor channel, core, and possibly the regional mode decay ratio. These calculations may be performed with TRACG04, in the case of the ESBWR, or with ODYSY in the case of the channel and core decay ratios for the operating fleet.

7.3.2 DIVOM Development

For cases in which the power oscillations are suppressed, analyses must be done to establish appropriate setpoints that ensure that these oscillations do not result in the fuel exceeding SAFDLs. In this case, detailed transient calculations are performed to assess the change in thermal margin with the oscillation magnitude. The NRC staff has approved TRACG02 for this purpose. However, the NRC staff understands that TRACG04 has been used for this application on a plant-specific basis under the provisions of 10 CFR 50.59, "Changes, Tests and Experiments," and has been demonstrated to be conservative or essentially the same as the TRACG02 methodology. GEH has submitted a supplement to LTR NEDO-32465-A to the NRC requesting review and approval of the use of TRACG04 for this application.

7.3.3 Anticipated Transient without Scram with Instability (MELLLA+)

MELLLA+ applications require analysis of ATWS instability on a plant-specific basis in accordance with the NRC staff's SE limitations and conditions for the approved MELLLA+ LTR. However, GEH/GNF has not submitted a methodology for review to perform these calculations. To date, generic TRACG calculations have been referenced to justify the effectiveness of the current EPGs to mitigate the consequences of thermal-hydraulic instabilities during ATWS events. The NRC staff has found that plants seeking to expand their allowable operating domain to MELLLA+ conditions must provide further justification. GEH has committed to submit a methodology for this purpose. The NRC staff intends to consider the adequacy of the modeling of the fuel pellet thermal performance during its review of the ATWS analysis methodology.

7.3.4 Conclusions Regarding Stability

The PRIME RAI 39 response contains comparisons of the corewide growth rate for a BWR/4 to the regional mode decay ratio for the ESBWR. The results predicted using the PRIME model result in lower decay ratio/growth rates. The results are intuitive because the decreased fuel thermal conductivity model in PRIME directly results in an increase in the fuel thermal time constant. Increasing the time constant is a stabilizing phenomenon which results in decreased coupling between the fluid and neutronic core response. Therefore, the results are expected.

The results confirm that the use of GSTRM models in the legacy stability codes will predict enhanced coupling relative to the PRIME models. Therefore, licensing stability calculations performed using the legacy codes will be conservative. The NRC staff finds the use of legacy methods for stability calculations to be acceptable. Similarly, the NRC staff finds the generic TRACG ATWS instability calculations to remain acceptable for reference.

For the special case of ATWS instability analyses for MELLLA+ plants, the NRC staff intends to review the ATWS methodology that GEH committed to submit in Reference 51. The NRC staff's review of this updated ATWS methodology will consider the adequacy of the fuel thermal performance models.

7.4 Design-Basis Accidents

The NRC staff's survey of GEH/GNF methods considered two DBAs. The first is an ECCS LOCA and the second is an RIA. To analyze postulated LOCA events, GEH/GNF uses the SAFER-GESTR methodology and its supporting codes. For RIAs, the standard licensing approach is to perform a plant-specific bounding analysis and to demonstrate that the results are bounding on a plant-specific basis. Therefore, the NRC staff survey considered the nature of the methodology used to perform the bounding analysis in the plant-specific final safety analysis report.

7.4.1 Loss-of-Coolant-Accident

7.4.1.1 Regulatory Criteria

As specified in 10 CFR 50.46, the following acceptance criteria apply to ECCS-LOCA evaluations: (1) the PCT remains below 2200 degrees Fahrenheit, (2) the maximum oxide thickness does not exceed 17 percent of the cladding thickness anywhere in the core, (3) the total hydrogen formed does not exceed 1 percent of the hypothetical amount if the entire cladding inventory (excluding plena) were reacted, (4) the core retains a coolable geometry, and (5) long-term cooling is maintained. For the operating reactor fleet, GESTR/SAFER analyses are performed to calculate the PCT, oxide thickness, and core volume oxidized.

7.4.1.2 Large-Break Loss-of-Coolant-Accident

7.4.1.2.1 <u>BWR/2–4 Peak Cladding Temperature</u>

GEH/GNF provided calculated PCTs for a BWR/4 and a BWR/2. The BWR/4 case indicates that the PCT predicted using the PRIME fuel thermal model results in an insignificant increase in PCT of approximately [ ]. For the BWR/2 case, the difference is even smaller (approximately [ ]). Table 1 summarizes the results.

Table 1 TRACG04 PCT Sensitivity Study Results

	GSTRM	PRIME	PRIME-GSTRM
BWR/4 PCT (°F)	[		
BWR/2 PCT (°F)			]

The NRC staff notes that the predicted initial fuel temperatures using the PRIME thermal model is significantly higher in both cases (approximately [ ]). This will affect the stored energy in the core.

For BWR/3–6 plants, the limiting DBA LOCA break occurs in the recirculation suction line. During the LOCA event, the CF stagnates after a few seconds due to uncovery of the jet pump. The stagnation results in two-phase separation and the formation of a liquid level in the core. The core is subject to transition boiling during the flow stagnation period. The occurrence of early transition boiling results in an early or first peak in the PCT. The level in the core is restored once depressurization occurs when the break uncovers. The level then drops as a result of mass loss through the break and the core will uncover. The second peak in PCT occurs when the core uncovers.

The relative magnitude of the first peak PCT is a stronger function of the stored energy, as this dictates when the fuel enters boiling transition during the flow stagnation. The effectiveness of the depressurization to remove the energy makes the second peak PCT a much stronger function of the relative capability of the ECCS to inject coolant and the degree of decay heat being generated within the core.

The analytical results provided in the sensitivity studies appear to confirm that the sensitivity of the second peak PCT to the difference in the stored energy is insignificant. That is, in both the BWR/2 and BWR/4 cases, the PCT differences were less than the significance threshold of 50 degrees Fahrenheit specified in 10 CFR 50.46.

The NRC staff compared the TRACG04 stored energy sensitivity to the GESTR/SAFER sensitivity reported in Table A.8 of NEDE-23785P-A (Reference 2). The NRC staff approximated the change in the stored energy based on the average fuel temperature, as represented in Equation 1. The NRC staff then normalized the PCT sensitivity to the magnitude of the stored energy perturbation. The specific heat is taken as the default urania model in Equation C.1-5 of the TRACG Model Description LTR (Reference 59).

$$\delta PCT_N \approx \frac{\Delta PCT_N}{1.645\sigma}\left(\frac{\int_0^{T_{PRIME}} C_p\,dT}{\int_0^{T_{GSTRM}} C_p\,dT}-1\right) \qquad (1)$$

Where δPCT is the PCT sensitivity to the initial average temperature difference,
ΔPCT is the PCT sensitivity to the initial stored energy [ ],
N denotes either first or second peak,
σ is the GESTR stored energy uncertainty [ ],
C_p is the specific heat,
T is the initial average fuel temperature,
GSTRM denotes calculated according to the GSTRM models, and
PRIME denotes calculated according to the PRIME models

The NRC staff's approximated second peak PCT sensitivity was calculated to be [ ] for a BWR/4. This is consistent with the PCT difference predicted by TRACG04 [ ]. Therefore, the NRC staff has confidence that the TRACG04 sensitivity studies are consistent with expected trends in the GESTR/SAFER methodology.

7.4.1.2.2 BWR/5–6 Peak Cladding Temperature

For BWR/5–6 plants it is not a foregone conclusion that the limiting PCT occurs for the second peak. These plants and some BWR/4 plants include low pressure injection into the core bypass that results in a more rapid delivery of coolant to the core relative to BWR/3–4 plant designs, in which the low pressure coolant injection is into the lower plenum.

Using Equation 1, the NRC staff estimated the impact of the difference in stored energy on the first peak PCT. The NRC staff calculation indicated a potential nonconservatism on the order of [ ], which is greater than the significance threshold according to 10 CFR 50.46. Therefore, the NRC staff could not conclude that the use of the GSTRM model in the legacy LOCA methods would not result in an insignificant difference in the first peak PCT. Therefore, the NRC staff could not reach a conclusion regarding the applicability of the interim process to BWR/5–6 plants.

To address this concern, the NRC staff requested additional information regarding the first peak PCT sensitivity to the differences in stored energy in PRIME RAI 39, Supplement 3, Part B (PRIME RAI 39S3-B).

The response to PRIME RAI 39S3-B provides the results of SAFER/GESTR calculations for two representative BWR plant configurations that are limited by the first peak PCT (Reference 52). The results indicate sensitivity in the first peak PCT of approximately [ ], which indicates consistency across the various BWR plant designs. The more detailed SAFER/GESTR calculations are (1) consistent with the analysis method outlined in Appendix K to 10 CFR Part 50 and (2) are representative of the detailed plant response sensitivity to differences in stored energy. Therefore, the NRC staff finds that these results provide a more robust and reasonable basis to determine the PCT impact of the PRIME thermal model relative to the NRC staff's simplistic approach described in Equation 1.

7.4.1.2.3 BWR/2 Oxidation

Cladding oxidation calculations were performed for a BWR/2. The BWR/4 PCT results indicate that the degree of cladding oxide formation would be insignificant based on the low temperatures. BWR/2 plants tend to have more limiting core oxidation during LOCA DBAs based on the nature of the recirculation piping. Therefore, the NRC staff finds it acceptable to compare the oxidation results for the BWR/2 plant without consideration of the BWR/4 plant.

The calculations compare the maximum local oxide layer thickness as well as the fraction of cladding oxidized. The fraction of cladding oxidized is a surrogate metric to ensure that the maximum hydrogen generation criterion of 10 CFR 50.46 is met. The results indicate close agreement between the TRACG04 calculations using both fuel thermal models. The NRC staff agrees that the oxidation results are essentially identical.

It is well understood that BWR/2 plant designs are most limiting in terms of the oxidation criteria because of the more aggressive rate and duration of core uncovery during design-basis LOCA events. The primary reason is that the recirculation system is designed with large lower vessel penetrations. Therefore, the primary phenomenon driving cladding oxidation for the BWR/2 design is the period of core uncovery, which is not very sensitive to the initial fuel temperature or stored energy. Since the BWR/3–6 plant designs incorporate jet pumps, the level drop during a DBA LOCA is not as severe, leading to significant margin to the cladding oxidation limits in 10 CFR 50.46.

Therefore, the NRC staff finds that it is acceptable to utilize the legacy methods to demonstrate compliance with the oxidation acceptance criteria specified in 10 CFR 50.46.

7.4.1.3 Small-Break Loss-of-Coolant Accident

The response to PRIME RAI 39 did not address the sensitivity of a small-break LOCA to the fuel thermal model. Significant changes in plant operations and other modifications have challenged the conclusions of the original GESTR/SAFER model qualification and application statement regarding the limiting nature of large-break LOCA events for the BWR/3–6 operating fleet. Therefore, the NRC staff requires that the impact of the sensitivity be quantified for small-break LOCAs. The NRC staff requested additional information regarding the sensitivity of small-break LOCA analyses in PRIME RAI 39, Supplement 3, Part A (PRIME RAI 39S3-A).

The response to PRIME RAI 39S3-A provides the results of calculations performed using the SAFER/GESTR analysis methodology for varying initial stored energies (Reference 52). The previous analyses using TRACG indicate approximately [ ]. The small-break LOCA Appendix K calculations indicate a negligible difference in the PCT, oxidation, and metal-water reaction results. The response states that, since core uncovery does not occur during the early stage of the small-break LOCA, the nucleate boiling occurring in the core during the event is sufficient to remove the initial stored energy. Consequently, the sensitivity is expected to be small once the transient evaluation period reaches the longer durations when PCT occurs for small-break LOCA events. The NRC staff has reviewed these calculations and the interpretation and agrees with the engineering judgment of GEH/GNF that small-break LOCA calculation results are expected to be negligibly impacted.

7.4.1.4 Conclusions Regarding the Design-Basis Loss-of-Coolant Accident

The NRC staff found that the continued use of the legacy fuel T-M models in the LOCA analyses may result in the under-prediction of the PCT. This was found in sensitivity analyses performed by GEH/GNF using both SAFER/GESTR and TRACG. However, the magnitude of the difference in PCT for a wide range of plants was shown to generally be less than 50 degrees Fahrenheit. Therefore, the NRC staff finds that the thermal conductivity impact on PCT for plant-specific analyses is not expected to meet the significance threshold of 50 degrees Fahrenheit found in 10 CFR 50.46.

7.4.2 Reactivity Initiated Accident

The design-basis RIA for a BWR is the CRDA, in which it is postulated that, during any point in the operation of the reactor, a control blade becomes stuck in the fully inserted position and becomes decoupled from the associated drive mechanism. At a later point in time, the drive is withdrawn leaving the control blade in the fully inserted position. An RIA occurs when the control blade is postulated to become free and drop to the position of the decoupled drive. Analysis of the CRDA must consider all possible control rod configurations and operating conditions to determine the consequences from a limiting control blade drop.

Typically, the consequences of a CRDA are greatest under cold zero-power conditions. Under these conditions, the control blade incremental worth is highest, the core is loosely coupled, and the RPV inventory is predominantly liquid water, which is potentially subcooled such that moderator voiding does not contribute negative reactivity feedback.

The control rod drop occurs when the dropped rod falls to the last position of the drive mechanism at a rate determined by the design of the velocity limiter at the bottom of the blade. When a control rod drops under cold conditions, the local power around the control rod increases rapidly and dramatically—typically on the order of a decade every 25 milliseconds. The rapid power increase results in an increase in the fuel temperature, which results in a negative reactivity addition due to the Doppler effect. The Doppler reactivity limits the peak transient power, and the event is terminated by a 120-percent average power range monitor scram.

During the CRDA, a potential exists for the local power to increase substantially and result in the formation of voids around the fuel pins in nucleation locations. The formation of these voids, while generally not credited in CRDA analyses, provides additional negative reactivity feedback to help limit the peak and integrated local power before the scram.

The power increase from the reactivity addition is terminated by prompt negative feedback from the Doppler effect and the heatup of the fuel surrounding the dropped blade. The nuclear dynamic response and the thermal-hydraulic models are used to determine the energy deposition in the fuel during the power increase in the early phase of the transient and through the termination after scram. These values are then compared to the fuel enthalpy limits provided in SRP Section 4.2 “Fuel System Design.” (Reference 48).

The limiting CRDA is determined on a plant-specific basis considering the particular plant hardware and technical specifications. For banked position withdrawal sequence (BPWS) plants, the rod worth minimizer (RWM) issues rod blocks to limit the incremental reactivity worth of any potential dropped rod. Additionally, the analysis must account for the minimum scram times based on the allowable limits provided in the plant technical specifications.

In determining the limiting control rod, consideration is given to the maximum rod worth based on achievable rod motion deviations from the BPWS allowed by the technical specifications, plant hardware (including the ability to bypass rod blocks issued by the RWM), and the worst single failure or operator error.

The NRC staff identified the CRDA as being potentially sensitive to the error in the fuel thermal conductivity because of the sensitivity of the power increase to the magnitude of the Doppler reactivity coefficient and, possibly, the sensitivity of the analysis to the heat conduction from the fuel pellet to the surrounding coolant.

7.4.2.1 Regulatory Criteria

GDC 28 is the applicable regulatory criterion for CRDA. GDC 28 requires, in part, that the reactivity control systems be designed to limit reactivity accidents so that the reactor coolant system boundary is not damaged beyond limited local yielding.

Compliance with GDC 28 is demonstrated by performing CRDA analyses for limiting control blades to ensure that the fuel integrity is not challenged by the worst possible CRDA. Specifically, compliance is demonstrated by analyzing the consequences of the CRDA and comparing the fuel enthalpy rise against the criteria for the cladding failure, specific energy design, and prompt fuel dispersal enthalpy limits.

7.4.2.2 Analysis Methodology

LTR NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," issued March 1972, and its supplements (References 61, 62, and 63) describe GEH/GNF's NRC-approved CRDA analysis methodology. The analysis approach is based on an adiabatic approximation. The primary design method for analysis of super prompt critical large-core nuclear excursions uses the adiabatic approximation with a two-dimensional, multigroup flux calculation.

In this method, the fuel enthalpy is calculated by integrating the solution of the reactor kinetics equations and ignoring the thermal-hydraulic effects. The integrated heat deposition to the fuel is used to determine the enthalpy, as well as the fuel temperature. The fuel temperature calculation is used to determine the reactivity feedback resulting from the Doppler effect.

The Doppler coefficient is determined from branch calculations using the lattice physics code. In the simulation of the CRDA, the analysis method uses the change in the square root of the temperature to calculate the core reactivity effect of the temperature rise during the power excursion.

Therefore, the heat deposition in the fuel is sensitive to the formulation of the heat capacity model and the magnitude of the Doppler coefficient. The fuel thermal conductivity does not enter into the analysis because heat transfer through the fuel to the coolant is conservatively ignored.

Thermal conductivity degradation with fuel exposure has the potential to affect two aspects of the CRDA analysis methodology: (1) the initial fuel temperature and (2) the magnitude of the Doppler coefficient. As to the first point, the CRDA events are most limiting at cold or low-power conditions. The analysis conservatively considers a power of 10^{-8} to 10^{-6} times the rated thermal power and, therefore, fixed fuel and moderator temperatures of 20 degrees Celsius. As to the second point, the initial temperatures are fixed by the limiting conditions of the reactor; therefore, the Doppler coefficient is determined from the lattice physics calculations assuming a fixed initial temperature.

The NRC staff finds that the treatment of the fuel thermal conductivity is not required to execute the accepted CRDA analysis methodology. Furthermore, the conservatism of the adiabatic approach to the simulation has been observed when simulations are compared to test data collected during the special power excursion reactor test (SPERT) 3.2-millisecond period experiment.

7.4.2.3 Conclusions Regarding Reactivity Initiated Accidents

The NRC staff has found that the treatment of the effect of thermal conduction within the fuel is not required to analyze hypothetical CRDA events for BWRs. Additionally, the adiabatic approach to these licensing evaluations is conservative. On these bases, the NRC staff finds the continued application of this methodology acceptable.

7.5 Beyond Design-Basis Accidents

Beyond design-basis accidents refer to the ATWS and SBO events. Section 6.8 discusses the methods employed by GEH/GNF to analyze these events. The NRC staff reviewed the sensitivity of these analyses to the models dictating fuel thermal behavior during accidents and considered the risk significance of ATWS and SBO events.

7.5.1 Risk Insights

In its survey of analysis methods employed for beyond design-basis accidents, the NRC staff considered any risk insights gathered from probabilistic risk assessments (PRAs) that have been performed for specific plants. The NRC staff referenced the BWR/4 and BWR/6 PRAs that were done for Peach Bottom 2 and Grand Gulf 1 as part of NUREG/CR-4550 (References 64 and 65, respectively). The results of the specific PRAs are provided below. On the basis of the PRAs, the NRC staff has determined that the ATWS and SBO events are large contributors to the overall plant risk and that this conclusion is likely applicable across the fleet of BWR plants.

7.5.1.1 BWR/4

One of the major purposes of the Peach Bottom analysis discussed in Volume 4 of NUREG/CR-4550 was to provide an updated perspective on the agency's understanding of the risks from the plant relative to the results of previous analyses. It has been determined that changes to the plant design and its procedures, the evolution of PRA methodology, and an increasing understanding of severe accidents have all impacted the perspectives on the dominant risks for Peach Bottom.

This study concluded that SBO accidents and ATWS scenarios are the dominant contributors to core damage at Peach Bottom. The possibility of successful containment venting and successful core cooling after containment failure have considerably reduced the significance of the loss of long-term heat removal accidents originally found to be important in the reactor safety study (WASH-1400, Reference 66).

Giving credit for more injection systems, using realistic system success criteria, and implementing plant modifications have also collectively reduced the importance of loss-of-injection-type sequences. Given the considerable redundancy and diversity of coolant injection and heat removal features at Peach Bottom, it is not surprising that common features of the plant tend to drive the mean core damage frequency.

Common features include common-cause failures of equipment, failure of common support systems, loss of ac power and emergency service water, and human error. In light of this conclusion, it must also be recognized that the calculated core damage frequency in the NUREG/CR-4550 study is subject to the nontrivial uncertainties associated with the common-cause and human error analyses.

The above insights can be considered applicable to other BWRs of similar design to the extent that the redundancy arguments are true for other plants of interest.

7.5.1.2 BWR/6

One of the major purposes of the Grand Gulf analysis was to provide an updated perspective on our understanding of the risks from the plant relative to the results of the reactor safety study methodology applications program (RSSMAP) analysis. Changes to the plant design and its procedures, the evolution of PRA methodology, and an increasing understanding of severe accidents have all impacted the perspectives on the dominant risks for Grand Gulf.

This study concludes that SBO accidents are the dominant contributors to core damage at Grand Gulf. Realistically allowing for successful core cooling after containment failure has considerably reduced the significance of the loss of long-term heat removal accidents, which the RSSMAP study originally found to be important. Giving credit for more injection systems, using realistic system success criteria, and implementing plant modifications have also collectively reduced the importance of loss-of-injection-type sequences.

Given the considerable redundancy and diversity of coolant injection and heat removal features at Grand Gulf, it is not surprising that common features of the plant tend to drive the mean core damage frequency. These include common-cause failures of equipment, failure of common support systems (ac power and standby service water), and human error. In light of this conclusion, it must also be recognized that the calculated core damage frequency in the NUREG/CR-4550 study is subject to the nontrivial uncertainties associated with the common cause and human error analyses.

The above insights can be considered applicable to other BWRs of similar design to the extent that the redundancy arguments are true for other plants of interest.

7.5.2 Anticipated Transients without Scram

SRP Section 15.8, “Anticipated Transients Without Scram,” specifies the ATWS acceptance criteria, which are based on meeting the relevant requirements of the following regulations:

(1) 10 CFR 50.62, as it relates to the acceptable reduction of risk from ATWS events by (a) inclusion of prescribed design features and (b) demonstration of their adequacy;

(2) 10 CFR 50.46, as it relates to maximum allowable PCTs, maximum cladding oxidation, and coolable geometry;

(3) GDC 12, “Suppression of Reactor Power Oscillations,” as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed;

(4) GDC 14, “Reactor Coolant Pressure Boundary,” as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary;

(5) GDC 16, “Containment Design,” as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents;

(6) GDC 35, “Emergency Core Cooling,” as it relates to ensuring that fuel and clad damage, should it occur, not interfere with continued effective core cooling, and that clad metal-water reaction must be limited to negligible amounts;

(7) GDC 38, “Containment Heat Removal,” as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment; and

(8) GDC 50, “Containment Design Basis,” as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

Insofar as analytical codes are used to demonstrate compliance with the regulatory criteria, calculations are performed for the limiting ATWS events to (1) determine the vessel pressurization to demonstrate compliance with GDC 14, (2) determine the suppression pool temperature to demonstrate compliance with GDC 16, 38, and 50, (3) determine the PCT and maximum oxidation to demonstrate compliance with the 10 CFR 50.46 criteria, and (4) determine whether the core remains in a coolable geometry.

The PRIME RAI 39 response provides sensitivity studies for the ESBWR MSIV closure ATWS event. The parameters compared are the maximum neutron flux, the vessel pressure, and the suppression pool bulk temperature. The response states that consideration of the criteria of 10 CFR 50.46 for ATWS is substantially similar to consideration of those sensitivities reported for the ECCS LOCA calculations. The NRC staff agrees with this assessment and finds that, when considered with the ECCS LOCA comparisons, the response adequately addresses the relevant safety figures for ATWS simulations.

Generally, an ATWS event may be described as having three distinct phases. In the first phase, the vessel is pressurized by an initiating event, in this particular case an MSIV closure. During this first phase, the reactor power and neutron flux will pulse as the initial void collapse introduces reactivity and a combination of negative void and Doppler worth terminate the power increase. In the second phase, the reactor power stabilizes at a critical configuration that is governed by the core flow rate (natural circulation conditions). In this second phase, the core attains an adjoint-weighted average void fraction that is similar to the initial condition. The reactor power remains relatively steady during this phase, but will change with any variation in the reactor vessel level, and steam is relieved to the suppression pool. In the third phase, boron injection shuts down the reactor and brings the core to a subcritical state.

7.5.2.1 Power Pulse

The phenomena that dictate the reactor behavior during the first phase include the reactivity feedback from void and Doppler and the intensity of the pressure wave impinging on the core.

Compared to transients, ATWS events tend to demonstrate a greater sensitivity to the Doppler coefficient as there is more significant fuel heatup because the event is not terminated with a scram. The PRIME RAI 39 response states that the Doppler feedback is stronger for higher initial temperatures. The NRC staff does not agree with the response in its assessment of the Doppler feedback. The NRC staff conducted a detailed review of the Doppler feedback trends associated with temperature during its review of the Migration LTR. During this review, the NRC staff found that TRACG04 (as well as legacy codes) will incorporate nodal temperature reactivity feedback response surfaces that are generated at the PANAC11-predicted initial fuel temperature. Therefore, the Doppler coefficient itself is not treated as being sensitive to the initial temperature. The NRC staff noted in its previous review of TRACG04 that as the temperature increases the magnitude of the Doppler coefficient tends to decrease.

The PRIME RAI 39 response may refer to a trend whereby increased initial temperature results in a greater temperature increase during the transient evaluation. This may be a result of increased heat holdup due to a smaller thermal conductivity of the pellet with increasing temperature. However, the dynamics of the power increase are a strong function of the core hydraulics, the void reactivity, and the fuel thermal time constant. Therefore, the NRC staff cannot conclude categorically that higher fuel temperatures result in increased Doppler feedback. In fact, the NRC staff believes that the opposite trend is expected, and the GEH analytical methods simply do not capture it.

The NRC staff expects that the peak neutron flux would be sensitive to the fuel modeling parameters. The NRC staff also expects that the calculated power pulse will be impacted by a combination of the void reactivity and Doppler feedback. These two reactivity effects will likely have a competing effect when the fuel thermal modeling is perturbed between the GSTRM and PRIME models. That is, the void formation that occurs after the pressurization is enhanced when the fuel thermal resistance is lower, thus contributing to a lower flux peak. However, when the fuel thermal resistance is low, the fuel temperature increase is dampened by effective heat transfer and the Doppler effect is lessened. Regardless of the relative magnitude of these two separate effects, the comparison provided in the PRIME RAI 39 response demonstrates that the peak flux predicted by either model is essentially identical.

The NRC staff considered the impact of a potential Doppler coefficient bias that is consistent with the predicted difference in average fuel temperature, assuming that the Doppler coefficient scales as the square root of the temperature. Using the values provided in the PRIME RAI 39 response, the NRC staff estimated that the temperature difference would indicate a bias in the Doppler coefficient on the order of [

] provided in Reference 31. Figure 8-11 of Reference 31 provides the peak pressure sensitivity to a Doppler coefficient variation of [ ]. The results indicate that the potential sensitivity to the Doppler coefficient bias introduced by the error in the GSTRM temperature prediction is on the order of 2 to 3psi. The NRC staff finds that this potential bias is negligible.

While there may be competing effects, the NRC staff finds that, during the initial power pulse, the kinetics solutions remain, overall, insensitive to the fuel thermal models. This is further evidenced by the high degree of agreement between the peak pressures calculated using either method. These, also, are essentially the same. Therefore, in terms of demonstrating compliance with GDC 14, the NRC staff finds use of the legacy methods to be acceptable.

7.5.2.2 Natural Circulation

Prediction of the containment performance during an ATWS event is particularly sensitive to the predicted core thermal power during the second phase of the event. The initial power pulse contributes only a small fraction of the total heat load that is deposited in the containment. In the operating fleet, during the second phase, the reactor is brought to a natural circulation condition by tripping the recirculation pumps. In the case of the ESBWR, the reactor core remains in a natural circulation condition in which EOPs dictate the evolution of the CF. In either case, during this phase of the event, the reactor power is still significant and the steam is routed to the suppression pool. Considering the relatively long duration of this phase relative to the initial power pulse, it is the most significant contributor to the containment heat load.

The PRIME RAI 39 response is correct insofar as the reactor power level is most sensitive to the core hydraulics. The power will stabilize at any given flow rate such that the adjoint-weighted void fraction is essentially the same (with some variations given the magnitude of the negative Doppler worth). Given that the void reactivity coefficient is much greater than the Doppler coefficient, the NRC staff agrees that the heat load to the suppression pool will not be sensitive to the fuel thermal modeling because this phase is dominated by void reactivity effects and only affected by the Doppler worth negligibly.

7.5.2.3 Boration

During the boration phase, the reactor power is governed primarily by the concentration of boron delivered to the active region. This is true for both operating reactors and the ESBWR. The boron worth is not sensitive to the fuel thermal modeling; therefore, use of either model (GSTRM or PRIME) is not expected to have a significant effect on this stage of the simulation. Additionally, the fraction of the total heat deposited in the suppression pool from this phase is small compared to the heat deposited from the second phase. Therefore, the NRC staff finds that close agreement between the two calculated suppression pool temperatures is expected.

7.5.2.4 Conclusions Regarding Anticipated Transients without Scram

Overall, when all phases are considered, the NRC staff finds that either fuel thermal modeling methodology generates essentially identical containment temperature response. Therefore, the NRC staff finds use of the legacy methods for ATWS containment analysis to be acceptable.

7.5.3 Station Blackout

SBO analyses are performed to demonstrate acceptable reactor system behavior during the coping period under postulated SBO conditions. The purpose of these analyses is to demonstrate that, for the specified duration, the capabilities of the plant equipment are sufficient to cool the core and maintain the containment within pressure and temperature design limits.

In terms of performing these analyses, Regulatory Guide 1.155, “Station Blackout,” issued August 1988, specifies guidance that the decay heat must be calculated assuming operation at 100-percent power for at least 100 days (Reference 57). The decay heat is then predicted using an appropriate ANS standard. Since the calculation is performed over a period of 2 to 16 hours, the specific transient evolution of the reactor power during the scram is inconsequential, and the transient evolution of the cladding heat flux may acceptably be modeled using a quasi steady-state approach. Therefore, the results of the analysis, while sensitive to the decay heat, are unlikely to be sensitive to fuel thermal models that dictate the short-term transient cladding heat flux response.

In terms of the effect on stored energy, heat removal is driven by the decay heat over the coping period. The inventory provided by the reactor core isolation cooling or HPCI balances the core steam production and subsequent relief to the suppression pool. The heat balance calculations over the coping period are not sensitive to the initial stored energy of the core.

On the basis that the SBO analyses are insensitive to fuel thermal modeling, the NRC staff finds the continued use of legacy codes to analyze SBO to be acceptable.

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Figure 1 GEH/GNF transient code system process diagram (NEDC-32084P)

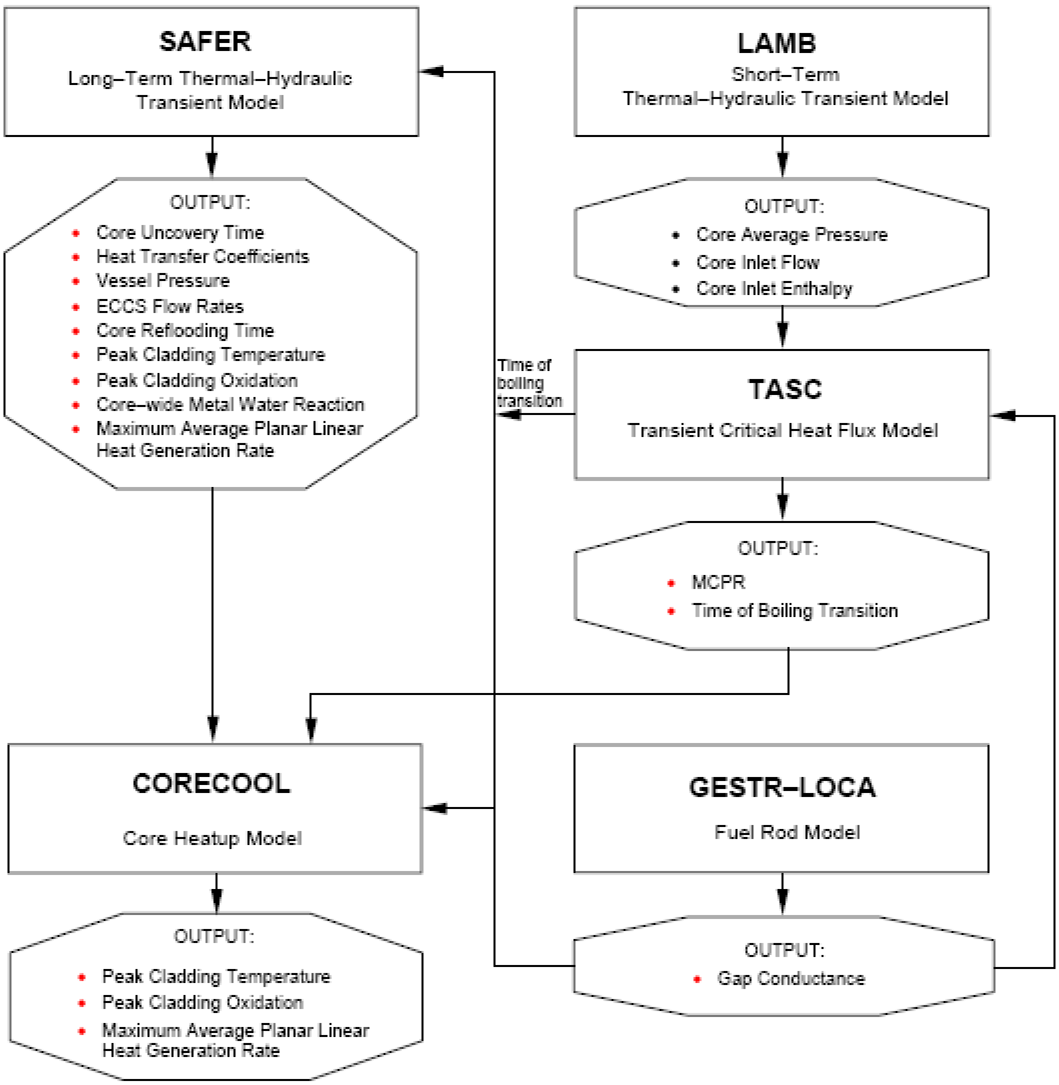

Figure 2 GEH/GNF LOCA evaluation model process diagram (NEDE-24011-P-A-16-US)