ENCLOSURE 2

M190192

Comment Summary Table and Draft SE Markup

Non-Proprietary Information

IMPORTANT NOTICE

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

Comment Summary for Draft Safety Evaluation for Licensing Topical Report NEDE-33885P, Revision 0, "GNF CRDA Application Methodology"

Note: Page numbers shown in this table reflect the page numbers in this enclosure.

Location	Comment
Page 1 / Line 13	Revise "Road" to "Rod."
Page 1 / Lines 14 through 17	Recommend revised language to clarify CRDA LTR as it pertains to prior TRACG and PANACEA methodologies.
Page 1 / Line 19	Recommend addition of the word "codes."
Page 2 / Lines 30 and 31	Recommend revised language to clarify licensee adoption and use of the new technology.
Page 5 / Line 48	Recommend revising "acceptance criteria" to "potential critical parameters."
Page 9 / Line 13	Should be GESTAR II and not GESTAR III.
Page 9 / Line 26	Recommend additional clarifying language.
Page 9 / Line 39	Recommend revision of [["]] for consistency.
Page 12 / Line 41	Recommend revision of "limit the time steps to sizes" to "allow the time steps sizes."
Page 13 / Lines 3 and 4	Recommend addition of "due to control blade movement."
Page 13 / Line 6	Recommend addition of "negative reactivity insertion."
Page 14 / Line 3	Recommend change of [[ ]]
Page 14 / Line 38	Recommend deletion of [[ ]]
Page 15 / Line 21	Recommend change of [[ ]]
Page 15 / Lines 43 through 45	Recommend revised language to better reflect the situation.
Page 22 / Line 36	Recommend addition of "of."
Page 25 / Line 14	Recommend revision of [[ ]]
Page 26 / Line 1	Recommend revision of "maximum average enthalpy" to "maximum radially averaged enthalpy."
Page 27 / Line 16	We refer to our version of TRAC as "TRACG" and not "TRAC-G."
Page 31 / Various Lines	There are three instances where a "t" was added to "weighted" under Delayed Neutron Fraction.
Page 32 / Various Lines	There are two instances where we recommend "3D" rather than "3-D" under Rod and Assembly Power Distribution.
Page 35 / Line 41	Recommend deletion of ", or less than," language.

Location	Comment
Page 38 / Line 13	Should be GESTAR II and not GESTAR III.
Page 38 / Lines 43 through 46	Most recent version of GESTAR is Revision 29.
Page 39 / Lines 47 through 49	Most recent version of NEDE-33173P-A is Revision 5.

OFFICE OF NUCLEAR REACTOR REGULATION

DRAFT SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL – AMERICAS, LLC (GNF)

LICENSING TOPICAL REPORT NEDE-33885P, Revision 0,

"GNF CRDA APPLICATION METHODOLOGY"

(EPID: L-2018-TOP-0006)

1.0 INTRODUCTION

By letter dated February 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18059A874), Global Nuclear Fuels – America, LLC (GNF-A), submitted for U.S. Nuclear Regulatory Commission (NRC) staff review a licensing topical report (LTR), NEDO-33885/NEDE-33885P, Revision 0, "GNF CRDA [Control Road Drop Accident] Application Methodology" (Ref. 1), herein described as the "CRDA LTR"). **In the CRDA LTR the previously approved TRACG and PANACEA analysis methodologies are extended** ~~The CRDA LTR is an extension of the previously approved TRACG and PANACEA analysis methodologies~~ for evaluation of the CRDA event. This safety evaluation (SE) only addresses the applicability of the CRDA LTR to the boiling water reactor (BWR) product lines and fuel types for which the TRACG and PANACEA **codes** have previously been approved (Ref. 2). In addition, the CRDA LTR includes discussion of how the update process inherent in the GESTAR-II methodology would be used to apply this methodology to potential future scenarios such as new fuel types or methodology updates.

TRACG is a thermal hydraulics analysis code package that also includes a three dimensional (3D) kinetics model for detailed calculation of neutronic feedback during transient events. PANACEA is a 3D core simulator code that primarily functions as a stand-alone steady state core simulator and depletion code. While it includes transient calculation capabilities, the heat transfer and hydraulics models are much simpler than those utilized by TRACG. TRACG is approved by the NRC for use in a broad set of BWR transient and accident scenarios, including anticipated operational occurrences (AOOs), anticipated transients without scram (ATWS), loss-of-coolant accidents (LOCAs), and potential instability events. PANACEA is primarily used for depletion and some limited applications. However, PANACEA has been accepted by the NRC for use in CRDA calculations as part of the certification of the Economic Simplified Boiling Water Reactor design (Ref. 3). A third code that is implicitly included in the overall analysis methodology is the PRIME fuel thermal mechanical performance evaluation code, which has been previously approved by the NRC. This code is not used directly in the CRDA calculations; however, it is used to derive a number of important fuel rod properties used as input by TRACG during the CRDA evaluation. In the CRDA LTR, GNF-A proposes to use PANACEA to perform [[ ]], use TRACG to perform [[ ]] to demonstrate that the acceptance criteria are met for the CRDA event.

Enclosure

2.0 BACKGROUND

The historical basis for GNF-A analysis methodologies for the CRDA event is the Banked Position Withdrawal Sequence (BPWS), as described in NEDO-21231, "Banked Position Withdrawal Sequence," January 1977. (Ref. 4). The intent of this approach is to establish a generic control rod withdrawal sequence that would ensure that control rod worths from a dropped rod would, in all cases, be sufficiently limited to meet the legacy NRC CRDA acceptance criteria (a peak enthalpy of no greater than 280 calories (cal)/gram (g), and rarely exceeding 170 cal/g for fuel cladding failure). The control rod worths are minimized through banking of control rod banks at specified positions, and generic analyses are used to demonstrate that the fuel rod enthalpies will be adequately limited by the given control rod worths.

Since NEDO-21231 (Ref. 4) was approved by the NRC, additional research in reactivity initiated accidents (RIAs) has identified that the previously mentioned legacy acceptance criteria (e.g., 280 cal/g peak enthalpy) are not adequate. In particular, two separate failure mechanisms were identified, high temperature cladding failure and pellet-clad mechanical interaction (PCMI). The former mechanism is sensitive to the differential pressure across the cladding, while the latter mechanism is sensitive to the hydrogen concentration within the cladding. This information was used to develop new interim CRDA acceptance criteria, as captured in Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," to Chapter 4.2, "Fuel System Design," of the Standard Review Plan (SRP) (Ref. 5). These criteria have been refined using more updated knowledge and published as part of a proposed draft guide, DG-1327, "USNRC Draft Regulatory Guide DG-1327, 'Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Drop Accidents'" (Ref. 6), that is expected to become a final regulatory guide superseding the current regulatory guide for RIAs.

The CRDA LTR describes a new methodology for analysis of the CRDA event, including evaluation against the more recent acceptance criteria. An approval of the CRDA LTR would allow licensees to **utilize this methodology in their licensing basis and in development of** ~~use this methodology to develop~~ their own rod withdrawal sequences that can be demonstrated to comply with the revised CRDA acceptance criteria, in lieu of the BPWS. At the time that this SE was written, DG-1327 is not expected to be finalized as a regulatory guide. However, the form of the acceptance criteria in DG-1327 is very similar to the interim acceptance criteria currently captured in Appendix B of SRP 4.2. As part of the review of the CRDA LTR, the NRC staff utilized both SRP 4.2 Appendix B and DG-1327, to the extent possible.

The NRC has previously approved specific applications of the PANACEA, TRACG, and PRIME codes as part of the GESTAR-II methodology. No changes were necessary to the technical models as previously reviewed and approved by the NRC, therefore, the CRDA LTR focuses on validation of the PANACEA and TRACG methods for fast reactivity transients, a description of the key technical models used to confirm the acceptance criteria for the CRDA event, and a discussion of the analysis procedure that will be used to identify and analyze all configurations that need to be evaluated. Since the NRC review of the CRDA LTR depends, in part, on the assumption that the technical models for the PANACEA, TRACG, and PRIME codes have been previously reviewed and approved by the NRC for general neutronics, transient analysis, and fuel thermal performance applications, any limitations and conditions associated with these analysis codes remain applicable. This is expected to be controlled as part of the overall GESTAR-II methodology as maintained by GNF-A.

3.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) 50.34, "Contents of Applications; Technical Information," requires that the licensee/applicant provide safety analysis reports to the NRC detailing the performance of systems, structures, and components provided for the prevention or mitigation of potential accidents.

General Design Criterion (GDC) 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," addresses the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges. This regulatory requirement primarily applies to ensuring that the limiting system operating parameters and other controls in place (i.e., rod withdrawal limitations) are sufficient to ensure that the CRDA acceptance criteria are not exceeded. This is satisfied by ensuring that the initial conditions and limitations on rod withdrawal represented in the CRDA analyses are sufficiently representative of the most conservative condition allowed by the aforementioned controls.

GDC 28, “Reactivity Limits,” of 10 CFR Part 50, Appendix A, requires that the effects of postulated reactivity accidents result in neither damage to the reactor coolant pressure boundary greater than limited local yielding nor result in sufficient damage to impair significantly core cooling capacity.

As per 10 CFR 100.11, “Determination of Exclusion Area, Low Population Zone, and Population Center Distance,” and 10 CFR 50.67, “Accident Source Term,” radiation dose limits are established for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

The acceptance criteria for CRDA events to satisfy GDC 28, 10 CFR 100.11, and 10 CFR 50.67 are defined in Chapter 15, “Transient and Accident Analysis,” of the SRP (Ref. 5). Satisfying these acceptance criteria is necessary for CRDA events to meet the aforementioned regulatory requirements. Specifically, SRP Section 15.4.9.II, “Acceptance Criteria,” states in part the following acceptance criteria:

1. Acceptance criteria from SRP Chapter 4.2. Appendix B provides interim acceptance criteria for reactivity initiated accidents (RIAs).

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that causes stress to exceed the “Service Limit C” as defined in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 7).

SRP Section 4.2 provides an extensive discussion of acceptance criteria related to high temperature cladding failure, PCMI induced cladding failure, core coolability, and fission product inventory determination for dose assessment purposes. Regulatory Guides 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors,” (Ref. 8) and 1.195, “Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors” (Ref. 9) are also referenced for further guidance related to fission product inventories.

The NRC staff has published a draft regulatory guide, DG-1327 (Ref. 6), for public comment. This guide contains new guidance on RIA acceptance criteria that, when final, will supersede the guidance currently contained in SRP Section 4.2. As part of this review, the NRC staff considered the applicability of the LTR methodology to DG-1327 (Ref. 6). Where appropriate, the new RIA criteria along with any potential implications to acceptability of the LTR methodology are discussed in this safety evaluation.

The CRDA LTR is an application of an evaluation model to perform licensing analyses for an accident that the evaluation model has not previously been approved for. As such, additional guidance for the evaluation may be found in SRP Chapter 15.0.2, “Review of Transient and Accident Analysis Methods” (Ref. 5). This chapter includes provisions for the review of submittals related to evaluation models.

In summary, the NRC staff used the review guidance in SRP Chapter 15.0.2 along with the applicable acceptance criteria in SRP Chapters 4.2 and 15.4.9 in conducting its review of the CRDA LTR. The new acceptance criteria applicable to the CRDA event contained in DG-1327 was also considered, with the understanding that the guidance has not yet been finalized. In accordance with SRP Chapter 15.0.2, the review covered the areas of: (1) documentation, (2) evaluation methodology, (3) accident scenario identification process, (4) code assessment, (5) uncertainty analysis, and (6) quality assurance plan. To the extent possible, the NRC staff leveraged the prior review and approval of the PANACEA, TRACG, and PRIME analysis methodologies as incorporated in the GESTAR-II methodology (Ref. 2).

4.0 TECHNICAL EVALUATION

The CRDA LTR describes a methodology by which the PANACEA and TRACG codes approved in the GESTAR-II methodology (Ref. 2) can be extended to analysis of the CRDA event. The NRC staff review of the CRDA LTR focused on four specific areas:

1. Accident scenario description and phenomena identification and ranking – GNF-A’s break-down of the CRDA event and its relevant phenomena, and characterization of the consequences. [[ ]], the NRC staff utilized other available approved PIRTs and relevant guidance to inform their assessment of whether all the relevant phenomena are appropriately addressed in the validation basis, acceptance criteria, and/or procedure used to confirm that the acceptance criteria are met.

2. Evaluation methodology – the proposed CRDA analysis methodology, including initial conditions, assumptions, and approach to ensuring that the SRP Chapters 4.2 and 15.4.9 acceptance criteria are met. Since this methodology includes use of the evaluation model, by extension, this area includes the application of the evaluation model to analyze the CRDA event.

3. Code assessment – the assessments performed by GNF-A to validate the PANACEA and TRACG code systems performance for CRDA specific phenomena.

4. Uncertainty analysis – GNF-A’s evaluation and propagation of uncertainties in the analysis.

In addition, the NRC staff considered whether GNF-A provided adequate QA and documentation support for the CRDA methodology. This aspect is not explicitly discussed in detail for this safety evaluation because the bulk of the QA and documentation support is captured by the various QA program documents, code documentation, and methodology discussion associated with the prior NRC approval of the PANACEA, TRACG, and PRIME methodologies in GESTAR-II. The additional documentation required to address the CRDA methodology is largely captured by the CRDA LTR. As such, NRC staff acceptance of the adequacy of the licensee's discussion of each area implicitly includes acceptance of the licensee documentation associated with that area.

Each of the four aforementioned areas will be discussed and evaluated in the following subsections.

4.1 <u>Accident Scenario Description and Acceptance Criteria</u>

As per the review guidance in Chapter 15.0.2 of the SRP, the accident scenario description and phenomena identification and ranking process is intended to ensure that the dominant physical phenomena influencing the outcome of the given accident scenario are correctly identified and ranked. Once an accident scenario has been described, then figures of merit can be determined for use in evaluating whether acceptance criteria are met. The subsequent phenomena identification and ranking process will determine the physical phenomena affecting the FoMs and rank them by their importance. By doing so, an applicant can demonstrate that reasonable assurance exists that they are accurately capturing and modeling the dominant physical phenomena necessary for evaluation of the accident scenario in question.

Section 1.1 of the CRDA LTR briefly describes the accident scenario. The description of the CRDA event is consistent with other readily available documents, such as updated final safety analysis reports and other topical reports (TRs) related to BWR CRDA events. The scenario is relatively simple in that it consists of a rapid reactivity addition due to a single control rod falling out of the core. The resulting local power excursion is terminated primarily by Doppler reactivity feedback as the fuel temperature increases. Long term shutdown is assured by negative thermal hydraulic reactivity feedback and/or a reactor scram. The CRDA event may occur during startup or when the reactor is operating at full power. In the former case, constraints imposed on rod movements due to technical specification (TS) restrictions and rod withdrawal sequences may serve to limit the potential rod patterns and the resulting rod worths. In the latter case, the initial operating characteristics of the fuel and moderator lend themselves to more effective Doppler reactivity feedback and quicker thermal hydraulic reactivity feedback through increased voiding from direct moderator heating.

[[

]] a review of NRC guidance and technical bases to identify appropriate acceptance criteria and critical parameters or characteristics. Each item was then addressed in the CRDA LTR along with a justification. Specifically, Section 3.0 of the CRDA LTR discusses the relevant technical models utilized in the PANACEA, TRACG, and PRIME analysis methodologies for analysis of the CRDA event and identifies how the relevant output parameters are to be determined for comparison to the applicable acceptance criteria. The SRP 15.4.9.II (and, by extension, the interim RIA acceptance criteria in SRP 4.2 Appendix B) ~~acceptance criteria~~ **potential critical parameters** are: (1) fuel enthalpy, (2) minimum critical power ratio (MCPR), (3) peak system pressure, (4) fission product inventory released, and (5)

core coolability. The acceptance criteria in DG-1327 are based on the same parameters. Of these parameters, [[ ]] is addressed in the CRDA LTR. Section 4.4 of the CRDA LTR discusses the calculations performed to address the at-power CRDA scenario and indicates that the [[ ]], which is acceptable because the NRC regulatory guidance indicates that the MCPR is essentially a surrogate parameter that captures the conditions under which high temperature cladding failure would occur (i.e., dryout). The remaining three criteria only become meaningful in the event of fuel rod failure. The heat added to the coolant by CRDA events which do not result in rods exceeding the acceptance criteria is expected to be small relative to the heat capacity of the coolant, and no fission release or fuel geometry deformation would occur. Therefore, the NRC staff finds it acceptable that the CRDA LTR [[ ]] because the GNF-A methodology is based on [[ ]] to those which do not lead to any fuel rod failure.

The NRC staff reviewed the PIRTs for other RIAs, prior precedents for the CRDA event, and the NRC staff's technical understanding of the relevant events in the accident progression. In summary, other PIRTs include: (1) initial conditions that would affect initial enthalpy or reactivity feedback, (2) parameters that would affect the positive reactivity addition from the rod drop, (3) parameters that would affect the timing and/or magnitude of the negative reactivity feedback terminating the power excursion, and (4) parameters affecting the transfer of heat away from the limiting locations. For BWRs, past precedents and the studies discussed in Section 4.4 of the CRDA LTR show that in the absence of specific controls intended to minimize the potential consequences of the CRDA, the conditions which maximize the potential for fuel failures occur at cold zero power (CZP) conditions. This is because increased temperatures result in increased mitigation via Doppler and moderator reactivity feedback mechanisms (see Section 4.2.5.1 for further discussion). The short time scale for the CZP CRDA scenario means that thermal hydraulic feedback is of relatively little consequence, since the limiting parameters reach their maximum values before significant heat transfer occurs. Consequently, the primary phenomena affecting the CRDA event are expected to be those that affect the magnitude of the reactivity addition or the Doppler reactivity feedback. The specific technical models and parameters affecting the Doppler reactivity feedback, along with other parameters of moderate importance, are discussed in the CRDA TR.

In summary, the NRC staff has determined that GNF-A appropriately characterized the CRDA scenario, identified the appropriate acceptance criteria, and evaluated the sensitivity of the acceptance criteria to the technical models and input parameters used to perform the CRDA evaluation.

4.2 Applicability of Evaluation Model to CRDA Event

Chapter 15.0.2 of the SRP describes the review of the evaluation model as part of the transient and accident analysis methods. The associated acceptance criteria indicate that models must be present for all phenomena and components that have been determined to be important or necessary to simulate the accident under consideration. The chosen mathematical models and the numerical solution of those models must be able to predict the important physical phenomena reasonably well from both qualitative and quantitative points of view. Restated in terms of the review procedures provided in Section III of Chapter 15.0.2, it must be determined if the physical modeling described in the theory manual and contained in the mathematical

models is adequate to calculate the physical phenomena influencing the accident scenario for which the code is used.
Each of the proposed codes (PANACEA, TRACG, and PRIME) have been evaluated and found to be acceptable for specific applications during the review and approval of a number of individual TRs (Refs. 10, 11, and 12). No changes or enhancements to the technical models in the codes are being proposed for NRC review and approval. As a result, this review focused on how the methodologies, as implemented by the codes, are applied to analyze the CRDA event. The scope of this review included the applicability of the modeling schemes discussed in the previously approved TRs to the CRDA event, and any potential limitations to the proposed analysis procedure to identify and assess the limiting CRDA scenarios.

4.2.1 <u>Applicability of PANACEA Technical Models to CRDA</u>

PANACEA is currently approved primarily for use in steady state methodologies used to establish the core design for reload licensing, monitor thermal limits, and perform selected calculations. GNF-A intends to use the steady state reactivity calculation capability in PANACEA to determine static control rod worths as part of the proposed CRDA methodology, which is discussed further in Section 4.2.1.1 of this SE. PANACEA has not formally been approved for transient calculations, however, the transient neutron kinetics model in PANACEA is identical to the model in TRACG. TRACG has been approved for AOO and stability related applications, where neutronic feedback is important. Therefore, the NRC staff review of the use of the PANACEA for transient calculations focused on the applicability of the models for their intended purpose in CRDA analyses, as discussed in Section 4.2.1.2 of this SE.

4.2.1.1 PANACEA – [[ ]]

The steady state neutronics calculational capabilities of PANACEA are currently used to perform shutdown margin calculations as part of GESTAR II (Ref. 2), which are essentially static control rod worth calculations based on an all rods in (ARI) configuration. PANACEA has been extensively benchmarked for normal core operations, which involves various rod configurations. The proposed use of the PANACEA steady state neutronics capabilities is merely in its use to compute the reactivities corresponding to the initial and final rod positions for possible rod drop scenarios. The [[ ]] for each rod drop scenario can then be defined as the difference in reactivity between the initial and final position for the postulated drop scenario.
[[

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The [[

]]
The most important requirement is that PANACEA be capable of calculating [[ ]] for different core configurations. The proposed calculations are similar to other existing calculations for NRC approved applications, and PANACEA has been benchmarked extensively against BWR core operations. As a result, reasonable assurance exists that any

calculated [[ ]] will be consistent with the data used to develop [[ ]].

Based on the calculational capabilities of PANACEA and the [[ ]] CRDA rod enthalpies, the NRC staff concludes that the use of PANACEA to determine [[ ]] as described in the CRDA LTR is appropriate.

4.2.1.2 PANACEA – Transient Calculations for CRDA

PANACEA contains a transient neutronic kinetics model that was previously reviewed and approved by the NRC as part of its review of the TRACG package. The thermal hydraulic models in PANACEA are much more limited than TRACG, however, the heat transfer to the surrounding coolant is minimal during the prompt power excursion. The SPERT III assessments (see Section 4.3 of this SE) demonstrate that the PANACEA transient models perform a reasonable job of capturing the prompt power excursion. According to the CRDA LTR, [[ ]].

The use of the pin power reconstruction methodology was assessed as part of the validation of the PANACEA methodology that was approved by the NRC (Ref. 10). The supporting LTR and references describe validation that was performed using cold and hot conditions. The main purpose of the pin power reconstruction methodology is to capture the impact of highly localized flux conditions experienced by individual pins, such as the presence of a nearby control rod. The validation suite referenced in the PANACEA licensing TR (Ref. 10) [[ ]]. The pin power reconstruction methodology itself was validated by comparison to calculations performed using DIF3D, a code developed by Argonne National Laboratory, as well as gamma measurements and TIP data at hot conditions for operating power plants. While the code-to-code comparisons were representative of pin powers [[ ]].

The PANACEA transient calculations do not capture the changes in heat transfer to the coolant as the fuel rods heat up, or the negative moderator feedback that would be expected from heatup and potential boiling of coolant. Section 4.2.4.6 of this SE contains further discussion regarding the conservatism associated with the approach used to determine the [[ ]] the PANACEA code as well as the inputs and process to determine [[ ]].

Based on the fact that the PPR methodology is expected to be applicable to the conditions being analyzed, and the conservatisms discussed in Section 4.2.4.6 of this SE associated with

the [[ ]], the NRC staff concludes that the calculated maximum rod enthalpies can reasonably be expected to bound the actual maximum rod enthalpies if the same scenario were to occur in the real world.

4.2.2 Applicability of TRACG Technical Models to CRDA

The actual analyses of limiting CRDA events are performed using the models and correlations in TRACG that have been previously reviewed and approved by the NRC for a broad set of transients. While TRACG was not reviewed with the CRDA event in mind, the NRC staff did review the key models necessary for accurate predictions of the CRDA specific phenomena, as follows. Note that several of these models utilize correlations that are updated and/or validated for each new fuel assembly design, in compliance with the process described in GESTAR**-II** ~~III~~ (Ref. 2).

- Bundle void correlations – supports void distribution calculations (for at-power CRDA)
- Transient neutron kinetics model qualification – supports calculation of neutronic response
- Gap models – supports calculation of heat transfer from fuel to coolant, as well as rod internal pressure (based on inputs from PRIME, see Section 4.2.3)
- Pressure drop and critical power tests – supports applicability of CPR calculations (for at-power CRDA)
- Peach Bottom turbine trip test – supports ability of TRACG to predict neutronic/thermal hydraulic coupled feedback (for at-power CRDA)
- Direct moderator heating model – specifies amount of fission heat that is deposited via gamma heating of moderator **and the slowing down of neutrons in the moderator**, which affects magnitude of fuel rod enthalpy increase and moderator density reactivity feedback (validated at cold conditions)

Additional model integral test assessments were provided to support the ability of TRACG to accurately evaluate the CRDA event for startup conditions based on tests performed at the Special Power Excursion Reactor Test III (SPERT III) reactor, as discussed in Section 4.3 of this SE.

Some of the models within TRACG are used in specific ways for the purposes of the CRDA analyses, as follows:

- [[ ]]
- The fission gas inventory is predicted by PRIME [[ ]].
- The fission gas inventory is increased [[ ]] to account for transient fission gas release.

None of these proposed modeling approaches would invalidate the basis for prior NRC approval of the relevant technical models. The acceptability of the modeling approaches is discussed further in Section 4.2.4 of this SE.

The CRDA LTR describes the technical models utilized within TRACG to perform analysis of the CRDA event. The majority of the technical models have previously been validated for other applications in which the neutronics and thermal hydraulic phenomena of interest for the CRDA event are important. Additional validation is discussed in Section 4.3 of this SE related to the unique conditions that exist for a CRDA during cold startup, where the primary mitigating factor is the Doppler reactivity feedback. As discussed above, the specific models used in TRACG to obtain the necessary information for comparison to the acceptance criteria (see Section 4.2.5.4) are applied in a conservative manner, and therefore, the approach is found to be acceptable by the NRC staff for use in analysis of the CRDA event.

4.2.3 Applicability of PRIME Technical Models to CRDA

The PRIME fuel thermal mechanical performance methodology is not used directly in the analysis of the CRDA event, however, several inputs for TRACG are derived from PRIME. For example, the data needed for the TRACG dynamic gap model is obtained from PRIME and is directly applicable based on the NRC review and approval of the implementation within TRACG. Additionally, the fission gas inventories assumed in the gap for the TRACG calculation are obtained from PRIME. The gap inventories are developed based on PRIME's steady state depletion capability to determine the fission gas production and release from the fuel pellet as a function of exposure, LHGR history, and instantaneous LHGR. This capability has been reviewed and approved by the NRC (Ref. 12) and is appropriate for use directly in the TRACG calculation. As discussed in Section 4.2.2 of this SE, the PRIME generated data [[ ]].

The CRDA LTR states that the PRIME fuel files [[ ]] are assumed to be appropriate for use as inputs [[ ]] in TRACG for CRDA analyses. Use of data from [[ ]]. The justification given for use of data from fuel rods [[ ]]. Gadolinium does have a second order effect in reducing the pellet thermal conductivity slightly. The impact of a slightly reduced pellet thermal conductivity on the calculated pellet enthalpy values is expected to have a negligible impact on the prompt enthalpy rise, since it occurs prior to any significant heat transfer from the fuel pellet. The relatively small potential effect on the total enthalpy, due to slowing of heat transfer from the fuel pellet, is expected to be less than the conservatism inherent in use of a [[ ]] for the gadolinium-bearing fuel, [[ ]].

The PRIME technical models used to produce input data for the CRDA analyses have previously been reviewed and approved by the NRC, and the application of the data for CRDA analysis purposes is consistent with NRC approved applications. The NRC staff evaluated the applicant's description of how the PRIME information would be used within TRACG for CRDA analyses and found the proposed approach to be acceptable.

4.2.4 Modeling Guidance

The CRDA LTR indicates that the general plant model is consistent with the models created for the application of PANACEA and TRACG to analyze AOO, ATWS, and stability events (Refs. 10) (and 11). Several of these transients require accurate predictions of rapidly changing axial flux shapes. When combined with the assessment against data from the SPERT III experiments (see Section 4.3), the NRC staff finds that reasonable assurance exists that the overall modeling as described in the licensing TRs describing the aforementioned PANACEA and TRACG applications is acceptable for use in modeling the CRDA event. A few specific modeling and input parameters are adjusted to accommodate the unique needs of the CRDA analysis procedure. These parameters are discussed in the following subsections.

4.2.4.1 TRACG Channel Grouping, Vessel Nodalization, & Time Step Guidance

The CRDA event primarily impacts the fuel assemblies grouped near the control rod of interest, so computational time savings can be realized by [[ ]] far from the control rod of interest. This is a strategy in which [[ ]]. This modeling approach effectively [[ ]] calculating the thermal hydraulic properties for every individual assembly.

When this type of approach is adopted, to ensure that the results are not non-conservative, the guidance for grouping channels must be established in a manner that ensures that:

1. Individual fuel channels are modeled when necessary, to capture highly localized limiting phenomena;

2. Fuel channels that are [[ ]] do not lead to a change in the hydraulic response of the channels of interest; and

3. Any other variations in input parameters would yield equivalent or conservative results relative to a higher resolution model.

The CRDA LTR describes the approach used to determine how to select the fuel channel groups. First, the individual fuel assemblies are explicitly modeled as individual channels [[ ]] for the drop evaluation. Secondly, the [[ ]]. Finally, the vessel is nodalized such that the [[ ]].

Requirement (1), above, is met by the use of individual fuel channels [[ ]] surrounding the target rod for the drop evaluation. The use of individual fuel channels also ensures that the hydraulic response of the channels of interest are explicitly captured, and the non-channel parameters (e.g., bypass flow) are captured with reasonable accuracy by the vessel nodalization strategy. Figure 4-2 of the LTR provides some representative enthalpy rises for fuel around a target dropped rod, showing that the enthalpy response [[

]].

The averaging of specified properties in areas less important to the enthalpy calculation may lead to variations in the peripheral neutron flux or changes in the boundary conditions for hydraulics. In general, averaging of thermal hydraulic quantities for areas closer to the target rod with those for areas farther from the target rod will lead to more conservative results due to the fact that averaging the moderator temperature and density for fuel close to the region of interest with fuel farther from the region of interest leads to a suppressed negative Doppler reactivity feedback response due to the lower temperatures of the farther fuel. A similar logic can be used to infer that other influences such as variations in burnup or power within a channel grouping would yield slightly more conservative results due to the dampening of the Doppler reactivity feedback mechanism for the more reactive fuel elements in the group. Therefore, the grouping strategy is primarily an attempt to simplify the calculation without becoming overly conservative in doing so. Hence, the above requirements (2) and (3) are met.

The CRDA LTR does not go into details regarding how the thermal hydraulic behavior for mixed cores will be treated. However, the fact that the fuel assemblies [[

]]. Based on the [[

]], the NRC staff did not find it necessary to review a detailed description of the [[

]].

The axial nodalization of the fuel channels is consistent with the accepted nodalization of the PANACEA and TRACG methodologies, as supported by their respective assessment bases. The CRDA event is primarily analyzed at zero power conditions when the coolant is below saturation (and thus more or less uniform except for pressure changes as a result of flow resistance and elevation), and the changes in the coolant properties outside the fuel channel boxes is minimal. As a result, the nodalization of the reactor vessel and other components is relatively little importance but is reasonable. The CRDA LTR, as submitted, lacked information regarding guidance for time step sizes. Since the CRDA event is a very rapid transient that may require shorter time steps relative to other transients, the NRC staff asked RAI-5 to better understand the sensitivity of the TRACG results to the time step size. In its response (Ref. 13), GNF-A provided justification that the internal logic used by TRACG to adjust the time step size coupled with the standard time step inputs will not **allow**~~limit the~~ time steps ~~to~~ sizes larger than necessary to capture the prompt power excursion. The code assessment (see Section 4.3 of this SE) provides additional assurance that the time step logic within TRACG is applicable to the CRDA event.

As a result of the above considerations regarding the potential impacts of the channel grouping strategy and vessel nodalization on the results calculated for the CRDA event, the NRC staff finds the proposed channel and vessel modeling strategy to be acceptable.

4.2.4.2 TRACG Reactivity Insertion Modeling

The CRDA event includes up to two separate reactivity insertions **due to control blade movement**. The first is the control rod drop that triggers the event, which is (for limiting cases) a positive reactivity insertion that is sufficient to cause a prompt excursion. The second **negative reactivity insertion** may not always be necessary to terminate the event, but if necessary, a scram is expected to occur based on the flux-based or period-based trip functions of the various core monitoring systems. The CRDA LTR describes a conservative modeling of both reactivity insertions.

The control rod is assumed to begin falling instantaneously at a speed of 3.11 ft/s. This speed is the maximum possible drop speed based on the velocity limiter associated with the control rods tested as documented in NEDO-10527 (Ref.14). This maximum drop speed would need to be confirmed for all control rod designs in the core that were not included in NEDO-10527. Since the control rod will begin accelerating from a resting position and may not reach the maximum velocity, this is a conservative approach in ensuring that the positive reactivity insertion occurs as quickly as possible. Since the maximum drop speed is a key assumption, a limitation and condition is imposed to ensure that this assumption remains valid for all future control rod designs that are not included in NEDO-10527.

The reactor scram trips [[

]].

The CRDA LTR describes highly simplified inputs for the reactivity insertions corresponding to the rod drop itself and any subsequent reactor scram (if needed). As discussed above, the simple inputs are inherently conservative, and therefore, acceptable for use in analysis of the CRDA event.

4.2.4.3 TRACG Fuel Rod Fission Gas Inventory

The inputs to be used at the beginning of the CRDA analyses include some key assumptions associated with the fuel rod fission gas inventory. The fission gas inventory is used to determine the rod internal pressure, which is needed for evaluation of the high temperature rod failure acceptance criterion, and to compute the thermal conductivity of the gap between the fuel rod and cladding. In both cases, a higher fission gas inventory is more conservative. High temperature rod failure may be predicted to occur at lower total enthalpies when the rod internal pressure is higher, which is directly proportional with the fission gas inventory due to the fixed available volume in the gap. A larger amount of fission gas in the gap leads to degraded heat transfer capability across the gap, which may increase the total enthalpy due to greater heat retention within the fuel during the trailing “tail” of the power excursion.

[[ ]]

Secondly, the initial fission gas inventory at the beginning of the CRDA analysis was [[ ]].

As a result of the above considerations regarding the potential impacts of the fission gas inventory on the results calculated for the CRDA event, the NRC staff finds the proposed fission gas inventory modeling strategy to be acceptable.

4.2.4.4 Initial Parameters

As part of the input description in the CRDA TR, GNF-A provided guidance and justification for the core operating parameters that should be assumed during the CRDA analysis, namely the coolant temperature, power, and flow. In order to evaluate the impact of the initial conditions on the results from the CRDA event, GNF-A performed a series of sensitivity studies using a worst-case drop scenario for a representative plant at BOC, MOC, and EOC.

The coolant temperature has a strong effect on the calculated enthalpies from the limiting CRDA event, with the enthalpies increasing as the initial coolant temperature reduces to cold conditions. This is consistent with trends that the NRC staff has observed in similar studies. [[ ]]

The power and flow sensitivity studies showed [[ ]]. The NRC staff has seen sensitivities in other studies that exhibited [[ ]] to the power and flow, however, this effect may be dependent on the reference plant/fuel used in the calculation or the specific methodology being utilized. The CRDA analysis methodology being proposed in the CRDA LTR contains [[ ]], so any small sensitivities would not affect the ability of this methodology to demonstrate compliance with regulatory limits.

The NRC staff reviewed the recommended input parameters for the initial conditions of the core, and the information presented to support the recommendations. The recommended initial core temperature will be set in a way that ensures that it bounds the intended application, [[ ]]. Specific applications may exhibit [[ ]], however, there is sufficient conservatism in the methodology to accommodate these kinds of variations (see Section 6.0 of this SE for further discussion). As a result, the NRC staff finds the initial condition input recommendations to be reasonable.

4.2.4.5 Doppler Coefficient Application

The most important phenomena for mitigation of the CRDA event is the Doppler reactivity feedback, which arrests and largely reverses the prompt power excursion that may occur after a rod drop. Consequently, the consequences of the CRDA event are expected to be very sensitive to how the Doppler reactivity feedback is modeled in the analysis methodology. In the CRDA TR, GNF-A states that TRACG utilizes [[ ]].

The NRC staff asked RAI-1 to better understand the behavior of the [[ ]] Doppler coefficients [[ ]] to be applicable to all fuel assembly designs. In response to this RAI (Ref. 13), GNF-A provided a more detailed comparison [[ ]] Doppler coefficients calculated with TGBLA (which is capable of performing the explicit lattice calculations at cold conditions). [[ ]] However, the trends are not conclusive and the technical basis for this behavior is not well understood. Instead, GNF-A addressed the relevant considerations through the following technical bases:

1) Applicability to fuel assembly designs or lattices of interest – GNF-A performed the comparative calculations to determine [[ ]] GNF2 and GNF3 fuel assembly designs. These two fuel assembly designs represent most of the GNF-A fuel assembly designs currently in operation. The exclusion of the [[ ]] is reasonable for typical configurations because the higher reactivity region [[ ]], **when the CRDA cases are most limiting**~~to the gradual upward shift in power for burned fuel as the bottom lattices become depleted~~. To ensure that the [[ ]] new fuel assembly designs, GNF-A proposed an addition to their procedure for performing TRACG analyses which will be applied [[

]]. This addition ensures that if a new fuel assembly design is utilized or the limiting enthalpy response occurs [[ ]] must be re-evaluated.

2) Applicability to a range of U-235 enrichments, Gd enrichments, and number of Gd pins – Different lattice nuclear designs were investigated to determine whether any strong sensitivities existed. An extreme scenario was included in the RAI response, where the [[ ]] are reasonably bounding. The overall results show that the [[ ]] is conservative for a reasonable typical range of pin compositions.

3) Doppler feedback uncertainty treatment – The conservatism in the [[ ]] uncertainty in the Doppler feedback. As shown in Figure [[ ]] of the CRDA LTR, [[ ]] based on comparison of Figure [[ ]] of the CRDA LTR with Figure [[ ]] from the RAI-1 response, [[ ]]. This treatment also effectively treats the uncertainty in the Doppler feedback [[ ]].

The NRC staff noted that there are some theoretical scenarios where the Doppler feedback [[ ]] may be significantly lower [[ ]]. However, such scenarios are expected to be rare, since the limiting fuel assembly as well as other adjacent fuel assemblies making significant neutronic contributions to the power excursion would need to coincidentally be at a burnup at or near the narrow ranges of burnups for which the Doppler feedback [[ ]]. This is more likely to occur at BOC, which is a statepoint with low risk significance for a CRDA event, because: (1) the delayed neutron fraction is higher, requiring insertion of larger control rod worths to produce prompt criticality, and (2) the enthalpy required for fuel failure will be high due to the lower rod internal pressures and low hydrogen pickup for the fuel assemblies at these burnups. Therefore, the overall Doppler feedback modeling, as described in the CRDA LTR, is expected to be sufficient to address the Doppler feedback uncertainty for the limiting scenarios.

The above considerations are based on the demonstration provided in the CRDA LTR using lattices from the GNF2 and GNF3 fuel assembly designs. Other fuel assembly designs are expected to yield comparable results, [[ ]]. Some variation may occur in the exposure ranges for which the Doppler [[ ]], however, the NRC staff finds that such variations would be accommodated by the inherent conservatisms in this methodology, as discussed in Section 6.0.

The approach used within PANACEA and TRACG to calculate the Doppler reactivity feedback has previously been reviewed and approved by the NRC for hot operating conditions, as part of the methodologies for analyzing other transients (Refs. 10 and 11). The conditions for the limiting CRDA event occur at cold conditions, which means that the temperature profiles within the fuel pellet will be different. Initially, the temperature across the fuel pellet will be uniform. When the power excursion occurs, the temperature increase will be proportional to the power generation within the pellet. The radial power profile for the pellets in fresh and burned fuel will differ, because the power generation will be significantly higher near the outer surface of the pellet for burned fuel due to the "rim effect" (increased Pu production relative to the interior of the pellet due to self-shielding). This may cause the Doppler reactivity response to differ from that expected during hot conditions. However, as discussed in Section 4.2.4.6 of this SE, [[ ]] TRACG to determine the final peak rod enthalpies are developed using a conservative approach. The NRC staff finds, based on engineering judgment, this inherent conservatism [[ ]] is sufficient to accommodate any small potential effects due to variations in the Doppler reactivity response based on differing pellet radial power distributions (due to self-shielding).

The approach used to predict the Doppler reactivity feedback is one of the most important aspects of the CRDA analysis methodology, since this is the primary source of accident mitigation. The NRC staff evaluated how the Doppler reactivity feedback is evaluated in TRACG and determined that the approach can reasonably be expected to yield conservative values for the peak rod enthalpies used for comparison to the acceptance criteria for the CRDA event. Additionally, the uncertainty in the Doppler reactivity feedback is treated in an acceptably conservative manner.

4.2.4.6 Fuel Rod Enthalpy Determination

One of the major output parameters from the calculations done to predict the consequences of a CRDA event is the peak fuel rod enthalpies for fuel assemblies surrounding the dropped control rod. The overall thermal hydraulics and neutronics response of the fuel and surrounding coolant is captured by TRACG, using neutronics inputs from PANACEA, on a nodal basis [[ ]] the overall thermal hydraulic or neutronics calculations in TRACG, [[ ]] to predict the coupled feedback mechanisms. This is consistent with what the NRC has previously reviewed and approved for the use of TRACG in analysis of other events, as well as general practices in the industry. The additional information that is needed to determine the enthalpy [[ ]] enthalpy for all rods is discussed in the next paragraph.

The CRDA LTR states that the enthalpy for individual rods is determined through use of a [[

]]. There are two separate enthalpy acceptance criteria, one based on the maximum enthalpy value attained during the transient (“total enthalpy”) and one based on the increase in enthalpy during a defined time interval (“prompt enthalpy rise”), which are discussed further in Section 4.2.5.4 of this SE. However, all rods are initially at the same enthalpy for CZP conditions, so the limiting rod will be the same for both enthalpy criteria.

The approach used by GNF-A to determine the limiting enthalpy values for comparison to the acceptance criteria for the CRDA event was found to be acceptable by the NRC staff, based on standard practices for this type of analysis, qualification of the codes used to perform the calculations, and several conservatisms, as described above.

4.2.5 CRDA Analysis Procedure

The CRDA LTR provides a specific procedure for performance of the CRDA analysis, which includes a description of what conditions should be evaluated, which control rods should be selected for evaluation, and how the acceptance criteria should be verified to have been met. Section 4.2.5.1 discusses the at-power CRDA scenario, and the remainder of the subsections discuss the CZP CRDA scenario.

4.2.5.1 Hot/At-Power/Intermediate CRDA Scenario

Section 4.4 of the CRDA LTR focuses on the range of applicability for the proposed CRDA analysis procedure. It strives to define the core conditions for which the CRDA event is clearly non-limiting. GNF-A does this by performing sensitivity studies to define a minimum power level and minimum reactor dome pressure for which the CRDA event is no longer limiting. The at-power CRDA scenario is distinguished from the CZP CRDA scenario by the presence of increased negative reactivity via the following mechanisms:

1. The presence of significant voiding in the coolant results in less moderation, so neutron spectrum skews more towards faster neutrons (i.e., the spectrum is “harder”). Consequently, the control rod absorber material is less effective at neutron absorption (i.e., rod worths are lower) and the reactivity consequence of the rod drop itself is milder.

2. The coolant is at saturated conditions, so the direct heating of the coolant can produce voiding. This produces a significant negative moderator density feedback effect that is not present for CZP conditions where the direct coolant heating does not result in a significant change in the coolant density.

3. While the magnitude of the Doppler reactivity coefficient tends to be smaller at higher fuel temperatures, the harder neutron spectrum results in a larger number of neutrons available for Doppler capture in the resonance regions.

All of these mechanisms only come into play when the coolant reaches saturation conditions. GNF-A performed a series of analyses for different core dome pressures and power levels at saturation condition, and the results generally confirm that the results from the at-power CRDA analyses are much less limiting than the results from the CZP CRDA event, despite the lack of the restrictions on control rod positions that normally ensure that the CRDA does not result in fuel rod failures. To provide convenient triggers for plant operators to identify when the CRDA event can safely be assumed to have become non-limiting enough to preclude the need to follow the rod withdrawal sequence explicitly, GNF-A chose 5 percent power as the Low Power Set Point (LPSP) and 300 psig as the Low Dome Pressure Set Point (LDPSP). Anything below these limits would require adherence to an analyzed rod withdrawal sequence and associated requirements.

The calculations performed by GNF-A show that all predicted limiting enthalpies at saturation conditions are much less limiting than the maximum enthalpies at cold conditions. The primary consideration is to ensure that the core is at saturation conditions, so that void reactivity feedback begins to make a significant contribution to mitigation of the consequences from a CRDA event. A power level of 5 percent or a reactor dome pressure of 300 psig would both clearly indicate that the coolant has heated to saturation conditions, based on core heatup practices employed by plant operators. The 5 percent limit on power accounts for the inherent uncertainties associated with power measurement under low flux conditions, and the 300 psig limit on reactor dome pressure is significantly higher than any expected measurement uncertainties relative to saturation at atmospheric conditions. The NRC staff did note that some of the maximum total enthalpy values may increase significantly with increasing power. However, this is due to the initial peak enthalpy being higher for hot rods that are already operating at partial power or starting from a higher fuel average temperature. The enthalpy rise is still mitigated such that the total enthalpy values remain non-limiting relative to the CZP CRDA values.

The analyses performed by GNF-A for CRDA events beyond cold conditions are limited and not necessarily conclusive to prove that the CRDA event will be non-limiting for hot and intermediate power conditions for all plants and core loadings. However, these results are consistent with previous analyses of the CRDA event using other methodologies and the NRC staff’s understanding of the relevant phenomena. Therefore, the NRC staff finds the information presented in the CRDA LTR to be acceptable to demonstrate that the CRDA event continues to be limiting at cold conditions, and furthermore, that GNF-A identified reasonable setpoints for

plant operators to use in identifying when rod withdrawal banking requirements are no longer needed to ensure that a CRDA event does not result in fuel rod failures.

4.2.5.2 CZP CRDA Scenario: [[ ]]

The methodology described in the CRDA LTR to verify that the CRDA acceptance criteria are met on a plant/cycle specific basis [[ ]]. The NRC staff agrees that there will be a strong correlation between [[ ]], given that the power excursion is almost completely defined by two parameters—the reactivity insertion as defined by the control rod worth and the Doppler coefficient (which itself is largely defined by the initial core temperature). [[ ]] the hydrogen content and rod internal pressure, which are the parameters other than rod enthalpy that are necessary to evaluate the CRDA acceptance criteria. The hydrogen content and rod internal pressure are strong functions of exposure, since hydrogen uptake can be correlated with exposure (as in the NRC provided best estimate hydrogen uptake model used by GNF-A) and the rod pressure is proportional to the total fission gas inventory in the gap (which increases with exposure).

Use of the PPE value [[ ]] ensures that the hydrogen concentration and transient fission gas release (FGR) for all fuel rods are bounded, because the rod failure thresholds are more limiting for higher exposures and the PPE is used directly as the basis for determining the hydrogen pickup and transient FGR for a given exposure. The PRIME calculations used to generate the steady FGR [[ ]]. The NRC staff noted that the hydrogen concentration is not a linear function of exposure, [[ ]].

[[ ]] are validated through a series of TRACG calculations that utilize conservative initial conditions (as determined through sensitivity studies, if necessary) that evaluate a postulated startup sequence. The TRACG modeling is discussed in greater detail in Section 4.2.4 of this SE. The postulated startup sequence is designed to achieve control rod patterns with dropped control rod worths that are [[ ]]. This may include consideration of out of sequence control rods (see Section 4.2.5.3.3, “Allowed Out of Sequence Control Rods,” of this SE for further discussion). The control rod patterns may not match the actual control rod sequences that are developed by plant operators for a given cycle, however, the intent is to accurately capture the control rod worth, which is the driving force behind the prompt power excursion that defines the CRDA event. Any other significant influences are captured by limiting the applicability [[ ]].

[[ ]] may be developed for different initial core temperatures, [[ ]]. This is intended to provide flexibility to plant operators by allowing use of less restrictive rod banking sequences when the core is at higher temperatures, since the consequences of a CRDA can clearly be demonstrated as being bounded by lower temperatures [[ ]] (see Section 4.2.4.4 of this SE). [[ ]] Therefore, an initial core temperature [[ ]] can be used as a representative temperature [[ ]], to confirm that the rod withdrawal sequences are acceptable until the LPSP or LDPSP discussed in Section 4.2.5.1 of this SE are met.

The NRC staff noted that [[ ]]. Since some of the aforementioned additional fuel assemblies may experience enthalpies that approach the limiting enthalpies [[ ]], the NRC staff asked RAI-2 to better understand how the [[ ]] surrounding the dropped rod. In the RAI response, GNF-A indicated that even though [[ ]]. The NRC staff agrees with this explanation; however, the NRC staff also notes that an unstated assumption in this approach is that the results from the fuel rod failure criteria for the surrounding fuel assemblies is [[ ]]. This assumption is acceptable because: (1) [[ ]], and (2) [[ ]].

The CRDA LTR described [[ ]] but did not satisfactorily describe [[ ]]. Since Section 4.2 of the CRDA LTR states that [[ ]], the NRC staff asked RAI-3 to understand how [[ ]]. This RAI was primarily intended to address the potential for use of [[ ]]. In the response to RAI-3, GNF-A proposed [[

]]. The NRC staff agrees that [[ ]].

Overall, the NRC staff found that the [[ ]] process described in the CRDA LTR, as updated by the RAI responses, is an acceptable approach to develop [[ ]]. In some cases, assumptions implicit in the process used to develop [[ ]] may not be sufficient to assure that all possible results are bounded because they neglect specific local characteristics that may be important for evaluating whether individual rods fail. [[ ]] also do not explicitly check for high temperature fuel failures, though the quantities used [[ ]] are expected to be correlated to the high temperature fuel failure criteria. However, the TRACG enthalpy calculation demonstration in Section 5.1.3 of the CRDA LTR clearly shows that even for rod worths [[ ]] TRACG evaluation shows significant margin to both the PCMI and high temperature failure thresholds. The NRC staff also notes that there are several conservatisms inherent in the methodology (as discussed in Section 6.0 of this SE) which would provide additional margin for "outlier cases" where [[ ]] process may also consider out of sequence control rods, as discussed in Section 4.2.5.3.3.

In conclusion, the NRC staff has evaluated the guidance described by GNF-A in the CRDA LTR and RAI responses for the purpose of developing [[ ]]. The NRC staff understands that use of the acceptance criteria will be subject to the requirements described in the CRDA LTR, as updated by the RAI responses. Based on the described process and requirements for use of a given set of criteria, the NRC staff finds the approach proposed by GNF-A [[ ]] is acceptable.

4.2.5.3 CZP CRDA Scenario: Analysis Procedure

The analysis procedure involves defining several inputs for a plant-specific, cycle-specific CRDA evaluation. The intent of the methodology described in the CRDA LTR is to allow for development **of** control rod withdrawal sequences [[ ]]. The reference loading pattern for the given cycle is appropriate for use, since any changes in control rod worth (which drives the CRDA response) due to changes which do not require any core redesigns or re-evaluation of the reload licensing basis are expected to be minimal. The initial core conditions are discussed in Section 4.2.4.4. The analysis is performed for all steps starting from [[ ]] until the point at which the at-power CRDA disposition becomes applicable (as discussed in Section 4.2.5.1 of this SE). This is acceptable because a [[ ]] reactivity anomaly band has previously been justified for GNF-A methodologies (Ref. 15). The remaining inputs are determined through specific procedures, as discussed below.

4.2.5.4 Cycle Exposure

The CRDA LTR, as submitted, included [[ ]] options to define the exposures for which each CRDA analysis is to be performed. [[ ]] To better understand how these options would be applied, the NRC staff asked RAI-4 to obtain further detail on how the proposed approach would ensure that the [[ ]]. In response, GNF-A proposed [[ ]] The resulting options are:

[[ ]]

These options provide reasonable flexibility to licensees in optimizing their banking requirements to meet their needs or preferences, while ensuring that the CRDA analysis results can be applied across the entire cycle length.

4.2.5.5 Control Rod Withdrawal Order

The control rod withdrawal sequence used by the plant operator can be defined through defining three constraints on the sequence: the rods assigned to each group, the order in which each control rod group is withdrawn, and the order for rod withdrawal within a group. Out of these three constraints, the first two are explicitly defined by the plant operator as part of the basis for the CRDA analyses. If the control rod groups or group withdrawal order is modified, this would require re-evaluation of the CRDA event. However, the last constraint, the order in which control rods are withdrawn within a given control rod group, may be specified in a more flexible manner.

[[ ]] options are provided for plant operators to specify the rod withdrawal order within a rod group:

1. Fixed order – the entire control rod withdrawal order is pre-determined and cannot be altered.
   [[

]]

Option 1 ensures that the control rod withdrawal order is consistent with the CRDA evaluation.

[[

]]

4.2.5.6 Allowed Out of Sequence Control Rods

Typical plant TSs allow a predetermined number of control rods to be inoperable. Additionally, a plant operator may find it necessary to deviate from an analyzed withdrawal sequence by leaving a control rod fully inserted when the sequence prescribes that the rod should be withdrawn. The CRDA LTR specifies 8 control rods as a typical number, though a plant operator can specify any number of out of sequence control rods as part of their analysis input requirements. The out of sequence control rods are addressed as part of the CRDA evaluation using both PANACEA and TRACG, and the results may be used to support [[

]].

This approach may also not conservatively increase the [[

]]. Two considerations result in a low risk significance for this type of scenario becoming the limiting scenario for the CRDA event. First, the local reactivity for at least one symmetric location associated with the above postulated scenario will most likely be maximized [[ ]]. Second, the evaluation of fuel assemblies for which [[ ]] to become more likely due to reduced ductility of the cladding or higher internal rod pressure is done in a very conservative manner, as discussed in Section 6.0 of this SE, which is sufficient to offset the variations [[ ]] for symmetric locations.

The NRC staff finds that the approach described in the LTR for selecting out of sequence rods for evaluation of a given withdrawal order is acceptable because the most likely rod configurations that would challenge the CRDA acceptance criteria will be analyzed. The low risk and safety significance of potentially more limiting configurations does not warrant further constraints on use of this approach.

4.2.5.7 CZP CRDA Scenario: Evaluation Against Acceptance Criteria

For CRDA evaluations utilizing TRACG, the relevant output parameters are compared to the fuel rod failure threshold curves as provided by the NRC. [[

]] The CRDA LTR references a technical document (Ref. 16) that serves as the basis for draft regulatory guide DG-1327 (Ref. 6), which is intended to supersede the current acceptance criteria for RIAs (including CRDAs). As such, the failure threshold curves are acceptable for use in determining whether a fuel rod will be expected to fail based on enthalpy, rod internal pressure, and/or hydrogen content, based on available data. These curves are applied directly as discussed in Section 4.2.5.3 of this SE. The limiting total enthalpy is defined

as the maximum radially averaged enthalpy achieved by any fuel rod during the CRDA event. The limiting delta enthalpy is based a quantity called “prompt enthalpy rise” which only considers the enthalpy increase during a time interval equal to the width of the power pulse. This definition is consistent with the current NRC acceptance criteria for PCMI failure in SRP 4.2 Appendix B; this definition was carried over to the current version of DG-1327.

The CRDA LTR discusses a third potential mechanism for fuel failures, based on the cladding perforation model in TRACG that was developed for LOCA conditions. As discussed in the CRDA TR, [[ ]]. In a small number of cases, the TRACG perforation model [[ ]]. GNF-A indicates in the CRDA LTR that the failure possibility as predicted by the TRACG perforation model will be used in addition to the two sets of enthalpy based criteria in current NRC guidance. This is conservative in that it will increase the number of scenarios in which fuel rod failures are assumed. However, the perforation model has not been validated for the specific conditions associated with a CZP CRDA event and there is currently no research indicating that this phenomenon would be a significant concern. Therefore, the NRC staff is not drawing any conclusions about the applicability of the perforation model or its conclusions for CRDA scenarios.

As a result of the above discussion, the NRC staff finds that the proposed procedure is acceptable to confirm that the acceptance criteria for the CRDA event are met. This review considered the acceptance criteria for both SRP 4.2 Appendix B and the draft acceptance criteria in DG-1327. At this time, DG-1327 has not been finalized and may be subject to change, but as long as the basis for the above findings continue to remain valid in the final regulatory guidance, then the methodology outlined in the CRDA LTR should remain valid for use in demonstrating that NRC requirements are met.

4.3 Code Integral Assessment

Following the review guidance provided in Chapter 15.0.2 of the SRP, the next area of review for transient and accident analysis methods focuses on assessment of the code. The associated acceptance criteria indicates that all models need to be assessed over the entire range of conditions encountered in the transient or accident scenarios. The review procedures provided in Section III of Chapter 15.0.2 of the SRP also indicate that the assessment of these models is commensurate with their importance and required fidelity. This assessment is generally performed via comparison of predicted results against both separate effects tests and integral effects tests. Additionally, assessments must compare code predictions to analytical solutions, where possible, to show the accuracy of the numerical methods used to solve the mathematical models.

Separate effects tests are generally used to demonstrate the adequacy of individual models and the closure relationships contained therein. Complementary to these types of tests are integral tests, which are generally used to demonstrate physical and code model interactions that are determined to be important for the full size plant. In either case, some tests may not be full-scale, and, in demonstrating applicability to full-scale plant conditions, the tests may contain

scaling distortions. These distortions can affect both local and overall elements. It is therefore necessary to examine the nature of the tests involved in the assessments. The abilities of TRACG, with incorporation of key models and inputs from PRIME and PANACEA, has been assessed against integral and separate effect data and found to be acceptable for performing AOOs, stability, and ATWS calculations (Refs. 10, 11, and 12). These kinds of events and their associated validation databases provide a robust assessment of the capability of these codes to capture coupled thermal hydraulic-neutronics physics phenomena, along with the dynamic fuel rod thermal mechanical response. As a result, the majority of this section of the report will focus on the specific assessments that were performed to demonstrate that the codes provide adequate predictions of the phenomena of interest for the CRDA event.

Additional model integral test assessments were provided to support the ability of PANACEA and TRACG to evaluate the CRDA event for startup conditions based on tests performed at the Special Power Excursion Reactor Test III (SPERT III) reactor. These tests provide a valuable assessment of the ability of PANACEA and TRAC-G to capture the Doppler reactivity feedback, since the SPERT III reactor does not include moderator voiding and the power pulses are short enough to ensure that no significant heat transfer to the moderator occurs prior to the mitigation of the prompt power excursion due to Doppler reactivity feedback. As such, this assessment provides confidence that the PANACEA and TRACG codes will predict the Doppler reactivity feedback in the absence of other reactivity feedback mechanisms (such as void feedback, as captured by the Peach Bottom turbine trip tests that were included in the prior assessments). The data available from the SPERT test documentation (Rev. 17) has some notable limitations, including a lack of detail regarding the exact worth of the control rod used to simulate the rod drop and the speed of withdrawal. Therefore, some assumptions had to be made to model the tests in PANACEA and TRACG, but the key quantities, such as the reactivity insertion, were explicitly captured via the appropriate model parameters.

The assessment shows that PANACEA predicts the prompt power pulse from the SPERT III experiments well for a variety of different reactivity insertions. Only one calculation was performed with TRACG, for the test with the largest reactivity insertion. This is acceptable because the transient being simulated is so short that no significant heat transfer to the coolant occurs, therefore, the more realistic heat transfer features of TRACG will have little effect. [[

]] When this is taken into consideration, the results compare very favorably.

The NRC staff reviewed the previous assessments performed to support the use of the models and data computed from the PRIME, PANACEA, and TRACG methodologies to analyze AOO, stability, and ATWS events, and determined that they were applicable to demonstrate that

specific phenomena relevant to the CRDA event are appropriately assessed. The one significant assessment gap, related to determining the Doppler reactivity feedback in the absence of any other significant reactivity feedback mechanisms, was filled by assessing PANACEA and TRACG against data from SPERT III tests of rod ejection accidents. Therefore, the NRC staff has determined that the PRIME, PANACEA, and TRACG have been satisfactorily assessed for their abilities to model the relevant phenomena for the CRDA event, within the bounds of their intended applications within the CRDA analysis methodology.

4.4 Uncertainty Analysis

Following the review guidance provided in Chapter 15.0.2 of the SRP, the next area of review for transient and accident analysis methods discussed in this SE focuses on uncertainty analysis. The associated acceptance criteria indicate that the analysis must address all important sources of code uncertainty, including the mathematical models in the code and user modeling such as nodalization. The major sources of uncertainty must be addressed consistent with the results of the accident scenario identification process.

The CRDA LTR discusses each of the individual parameters identified as being high importance in available regulatory guidance. In general, the uncertainty associated with each parameter was dispositioned in one of the following ways:

1. The parameter is set to bounding values, therefore, no uncertainty needs to be considered. (example: [[ ]])

2. Studies were performed to establish the sensitivity of the results to the parameter across the range of uncertainty (based on available references). (example: [[ ]])

3. The uncertainty within a parameter is accommodated by conservatisms in the analysis (example: [[ ]])

The following table summarizes the parameters evaluated, how the uncertainties were addressed in the proposed CRDA analysis methodology, and the NRC staff's assessment of the acceptability of the approach used for the purpose of determining the expected impact on the limiting enthalpy rises for the CRDA analysis. Most of the parameters are identified in the CRDA TR, but the NRC staff identified some additional parameters that are expected to impact the results from the CRDA analysis. GNF-A addressed these parameters in their response to the NRC RAIs.

Parameter	GNF-A Analysis	NRC Assessment
Doppler reactivity coefficient	[[	The Doppler reactivity coefficients are expected to have a direct relationship to the severity of the power excursion, given that the Doppler reactivity is the primary mechanism by which the power excursion is arrested. The information provided by GNF-A is a reasonable basis to infer some general conclusions. The two sigma uncertainty of [[ ]] is consistent with previously

Parameter	GNF-A Analysis	NRC Assessment
	]]	approved NRC methodologies (Rev. 10). There is clearly an effect on the enthalpies, and the application of [[ ]] Doppler coefficients has been shown to capture sufficient margin to offset the observed effects (see Section 4.2.4.5 of this SE). Therefore, the NRC staff finds the conservatism in the application of the Doppler coefficients to be sufficient to accommodate the uncertainty in the Doppler reactivity feedback.
Void reactivity coefficient	[[ ]]	The range of feedback variation analyzed is somewhat arbitrary, but significantly larger than expected based on the assessment of PANACEA and TRACG for AOO and ATWS events. The impact of void reactivity feedback on the limiting enthalpies is expected to be very small due to the fact that significant heating of the moderator would be required to reach saturation conditions, so significant voiding is not expected to occur. Therefore, the NRC staff finds this analysis to be sufficient to demonstrate that the impact of the void reactivity feedback [[ ]].
Manufacturing uncertainties	[[ ]]	The NRC staff agrees that the use of a [[ ]] is sufficient to account for any impacts (expected to be small) due to manufacturing tolerances on the [[ ]]. However, the more important aspect is that the NRC fuel cladding failure thresholds were established based on test results that covered a variety of fuel rod designs and claddings. Therefore, a variety of different manufacturing specifications are already implicit in the failure thresholds. Therefore, manufacturing tolerances do not need to be explicitly addressed in the CRDA model.
Fuel Cladding Failure Thresholds	GNF-A discusses the basis for the failure thresholds. Also, uncertainties in the best estimate hydrogen pickup model used to evaluate the PCMI failure threshold	The NRC is currently in the process of finalizing the failure thresholds, as described in draft regulatory guide DG-1327. Once final, the thresholds can be used without further justification. The NRC staff finds that [[ ]] is sufficient to

Parameter	GNF-A Analysis	NRC Assessment
	are accounted for by [[ ]].	account for the two sigma uncertainty in the best estimate model used by GNF-A.
Burnup	[[ ]]	The NRC staff agrees that the conservative assumption [[ ]] is sufficient to bound any variations in fission gas release from the pellets. The NRC staff discussion of the approach used to ensure that the limiting exposure for a cycle is identified can be found in Section 4.2.5.3 of this SE.
Fission Gas Release	[[ ]]	The NRC staff discussion of the fission gas release assumptions can be found in Section 4.2.4.3 of this SE.
Control Rod Worths	[[	The NRC staff agrees that the approach used to model the reactivity insertion due to the control rod drop generally models all relevant parameters [[ ]]. However, the CRDA LTR states [[ ]]. The NRC staff asked RAI-6 to request justification that the results of this evaluation approach would bound [[ ]]. In their response (Ref. 13), GNF-A stated that [[ ]].

Parameter	GNF-A Analysis	NRC Assessment
	]]	The NRC staff noted that there is also an uncertainty associated with manufacturing tolerances for the control rod. However, the NRC staff does not expect that this uncertainty would be significant because if it were, it would adversely impact the reactivity anomaly by significantly broadening the variance in measured eigenvalues compared to predicted values. The overall CRDA analysis procedure is discussed in Section 4.2.5 of this SE.
Reactor Scram	[[ ]]	The NRC staff agrees that this is a conservative assumption, [[ ]].
Delayed Neutron Fraction	The initial submittal of the CRDA LTR did not address the uncertainty in the delayed neutron fraction. In response to RAI-7 from the NRC (Ref. 13), GNF-A provided information from an uncertainty analysis performed by [[ ]] delayed neutron fraction values from a normal distribution corresponding to a standard deviation of 12%. This standard deviation was based on a weighted combination of the delayed neutron fraction uncertainties for U-235, U-238, and Pu-239 near the end of life for a fuel assembly, which is when the weighted total uncertainty is at a maximum. The results showed small impacts on	Prior assessments of the PANACEA neutron kinetics model indicate that the [[ ]]. The delayed neutron fraction used by PANACEA is [[ ]]. Therefore, the uncertainty in delayed neutron fraction is expected to originate from two [[ ]] sources: (1) the uncertainty in the relative number densities of the isotopes contributing to fission as a result of accumulated code and cross section uncertainties during the depletion, and (2) uncertainty in the experimental values determined for the delayed neutron fraction for each contributing isotope. GNF-A used an appropriate reference from open literature to calculate a weighted uncertainty that accounts for the different fission yields of each contributing isotope. The 12% value was selected as the highest standard deviation, driven by the higher percentage of uncertainty in the delayed neutron fraction for Pu-239 which dominates at the end of a fuel assembly's life. The first uncertainty was not addressed by GNF-A, however, the expected maximum variation

Parameter	GNF-A Analysis	NRC Assessment
	the peak enthalpy and PCT.	would be small compared to the overall increase in Pu-239 relative to U-235 and U-238, and the limiting CRDA events would be driven by lattices that are still relatively early in their life. As discussed in the RAI response, GNF-A performed a statistical analysis [[ ]] to assess the impact of the uncertainty on the calculated power, enthalpy, and PCT in TRACG. The results show that the estimated 95/95 increase in enthalpy and PCT due to uncertainty in the delayed neutron fraction is on the order of [[ ]].
Rod and Assembly Power Distribution	The 3-D neutronic models in PANACEA and TRACG have an inherent uncertainty associated with local rod and assembly power distributions. This uncertainty was not addressed in the CRDA LTR, so NRC staff requested further justification in RAI-8. In response (Ref. 13), GNF-A provided some discussion stating that the power distribution uncertainties are addressed by the fact that [[ ]].	The NRC staff agrees that the effect of the power distribution uncertainties is implicitly captured in [[ ]]. However, the NRC staff also expects that there would be a more direct impact on the results of the enthalpy calculation—an increase in local power generation for the limiting rod would result in a higher enthalpy rise. The 3-D neutronic models may not exhibit a consistent bias, but the analysis of the CRDA event is intended to investigate the highly local conditions associated with the single most limiting fuel rod. Therefore, the power distribution uncertainties should be addressed. The total uncertainty may be conservatively considered to have a direct proportionate impact on the enthalpy rise—i.e., a [[ ]] increase in power would produce a [[ ]] increase in deposited enthalpy. In reality, greater power deposition in the fuel rod would cause a more rapid Doppler reactivity response, dampening the power pulse and reducing the total power deposition. As discussed in Section 6.0, the NRC staff believes that the inherent conservatisms in the proposed CRDA

Parameter	GNF-A Analysis	NRC Assessment
		analysis methodology are sufficient to offset this uncertainty.
Core Initial Conditions	Sensitivity studies were performed to identify bounding or representative values.	See Section 4.2.4.4 of this SE.

Based on the above discussion, the NRC staff finds that GNF-A has appropriately considered and accommodated all uncertainties through demonstrations that the uncertainty would have a minimal impact on the results of the CRDA analysis, or through conservative modeling approaches that bound the effects of the uncertainties.

4.5 Methodology Implementation

Section 6.0 of the CRDA LTR describes changes to GESTAR II methodology and standard TSs (STS) that will be necessary to allow licensees to use the proposed CRDA methodology. The NRC staff reviewed the proposed changes to confirm that they are consistent with the intended use of the CRDA methodology.

The changes to GESTAR II primarily consist of the addition of the proposed CRDA methodology as an option for licensees to utilize for their licensing basis associated with CRDA analyses. Several documentation requirements are incorporated into the application of the CRDA methodology as part of GESTAR II, namely:

1. [[ ]] will be included in the fuel product compliance report. [[ ]] Inclusion of this information in the fuel product compliance report ensures that this information is readily available for NRC audit. [[ ]]

2. Control rod withdrawal sequences that have been confirmed to meet the acceptance criteria in the CRDA LTR are to be captured in the plant reload document associated with each cycle. In addition, the plant's supplement reload design report will confirm that the methodology described in the CRDA LTR, as approved, was used to validate the cycle as being compliant with the plant licensing basis for the CRDA event.

The STS changes are provided only for the BWR/4 STS, which contain required actions and surveillances that are specific to the BPWS. Since this CRDA analysis methodology is intended to provide an alternative to the BPWS, the proposed changes are intended to replace the references to the BPWS with requirements applicable to the control rod withdrawal sequences developed using the methodology described in the CRDA LTR. The NRC staff confirmed that the revised requirements include references to all the relevant constraints to ensure that the CRDA analyses remain valid, including adherence to the analyzed control rod withdrawal sequence, the maximum number of fully inserted out of sequence rods, and the reactor power/pressure at which the at-power CRDA basis becomes applicable.

As a result of the above review of the proposed changes to GESTAR II and the STS, the NRC staff finds that the proposed updates will be adequate to incorporate the proposed CRDA analysis methodology in plant licensing bases by capturing the relevant details in licensing basis documentation.

4.6 Methodology Updates & Extended Applicability

The final area of review for the NRC staff pertains to the allowed updates and extended methodology applications discussed in Section 7.0 of the CRDA LTR. The intent of this section is to indicate when new models and codes can be substituted in lieu of the ones assumed to be used in the CRDA LTR, and to clarify acceptable applications of the proposed CRDA analysis methodology beyond that described in the CRDA LTR. The NRC staff considerations regarding each item are provided below.

1. (Section 7.1 of the CRDA LTR) The NRC staff agrees that, [[ ]].

2. (Section 7.2.1 of the CRDA LTR) The NRC staff agrees that the procedure described in Section 4.1 may be used to allow use of [[ ]].

3. (Sections 7.2.2 and 7.2.3 of the CRDA LTR) The failure threshold curves and hydrogen pickup model used in the CRDA LTR are both provided in NRC guidance, and are only used to determine whether the enthalpy results from TRACG indicate fuel failure or not. Consequently, if the NRC approves new curves or models that are applicable to the fuel being analyzed, the new curves or models can be used without affecting the acceptability of the analysis methodology.

4. (Section 7.2.4 of the CRDA LTR) [[ ]] are generally expected to be analyzed using [[ ]]. However, an alternative approach is to confirm [[ ]]. This is consistent with one possible application of item 1 (above).

5. (Section 7.3 of the CRDA LTR) The NRC staff agrees that the methodology described in the CRDA LTR is primarily a procedure that utilizes functional models and elements associated with approved codes for predicting fuel rod thermal mechanical, core neutronic, and thermal hydraulic performance. As such, other models and elements that serve a similar purpose may be substituted as long as they are consistent with the applicable NRC approvals. However, the NRC staff notes that the approval of the proposed CRDA analysis methodology is partly dependent on the offsetting effects of methodology conservatisms, sensitivities, and uncertainties as determined by use of the codes specified in the CRDA LTR. Therefore, use of updated models or elements,

including use of new approved codes such as LANCR or AETNA, is acceptable only if the updated or new codes do not have larger uncertainties than those discussed in the CRDA LTR and RAI responses. A limitation and condition is placed on the use of updated or new codes to ensure that the uncertainties remain within the bounds of those considered as part of the NRC review and approval of the CRDA LTR.

5.0 CONDITIONS AND LIMITATIONS

As discussed in Section 2.0 of this SE, conditions and limitations have been applied to use of the PANACEA, TRACG, and PRIME models as part of their application-specific approvals in (Refs. 10, 11, and 12. These conditions and limitations must be addressed in addition to the below conditions and limitations, which have previously been discussed in this SE and are summarized here.

1. For each application of this methodology to perform licensing basis evaluations of the CRDA event, the maximum drop speed for all control rods shall be confirmed to be bounded by the 3.11 ft/s speed assumed in this LTR or the actual maximum drop speed shall be applied.

2. When utilizing Option 2 in prescribing the control rod withdrawal order within a group, as described in Section 4.3.5.1 of this LTR, if control rods other than the highest worth rod [[

   ]].

3. When utilizing Option 3 in prescribing the control rod withdrawal order within a group, as described in Section 4.3.5.1 of this LTR, [[

   ]] control rod withdrawal sequences (i.e., all control rods within a group are withdrawn to the same intermediate position before any control rod is withdrawn past that position).

4. If updated models, elements, or codes are used with this methodology as described in Section 7.3 of this LTR, the validation results shall be similar to~~, or less than,~~ the results for the specific models, elements, and codes referenced in this LTR. Within this context, validation results [[
   ]] with consistent results, but also code/model uncertainties that are similar to, or less than, those determined for the models, elements, and codes referenced in this LTR.

6.0 CONCLUSIONS

In the CRDA LTR, GNF-A presented a new methodology to use previously approved codes—PRIME, PANACEA, and TRACG—for evaluation of the CRDA event. The new methodology is applicable for all BWR types and fuel product lines for which the approved codes are qualified. Part of the methodology includes development and application of [[ ]].

The CRDA LTR presents a description of the CRDA event and discusses the ability of the relevant technical models utilized in the PRIME, PANACEA, and TRACG analysis methodologies to capture the relevant phenomena for the CRDA event. The acceptance criteria for the CRDA event are also discussed. In the CRDA LTR, fuel rod enthalpy is the most significant output parameter considered, since this parameter drives the potential for fuel failure during a CRDA event. GNF-A specifies that this methodology is intended to ensure that no fuel failures occur, therefore, acceptance criteria such as the peak system pressure, fission product inventory release, or core coolability do not need to be addressed due to the lack of significant enthalpy production, radioisotope release, or deformation of the fuel.

No new elements, models, or codes were necessary to use the proposed methodology; therefore, the description of the methodology primarily consists of input requirements and analysis procedure guidance. This formed the bulk of the NRC staff review of the CRDA LTR and included a review for acceptability of model nodalization guidance, modeling input specifications, recommended initial conditions, control rod evaluation procedure, and acceptance criteria. The analysis procedure is specific to the CRDA event at CZP conditions; the NRC staff also reviewed information presented in the CRDA LTR to generically identify the CRDA event as non-limiting for other core conditions.

The NRC staff identified several technical issues that were not explicitly addressed as part of the proposed CRDA analysis methodology. The most significant general issues were the lack of an explicit disposition within the methodology for the delayed neutron fraction uncertainty (estimated impact: [[ ]]) and the rod/assembly power distribution uncertainty (estimated impact: [[ ]]). Additional technical issues were identified with use of relatively simple procedures to address specific considerations which did not consider the dependence of fuel failure thresholds on not only enthalpy [[ ]], but also on the hydrogen concentration and rod pressures of the surrounding fuel [[ ]]. In yet other cases, reasonable qualitative evidence was presented to indicate that the limiting cases would probably be bounded by the analysis, but insufficient quantitative evidence existed to confirm these findings. To address these issues, the NRC staff considered the significant conservatisms that were incorporated in the proposed methodology, as follows:

1. The proposed methodology is designed to confirm that no fuel failures occur. This is a more conservative approach than required to meet regulatory limits. In reality, limited numbers of fuel failures are likely to be accommodated by typical licensing bases as long as the radioisotope release is not large enough to challenge dose release limits.

2. The Doppler reactivity feedback is modeled [[ ]]. Due to the natural variation of exposures for fuel within core loading patterns, conservatism in the

Doppler reactivity feedback modeling is expected to exist for most, if not all dropped rods, especially at more limiting cycle exposures such as EOC.

3. The core is expected to be critical at the given minimum temperature for a given control rod sequence evaluation. In reality, this will be the case (or nearly the case) for a very limited number of steps. For steps prior to this point, the control rod worth will be partially or fully offset by the subcriticality of the core, and for steps beyond this point, the increasing core temperature will reduce control rod worths.

4. [[ ]], which contain several simplifications that are expected to [[ ]] (as discussed in Section 4.2.4.6 of this SE). Consequently, the calculated [[ ]] enthalpy for all fuel assemblies will be higher than the [[ ]] enthalpy during a CRDA event.

5. The PPE and peak enthalpy [[ ]]. For exposures at which the enthalpy for fuel failure decreases significantly due to loss of cladding ductility (for PCMI failure) or higher rod internal pressures (for high temperature failure), the reactivity of the fuel rod is expected to be significantly lower than the fuel rods driving the prompt power excursion. Consequently, the deposited enthalpy for the higher burnup fuel rods will not be as high as the [[ ]]. Therefore, use of a PPE combined with the maximum enthalpy [[ ]] will lead to a conservative evaluation against the acceptance criteria for higher burnups.

6. [[ ]] This will produce a more conservative value for the prompt enthalpy rise, since the [[ ]] at the time of the peak pulse (when the prompt enthalpy rise is determined) will tend to be smaller. Consequently, the PCMI failure criteria will be evaluated with conservative peak enthalpy values.

7. The FGR for fuel rods is calculated in PRIME [[ ]]. This conservatively increases the rod pressure, so evaluation of the high temperature criteria is more likely to occur when the enthalpy threshold is lower. This effect can be observed in Figure [[ ]] of the CRDA LTR, where a significant number of rods are evaluated based on the lower enthalpy threshold corresponding to a higher differential pressure than expected based on the PPEs shown in Figure [[ ]].

Three limitations and conditions were imposed to ensure that key assumptions inherent in the NRC staff understanding of the methodology are consistent with the plant/cycle configurations being analyzed, due to the sensitivity of the CRDA event to these assumptions.
In order to demonstrate the capability of the PRIME, PANACEA, and TRACG codes to analyze the CRDA event, assessments have been made against separate effects tests and integral tests. In most cases, these assessments were already performed as part of the qualification of these codes for analysis of AOO, stability, and ATWS events. One additional assessment was added, for tests performed at the SPERT III reactor to simulate rapid rod withdrawal scenarios.

The data from this assessment was valuable in that it provided confidence that the neutron kinetics models in PANACEA and TRACG could accurately predict the Doppler-only component of the reactivity feedback.

Finally, the CRDA LTR presented an evaluation of the uncertainties associated with the proposed CRDA analysis methodology. GNF-A dispositioned each uncertainty in one of three different ways: (1) by demonstration that the effect on the peak enthalpy was minimal; (2) by conservatively bounding the effect of the uncertainty; or (3) by indicating that the remaining uncertainties were bounded by the inherent conservatisms in the methodology (as discussed earlier in this section).

In addition to the description of the methodology, the CRDA LTR also included a description of the changes that would be needed to GESTAR ~~III~~ **II** and the STS in order to allow full use of the new CRDA methodology, as well as a discussion of the updates or applications that could be used with this methodology without requiring additional NRC review and approval. The NRC staff agreed with the implementation changes and the scope of the methodology applications, except for the assertion that new elements, models, or codes that had received NRC approval for other purposes could be utilized with the CRDA analysis methodology without NRC review and approval. Since some of the considerations in determining this methodology to be acceptable for use depend on the findings from the validation and uncertainty quantification for the codes, a limitation and condition was imposed to appropriately define the scope of how such applications of the CRDA analysis methodology can be implemented.

In summary, the NRC staff finds that the assessment of the PRIME, PANACEA, and TRACG codes, as described in the CRDA LTR and responses to NRC staff RAIs, adequately demonstrates that the codes are suitable to analyze the CRDA event by demonstrating acceptable predictions of the highly ranked phenomena. In addition, the NRC staff finds that the procedure described in the CRDA LTR for performance of the CRDA analyses provides appropriate guidance to appropriately identify and analyze potential limiting scenarios. Since the CRDA event is relatively insensitive to thermal hydraulic performance of the plant and appropriate guidance has been presented to address the relevant factors, the NRC approval of this CRDA analysis methodology purposes extends to all operating conditions up to and including Extended Power Uprate conditions with expanded power and flow windows. Additionally, NRC approval of the methodology described in this LTR for analysis of the CRDA event is contingent on adherence to the conditions and limitations set forth in Section 5.0.

7.0 REFERENCES

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