ENCLOSURE 3

M200034

GNF CRDA Application Methodology NEDO-33885-A, Revision 1, March 2020

Non-Proprietary Information – Public

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Global Nuclear Fuel

NEDO-33885-A
Revision 1
March 2020

Non-Proprietary Information–Public

Licensing Topical Report

GNF CRDA Application Methodology

INFORMATION NOTICE

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The design, engineering, and other information contained in this document is furnished for the purpose of supporting the NRC review and approval of the CRDA Application Methodology licensing topical report. The only undertakings of GNF-A with respect to information in this document are contained in the contracts between GNF-A and its customers or participating utilities, and nothing contained in this document shall be construed as changing that contract. The use of this information by anyone for any purpose other than that for which it is intended is not authorized; and with respect to any unauthorized use, GNF-A makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 16, 2020

Ms. Michelle P. Catts
Senior Vice President, Regulatory Affairs
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P.O. Box 780, M/C A-10
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SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL – AMERICAS, LLC (GNF) LICENSING TOPICAL REPORT NEDE-33885P, REVISION 0, "GNF CONTROL ROD DROP ACCIDENT APPLICATION METHODOLOGY" (EPID L-2018-TOP-0006)

Dear Ms. Catts:

By letter M180035 dated February 28, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18059A874), Global Nuclear Fuel – Americas, LLC (GNF-A) submitted "Licensing Topical Report (TR) NEDE-33885P, Revision 0, 'GNF Control Rod Drop Accident (CRDA) Application Methodology,'" to the U.S. Nuclear Regulatory Commission (NRC) staff for review.

By letter dated October 16, 2019 (ADAMS Accession No. ML19253C866), an NRC draft safety evaluation (SE) regarding our approval of TR NEDE-33885P, Revision 0, was provided for your review and comment. By letter M190192 dated October 29, 2019 (ADAMS Accession No. ML19302F246), you provided comments on the draft SE. The NRC staff's disposition of the GNF comments on the draft SE are discussed in the attachment of the final SE enclosed with this letter.

The NRC staff has found that TR NEDE-33885P, Revision 0 is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

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

NOTICE: The enclosure contains SUNSI information, when the enclosure is separated from this transmittal letter the letter is decontrolled

In accordance with the guidance provided on the NRC website, we request that GNF publish approved proprietary and non-proprietary versions of TR NEDE-33885P, Revision 0, within three months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. For Non Proprietary versions, GNF shall strike the proprietary information markings in this letter and make the appropriate redactions and adjustments to document security classifications to the attached SE. Also, they must contain historical review information, including NRC requests for additional information (RAIs) and your responses. The approved versions shall include a "-A" (designating approved) following the TR identification symbol.

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

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.

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

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

Sincerely,

/RA/

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 99901376

Enclosure: Final SE (Proprietary)

cc w/ encl.: See next page

SUBJECT: FINAL SAFETY EVALUATION FOR GLOBAL NUCLEAR FUEL – AMERICAS, LLC (GNF) LICENSING TOPICAL REPORT NEDE-33885P, REVISION 0, "GNF CONTROL ROD DROP ACCIDENT APPLICATION METHODOLOGY" (EPID L-2018-TOP-0006) DATED JANUARY 16, 2020

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ADAMS Accession Nos.:
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ML19345E185 (Final SE - Proprietary) ***concurrence via e-mail**

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OFFICE OF NUCLEAR REACTOR REGULATION

FINAL 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 Rod 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 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.

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 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, the NRC staff had not yet completed the process of issuing DG-1327 as a final 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 Accident Scenario Description and Acceptance Criteria

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) 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 Applicability of PANACEA Technical Models to CRDA

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. [[

]]

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 (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 time steps 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 [[extrapolated]] 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 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 <u>CRDA Analysis Procedure</u>

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:

[[

1.

2.

]]

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.
   [[
2.

3.

]]

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, the NRC staff has not yet completed the process of issuing DG-1327 as a final regulatory guide and may be subject to change, but if 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 TRACG 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 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

Parameter	GNF-A Analysis	NRC Assessment
	]]	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 are accounted for by [[ ]].	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 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 [[

Parameter	GNF-A Analysis	NRC Assessment
	• •]]	]]. 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 [[ ]]. 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 maximum. The results showed small impacts on the peak enthalpy and PCT.	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 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 3D 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 3D 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 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 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 <u>CONCLUSIONS</u>

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.

   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-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

1. Global Nuclear Fuel Report NEDO-33885/NEDE-33885P, Revision 0, "GNF CRDA Application Methodology," February 2018 (ADAMS Accession Nos. ML18059A878 (Public)/ML18059A880 (Non-Public)).

2. Global Nuclear Fuel Report NEDE-24011-P-A, Revision 29, "General Electric Standard Application for Reactor Fuel (GESTAR II) (Main and United States Supplement)," October 2019 (ADAMS Accession No.ML19276D426).

3. General Electric Hitachi Letter, "Response to NRC Request for Additional Information Letters No. 115 and No. 137 – Related to ESBWR Design Certification Application – RAI Numbers 4.6-23 Supplement 2 and 4.6-38, Respectively," April 14, 2008 (ADAMS Accession Nos. ML081090147 (Public)/ML081090148 (Non-Public)).

4. General Electric Company Report NEDO-21231, "Banked Position Withdrawal Sequence," January 1977 (ADAMS Accession No. ML090771242 (Non-Public)).

5. USNRC Document NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition,” March 2007 (ADAMS Accession No. ML070660036).

6. USNRC Draft Regulatory Guide DG-1327, "Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Drop Accidents," released for public comment July 2019 (ADAMS Accession No. ML18302A106).

7. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, as incorporated by reference in 10 CFR 50.55a.

8. USNRC Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000 (ADAMS Accession No. ML003716792).

9. USNRC Regulatory Guide 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," May 2003 (ADAMS Accession No. ML031490640).

10. General Electric Hitachi Report NEDE-32906P-A, Supplement 3-A, Revision 1, "Migration to TRACG04/PANAC11 from TRACG02/PANAC10 for TRACG AOO and ATWS Overpressure Transients," April 2010 (approved by USNRC in ADAMS Accession Nos. ML091751102 (Public)/ML091400057 (Non-Public)).

11. General Electric Company Report NEDE-32906P-A, Revision 3, "TRACG Application for Anticipated Operational Occurrences (AOO) Transient Analyses," September 2006 (ADAMS Accession No. ML062720163).

12. Global Nuclear Fuel Report NEDC-33840P-A, Revision 1, "The PRIME Model for Transient Analysis of Fuel Rod Thermal-Mechanical Performance," August 2017 (ADAMS Accession No. ML17230A008).

13. Global Nuclear Fuel Letter, "Response to Request for Additional Information for NEDE-33885P, 'GNF CRDA Application Methodology,'" April 25, 2019 (ADAMS Accession No. ML19115A084).

14. General Electric Company Report NEDO-10527, “Rod Drop Accident Analysis for Large Boiling Water Reactors," March 1972 (ADAMS Accession No. ML010870249).

15. General Electric Hitachi Report NEDC-33173P-A, Revision 5, "Applicability of GE Methods to Expanded Operating Domains," August 2019 (ADAMS Accession No. ML19228A268).

16. USNRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1," March 16, 2015 (ADAMS Accession No. ML14188C423).

17. AEC Research and Development Report IDO-17281 (TID-4500), "Reactivity Accident Test Results and Analysis for the SPERT III E-Core - A Small Oxide-Fueled, Pressurized Water Reactor," March 1969 (ADAMS Accession No. ML080320431).

Attachment: Comment Resolution

Principal Contributor: Scott Krepel, NRR/DSS/SNPB

Date: January 16, 2020

NRC Staff Response to GNF Comment Summary for Draft SE for Licensing Topical Report NEDE-33885P, Revision 0, "GNF CRDA Application Methodology"

Location	GNF Proposed Change	NRC Staff Response
Page 1 / Line 13	Revise "Road" to "Rod."	The changes are editorial or clarifying in nature, without changing the technical content of the SE. The NRC staff agrees with the proposed changes and has incorporated them as-is.
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 [[ ]]	

<table>
<tr><th>Location</th><th>GNF Proposed Change</th><th>NRC Staff Response</th></tr>
<tr><td>Page 15 / Lines 43 through 45</td><td>Recommend revised language to better reflect the situation.</td><td rowspan="2"></td></tr>
<tr><td>Page 22 / Line 36</td><td>Recommend addition of "of."</td></tr>
<tr><td>Page 25 / Line 14</td><td>Recommend revision of [[ ]]</td><td>This language is intended to reference [[ ]] assumed in the methodology, so the proposed wording would not be consistent with the intent of the phrasing. The language was clarified to use the phrasing [[ ]]</td></tr>
<tr><td>Page 26 / Line 1</td><td>Recommend revision of "maximum average enthalpy" to "maximum radially averaged enthalpy."</td><td rowspan="6">The changes are editorial or clarifying in nature, without changing the technical content of the SE. The NRC staff agrees with the proposed changes and has incorporated them as-is.</td></tr>
<tr><td>Page 27 / Line 16</td><td>We refer to our version of TRAC as "TRACG" and not "TRAC-G."</td></tr>
<tr><td>Page 31 / Various Lines</td><td>There are three instances where a "t" was added to "weighted" under Delayed Neutron Fraction.</td></tr>
<tr><td>Page 32 / Various Lines</td><td>There are two instances where we recommend "3D" rather than "3-D" under Rod and Assembly Power Distribution.</td></tr>
<tr><td>Page 35 / Line 41</td><td>Recommend deletion of ", or less than," language.</td></tr>
<tr><td>Page 38 / Line 13</td><td>Should be GESTAR II and not GESTAR III.</td></tr>
</table>

Location	GNF Proposed Change	NRC Staff Response
Page 38 / Lines 43 through 46	Most recent version of GESTAR is Revision 29.	These revisions to the referenced documents were not yet approved at the time that the safety evaluation was being completed. However, the NRC staff confirmed that the more recent revisions would not affect the findings in this safety evaluation, therefore, the references were updated.
Page 39 / Lines 47 through 49	Most recent version of NEDC-33173P-A is Revision 5.	

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Revisions

NEDO-33885, Revision 0 was submitted to the NRC for review on February 28, 2018 (ADAMS Accession Number ML 18059A874).

Revision 1 is the acceptance version of Revision 0. The NRC Safety Evaluation has been included in the front of the document and the RAIs and corresponding responses have been added as Appendix C. In addition, changes committed in the RAI responses were made in Sections 4.1.1.5 (NRC RAI #1), a new Section 4.2.2 and new Table 4-5 were added (NRC RAI #3), Section 4.3.4 (NRC RAI #4), and Section 4.3.3 (NRC RAI #6).

Acronyms and Abbreviations

Term	Definition
3D	Three-Dimensional
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
BOC	Beginning-of-Cycle
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CHAN	Channel
CPR	Critical Power Ratio
CRD	Control Rod Drive
CRDA	Control Rod Drop Accident
CRE	Control Rod Ejection
DMH	Direct Moderator Heating
DNBR	Departure from Nucleate Boiling Ratio
DSS-CD	Detect and Suppress Solution − Confirmation Density
ECP	Engineering Computer Program
EOC	End-of-Cycle
EPF	Enthalpy Peaking Factor
ESBWR	Economic Simplified Boiling Water Reactor
FGR	Fission Gas Release
FWHM	Full-Width Half-Maximum
GDC	General Design Criteria
GE	General Electric
GEH	GE-Hitachi Nuclear Energy Americas LLC
GNF	Global Nuclear Fuel – Americas, LLC
HTCF	High Temperature Cladding Failure
IRM	Intermediate Range Monitor
LCO	Limiting Condition for Operation
LDPSP	Low Dome Pressure Set Point
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LPF	Local Peaking Factor
LPSP	Low Power Set Point
LTR	Licensing Topical Report
MOC	Middle-of-Cycle

Term	Definition
MSIV	Main Steam Isolation Valve
N/A	Not Applicable
NPR	No Prompt Response
NRC	Nuclear Regulatory Commission
OEM	Original Equipment Manufacturer
OOS	Out-of-Sequence
PCMI	Pellet-Cladding Mechanical Interaction
PIRT	Phenomena Identification and Ranking Table
PLR	Part Length Rod
PPE	Peak Pellet Exposure
PPR	Pin Power Reconstruction
PRNM	Power Range Neutron Monitor
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RFRAC	Release Fraction
RG	Regulatory Guide
RIA	Reactivity-Initiated Accident
RLP	Reference Loading Pattern
RPS	Reactor Protection System
RTP	Rated Thermal Power
RWE	Rod Withdrawal Error
SE	Safety Evaluation
SNPB	Nuclear Performance and Code Review Branch
SPERT	Special Power Excursion Reactor Test
SR	Surveillance Requirement
SRP	Standard Review Plan
STS	Standard Technical Specifications
TER	Technical Evaluation Report
TOPP	Time of Peak Power
TS	Technical Specifications
WRNM	Wide Range Neutron Monitor

1.0 INTRODUCTION

In Boiling Water Reactors (BWRs), the Control Rod Drop Accident (CRDA) is a postulated design basis reactivity insertion accident. This Licensing Topical Report (LTR) provides a method for analyzing the effects of such an event.

1.1 Event Description

A postulated CRDA scenario is described below. The plant administrative and/or instrumentation controls on the rod pattern sequence are in operation. The operator begins to withdraw control rods following a predetermined withdrawal sequence. A control rod becomes decoupled from its drive and remains stuck at the full-in position. Later in the startup sequence, the rod falls at the maximum velocity and produces a high local reactivity increase in a small region of the core. The reactor attains a positive period; however, the initial power burst is limited by the Doppler reactivity feedback. The Reactor Protection System (RPS) flux- and/or period-based trip signals scram the reactor but the transient is largely terminated by the Doppler reactivity feedback before the scram has time to influence the power. Other inherent feedback mechanisms, primarily in the form of steam voids, also decrease reactivity and reduce the power and enthalpy rise in the fuel.

1.2 Regulatory Background

The original basis for CRDA analysis was developed beginning in 1972 with the previous CRDA LTR, NEDO-10527 (References 1, 2, and 3). Guidance was provided by the Nuclear Regulatory Commission (NRC) for a Pressurized Water Reactor (PWR) Control Rod Ejection (CRE) in Regulatory Guide (RG) 1.77 in 1974 (Reference 4), consistent to what was applied for BWRs. Subsequently General Electric (GE) developed the Banked Position Withdrawal Sequence (BPWS) LTR, NEDO-21231 (Reference 5), which describes a process to limit control rod worth from a postulated CRDA.

New information on fuel performance under prompt power excursion conditions has since become available, and the NRC has provided new guidance for Reactivity-Initiated Accidents (RIAs), starting with Revision 3 of the Standard Review Plan (SRP), Section 4.2, in 2007 (Reference 6). Reference 6 provided acceptance criteria thresholds for fuel cladding failure, which were based on more recent prompt power testing. Two types of criteria were provided, a High Temperature Cladding Failure (HTCF) criterion that varied with fuel rod internal pressure, and a Pellet-Cladding Mechanical Interaction (PCMI) criterion that varied with hydrogen content. These new criteria were then updated to reflect the state-of-the-art knowledge of the NRC in 2015 (Reference 7), which provides the basis for the methodology is this LTR.

1.3 Summary

This document describes and demonstrates a methodology for assuring compliance with the applicable BWR CRDA licensing acceptance criteria. The CRDA consequences are evaluated based on the fuel enthalpy response during the event. The proposed methodology evaluates these enthalpy responses in relation to the NRC-provided guidance on the fuel cladding failure thresholds (Sections 3.2 and 3.3) and other related failure mechanisms to confirm that no cladding failures occur during the event.

The CRDA calculations are performed with TRACG, which has been previously approved for a multitude of BWR analyses, including Anticipated Operational Occurrence (AOO) (References 8 and 9), Anticipated Transient Without Scram (ATWS) (References 9 and 10), Loss-of-Coolant Accident (LOCA) (Reference 11), and Detect and Suppress Solution – Confirmation Density (DSS-CD) (Reference 12) as described in Section 3.1.3. The methodology also makes use of the PANACEA Three-Dimensional (3D) core simulator, which uses the same 3D kinetics model as TRACG, and is also separately approved for steady-state BWR neutronics applications (Section 3.1.1). Throughout this document PANACEA and TRACG are used as general terms for programs that implement the models and methodology described in Section 3.1 or their NRC approved successors.

The use of these proven methods allows for modeling the feedback during the CRDA event to calculate the response for the given conditions. The comparison of the calculated enthalpies to the failure thresholds demonstrates that the cladding failure thresholds are not exceeded, and thus no cladding failures are predicted. The results of the application demonstration show that enthalpy response is limiting in BWRs at cold conditions.

This methodology applies a number of conservatisms, [[

]] and the use of conservative failure thresholds. [[

]] Section 3.7.1 provides detailed information on the process conservatisms.

[[

]]
See Section 3.5 for more information.

This methodology is applicable to all BWR types and all fuel designs for which the methods referenced in Section 3.0 are applicable. [[

]] See Section 7.0 for more information.

2.0 LICENSING REQUIREMENTS AND SCOPE OF APPLICATION

2.1 Applicable Guidance

The *General Design Criteria for Nuclear Power Plants* are stipulated in Appendix A to Part 50 of Title 10 of the Code of Federal Regulations (CFR). Each SRP section describes the acceptance criteria to meet the relevant requirements of the NRC's regulations, as they relate to the General Design Criteria (GDC). NRC approval of licensing methods used for CRDA analysis implies that the methods are capable of assessing a CRDA response as it relates to the GDC. Reactivity insertion events, specifically the CRDA in BWRs, are classified in Section 15.0 of the SRP (Reference 13) as postulated accidents.

The NRC guidelines for review of BWR CRDA events are identified in SRP Section 15.4.9 (Reference 14). Additional acceptance criteria and guidance for RIAs are provided in SRP Section 4.2 Appendix B (Reference 6). These interim criteria are further refined in NRC staff technical memorandum ML14188C423 (Reference 7).

Applicable design criteria for CRDA are Criterion 13 and Criterion 28. Criterion 28 is discussed further in Section 2.2.1; however, Criterion 13 is generically addressed by the methodology described herein because 1) rod pattern controls (administrative or instrumentation-based) are operating normally, 2) [[ ]] and 3) no additional controls are necessary for this event. Applicable guidance from the SRP related to the design criteria is listed in Section 2.2.1.

2.2 Application Methodology

This report demonstrates a methodology for determining the consequences of a CRDA using TRACG. [[

]] TRACG and PANACEA have been qualified as described in Section 3.1.

2.2.1 Regulatory Compliance

GDC 28 of 10 CFR 50, Appendix A requires reactivity control systems to be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than local yielding value, nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals so as to impair significantly the capability to cool the core. GDC 28 also requires that these postulated reactivity accidents include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold-water addition.

This application methodology using TRACG and PANACEA for BWR CRDA evaluations addresses all elements of the NRC review guidelines (References 6, 7, and 14). These criteria are collected and directly quoted in Table 2-1, which also shows where these criteria are addressed in the methodology.

Table 2-1: BWR CRDA Regulatory Guidance

Reference	Description	Section of the LTR Where Guidance is Addressed
Reference 14 II.1	GDC 13, as to 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.	See supporting review items below.
Reference 14 III.1.A	The reviewer verifies whether the applicant considers for this event a spectrum of initial conditions that cover the range of time-in-cycle and initial power levels.	4.1.1.4 / 4.3.4
Reference 14 III.1.B	The reviewer verifies the use of maximum expected individual control rod worths. The nominal control rod withdrawal pattern and abnormal patterns not precluded by an instrumentation system accepted under SRP Chapter 7 review must be considered in the development of control rod worth criteria.	4.2 / 4.3
Reference 14 III.1.C	The reviewer determines whether an acceptable and conservative function describes the control rod worth as a function of control rod position and whether the control rod position as a function of time is suitably conservative.	3.1.1 / 3.1.2.1 / 3.7.1 / 4.1.1
Reference 14 III.1.D	The reviewer determines whether conservative reactivity coefficients, notably the Doppler, are compatible with those described in SRP Section 4.3.	3.7 / 4.1.1.5
Reference 14 III.1.E	The reviewer ensures that the scram action is represented conservatively in the integral scram worth curve (SRP Section 4.3) and in the scram delay time.	3.7.1 / 4.1.1
Reference 14 III.1.F	Analytical methods are checked for previous review and approval. The applicant's methods should account conservatively for all major reactivity feedback mechanisms.	3.1

Reference	Description	Section of the LTR Where Guidance is Addressed
Reference 14 II.2	Acceptance criteria are based on GDC 28 requirements as to the effects of postulated reactivity accidents that 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. Regulatory positions and specific guidelines necessary to meet the relevant GDC 28 requirements are in SRP Section 4.2. 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.	See supporting review items below.
Reference 14 III.2	The reviewer inspects the results of the calculation of maximum reactor pressure for compliance with the acceptance criterion in subsection II of this SRP section (the reviewer may do an audit calculation when appropriate).	3.8
Reference 14 II.3	10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation to SRP Section 15.0.3. SRP Section 4.2 describes fuel rod failure mechanisms. Guidance for calculating radiological consequences is in RGs 1.183 and 1.195.	See supporting review items below.
Reference 14 III.3.A	The reviewer determines whether the transient critical power ratio is computed by an acceptable technique (reviewed either previously or de novo during this review) for analyses using at-power conditions.	4.4
Reference 14 III.3.B	The reviewer must determine the number of failed rods for the radiological evaluation. The number of fuel rod failures for each of the failure mechanisms addressed in SRP Section 4.2 (Reference 6) (see below) must be combined.	3.2 / 3.3 / 3.4 / 3.5 / 3.8 / 4.2 / 4.3
Reference 14 III.3.C	The reviewer determines the acceptability of the time-dependent steaming and activity releases from each potential release path (condenser, etc.). Each scenario should be investigated in combination and separately for the most severe release path.	3.8

Reference	Description	Section of the LTR Where Guidance is Addressed
Reference 6 Appendix B, Section B.1 As modified by Reference 7	The high cladding temperature failure criteria for zero power conditions is a peak radial average fuel enthalpy greater than 170 cal/g for fuel rods with an internal rod pressure at or below 1 MPa above system pressure, linearly decreasing to a minimum value of 100 cal/g for fuel rods with an internal rod pressure between 1 MPa and 4.5 MPa above system pressure. For intermediate (greater than 5% Rated Thermal Power (RTP)) and full power conditions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., Departure from Nucleate Boiling Ratio (DNBR) and Critical Power Ratio (CPR)).	3.3 / 4.4
Reference 6 Appendix B, Section B.2 As modified by Reference 7	The PCMI failure criteria is a change in radial average fuel enthalpy greater than the corrosion-dependent limit provided. See Reference 7 for applicable limits.	3.2
As added by Reference 7	If fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions, then fuel cladding failure is presumed.	3.5
Reference 6 Appendix B, Section C As modified by Reference 7	Fuel rod thermal-mechanical calculations, employed to demonstrate compliance with the peak radial average fuel enthalpy and peak fuel temperature criteria, must be based upon design-specific information accounting for manufacturing tolerances and modeling uncertainties using NRC approved methods including burnup-enhanced effects on pellet power distribution, fuel thermal conductivity, and fuel melting temperature.	3.1.3.6 / 3.7.1
Reference 6 Appendix B, Section C.1 As modified by Reference 7	Peak radial average fuel enthalpy must remain below 230 cal/g.	3.8

Reference	Description	Section of the LTR Where Guidance is Addressed
Reference 6 Appendix B, Section C.2 As modified by Reference 7	A limited amount of fuel melting is acceptable provided it is restricted to (1) fuel centerline region and (2) less than 10% of pellet volume. For the outer 90% of the pellet volume, peak fuel temperature must remain below incipient fuel melting conditions.	3.5 / 3.8
Reference 6 Appendix B, C.3 As modified by Reference 7	Mechanical energy generated as a result of (1) non-molten fuel-to-coolant interaction and (2) fuel rod burst must be addressed with respect to reactor pressure boundary, reactor internals, and fuel assembly structural integrity. Per Reference 7, this is no longer applicable until such time as regulatory guidance exists.	3.8
Reference 6 Appendix B, C.4 As modified by Reference 7	No loss of coolable geometry due to (1) fuel pellet and cladding fragmentation and dispersal and (2) fuel rod ballooning. Per Reference 7, this is no longer applicable until such time as regulatory guidance exists.	3.8
Reference 6 Appendix B, Section D As modified by Reference 7	The total fission-product gap fraction available for release following any RIA would include the steady state gap inventory (present prior to the event) plus any fission gas released during the event. Updated FGR correlations provided in Reference 7.	3.3.1 / 3.8

3.0 METHODOLOGY

The methodology for evaluation of a postulated BWR CRDA applies methodologies from three main technology areas: (1) core neutronics modeling, (2) hydraulic modeling, and (3) fuel thermal/mechanical modeling. Core modeling performed using the PANACEA BWR core simulator depends on lattice physics inputs generated using the TGBLA lattice physics program. Hydraulic modeling of the reactor system, core, and fuel channels is accomplished using TRACG, which relies on the nuclear modeling from TGBLA and PANACEA to calculate the transient power distribution in the core. TRACG calculates the thermal response of the fuel rods using specific inputs from the PRIME fuel thermal/mechanical program to represent the initial conditions of the fuel pellets and rods.

Implementation of the CRDA application methodology is accomplished without any technical modifications to any of the three main technology areas. The technical models in PANACEA, TRACG, and PRIME as used previously in NRC-approved applications (Reference 15) have not been modified. As will be demonstrated, these programs can all be applied within their qualified application ranges without any changes in the technical modeling capability. The CRDA application methodology describes how the inputs for these programs are specified and how the calculated outputs are applied to demonstrate compliance with the CRDA acceptance criteria.

3.1 Models

The PANACEA, TRACG, and PRIME models are applied in a complementary way to implement the CRDA application methodology described in this LTR. The elements from each model are described in further detail below as they pertain to their use in the CRDA application.

3.1.1 3D Kinetics in PANACEA and TRACG

PANACEA and TRACG use the same 3D kinetics model to solve for the nodal power distribution. The lattice physics treatment with TGBLA is integral to application of PANACEA, and is inherently a part of the approved applications (Reference 16) of both PANACEA and TRACG. The TGBLA lattice physics data needed to calculate the nuclear parameters is processed in PANACEA and passed into TRACG via the PANACEA wrapup file. The PANACEA wrapup file is the interface between PANACEA and TRACG exactly as it is in the NRC-approved AOO, ATWS, and stability applications (References 8, 9, 10, and 12). The basic model for calculating the transient neutron flux and power distribution is described in Section 9.1 of Reference 17. Inputs to the model are different between TRACG and PANACEA because of different intended applications. PANACEA is primarily intended for steady state core design and simulation, so it has many functions and associated inputs that are not needed by TRACG. For example, TRACG does not need PANACEA's capability to expose the core through the fuel cycle because TRACG can initialize the transient by accessing the exposed core conditions that PANACEA creates. TRACG is primarily intended for transient evaluations, so it has hydraulic and heat transfer models that supplant the simpler models in PANACEA. [[

]]

The nodal powers calculated by either PANACEA or TRACG are based on the solution of the 3D neutron diffusion equation using a modified one neutron energy group representation and up to six delayed neutron precursors. PANACEA solves the equation either in its steady-state or transient form depending on which module is being used. Either form accounts for the presence or absence of a control blade for each node. TRACG always solves the more general transient form which will reduce to the steady state form if all inputs are converged to a consistent set of steady conditions. The transient solution evaluates the time-dependent neutron flux and neutron precursor concentrations at every (i,j,k) node as a function of time. Control blade movements during the transient are explicitly modeled. The 3D kinetics modeling for CRDA in TRACG is no different from the modeling employed for TRACG AOO transient (References 8 and 9) or stability (Reference 12) applications previously approved by the NRC.

3.1.2 PANACEA

PANACEA is used in several different ways for design, licensing, and monitoring of the current BWR fleet in the United States. No technical changes in PANACEA are necessary to implement the CRDA methodology, as the models currently used in various applications are applicable. Some current applications include: core design to determine the Reference Loading Pattern (RLP) for reload licensing, evaluation of shutdown margin, AOO evaluations of Rod Withdrawal Error (RWE), and monitoring of thermal limits (Reference 15). Use of PANACEA for CRDA calculations has previously been accepted by the NRC as part of the certification of the Economic Simplified Boiling Water Reactor (ESBWR) (Reference 19).

3.1.2.1 Control Blade Worth

PANACEA has the proven ability to calculate static control blade worths. This capability is already applied when performing licensing calculations for cold shutdown margin demonstration (Reference 15). Positive reactivity due to control blade withdrawal during the CRDA is the single most important variable in the assessment of CRDA scenarios. [[

]] TRACG uses the transient 3D neutron kinetics model from PANACEA, thus TRACG has the same capability as PANACEA to simulate control blade movements during a CRDA transient. For the most limiting high-worth control blades that produce local prompt criticality during the CRDA, the scram occurs too late to affect the amplitude of the power pulse or the corresponding prompt enthalpy deposition. In the longer term, the scram reduces the overall core reactivity, which will reduce the transient power response calculated by PANACEA or TRACG. [[

]]

3.1.2.2 PANACEA Nodal Power and Nodal Fuel Enthalpy Responses

Nodal fuel enthalpy and nodal fuel temperature are directly related via the fuel specific heat as indicated in Section C.1.2 of Reference 17. The fuel temperature can be calculated from the fuel enthalpy or the fuel enthalpy can be calculated from the fuel temperature using Equation C.1-7 of Reference 17. [[

]]

As will be shown later, the PANACEA model can appropriately calculate the prompt enthalpy rise that occurs in the fuel prior to any perceptible changes in the moderator conditions if the core is initially cold. The analysis methodology has been qualified against the Special Power Excursion Reactor Test (SPERT) reactivity transient tests, as documented later in this report. [[

]] TRACG can be used as an alternate calculation [[

]]

3.1.2.3 Rod Power Local Peaking Factor

An estimation of the LPF during the CRDA transient is needed so that the enthalpy response for the most limiting fuel rod can be calculated. [[

]] The NRC previously reviewed and accepted the calculation of pin powers using TGBLA lattice inputs and PANACEA with PPR as indicated in their final Safety Evaluation (SE) contained in Reference 20.

3.1.2.4 Rod Fuel Enthalpy Peaking Factor

[[

]]

[[

]]

Figure 3-1: EPF as a Function of Time

3.1.3 TRACG

The detailed models used in TRACG and their verification are described in the TRACG licensing documents contained in References 8, 9, 17, and 21. None of these models have changed. The models, processes, and interfaces are unchanged from those previously reviewed by the NRC for other applications that apply the 3D kinetics model (i.e., AOO (References 8 and 9), stability (Reference 12), and ATWS (References 9 and 10)). TRACG also contains heat transfer, hot rod, and cladding perforation models developed, quantified, and reviewed for LOCA applications (Reference 11). All these models, processes, and interfaces are also applicable for CRDA applications. CRDA calculations performed for the ESBWR using TRACG have previously been accepted by the NRC (Reference 19). This document focuses on how the existing TRACG capabilities are applied in CRDA applications.

3.1.3.1 TRACG Modeling Incorporated from PANACEA

TRACG and PANACEA use the same 3D kinetics model to solve for the nodal power distribution. The basic model for calculating the transient neutron flux and power distribution is described in Section 9.1 of Reference 17. The description of the model application and differences between TRACG and PANACEA are described in Section 3.1.1.

The initial control blade pattern that will be evaluated during the TRACG transient CRDA calculation is also specified via the PANACEA wrapup file. [[

]]

Static control blade worths in TRACG are the same as they are in PANACEA because they are calculated from the same 3D kinetics model using the same initial inputs. The user interface to move control blades is different in TRACG but ultimately it sets the nodal control fractions in the same array in the 3D kinetics model that is used in PANACEA.

Nuclear parameters required for the 3D kinetics model are the same between TRACG and PANACEA. They are described in Section 9.2 of Reference 17. The distinction between TRACG and PANACEA is related to the hydraulic and fuel temperature values that are needed to evaluate the nuclear parameters. As discussed in Section 3.1.1 the PANACEA hydraulic models are simpler because they are intended mainly for steady-state analyses whereas the TRACG models are designed for transient applications. Notwithstanding, by design both models produce exactly the same initial nodal power distribution for the steady state. Section 9.4 of Reference 17 describes the implementation of the TRACG thermal hydraulics with the PANACEA 3D kinetics model both for the initial steady state and in general during the evaluation of the transient.

3.1.3.2 TRACG Nuclear-Related Modeling that is Different from PANACEA

[[

]] The TRACG model is described in Section 9.3 of Reference 17. [[

]]

TRACG has a model for calculating the amount of Direct Moderator Heating (DMH) that depends on local moderator density. Equation 9.4-14 of Reference 17 describes the basic TRACG model. The TRACG model was developed originally for AOO applications as explained in Subsection C3DX in Section 5.1 of Reference 8. [[

]]

TRACG has a detailed mechanistic model for calculating fuel rod temperatures that is applicable for both steady state and transients (Reference 11). For steady state, PANACEA estimates the nodal effective average fuel temperature using heat flux tables that relate nodal power to fuel temperature. Note that one downstream effect of implementing the PRIME models (Reference 18) was to change the PANACEA heat flux tables to be consistent with how reduced fuel pellet thermal conductivity with exposure produces higher fuel temperatures given the same nodal power. [[

]]

3.1.3.3 TRACG Fuel Rod Enthalpy

In TRACG, transient heat transfer from the fuel to the moderator is calculated for each axial cell of each fuel rod group of each simulated channel. [[

]]

Calculated fuel temperatures are obtained from a one-dimensional heat conduction equation performed at each axial cell of each simulated fuel rod group in each simulated fuel channel group. The technical details for the fuel rod heat conduction model are provided in Section 4.2.3 of Reference 17. [[

]]

3.1.3.4 TRACG Models that Affect Calculated Fuel Temperatures

The models that are used for CRDA calculations are the same models that are used for all other TRACG applications: AOO (References 8 and 9), stability (Reference 12), ATWS (References 9 and 10), and LOCA (Reference 11). The nominal models are used [[

]]

Heat transfer coefficients at the cladding surface are calculated using the best-estimate models described in Section 6.6 of Reference 17. These models have been validated for a wide range of conditions that include ranges of application expected for CRDA evaluations.

CRDA applications utilize the TRACG dynamic modeling for the pellet-cladding gap. The model as described in Section 7.5.2 of Reference 17 refers to the original GESTR-LOCA gap conductance model. The model based on PRIME is essentially unchanged and it is accurate to substitute PRIME for GESTR in this description. [[

]] More details about the PRIME interface with TRACG are provided in Section 3.1.3.6.

[[

]]

The thermal conductivity of the fuel is calculated within TRACG using the model that has been updated to be equivalent to the thermal conductivity model used in PRIME (Reference 18). Details are provided in Section C.1.4.1 of Reference 17. The PRIME thermal conductivity model is now the default model in TRACG and is used for all applications. The PRIME fuel thermal conductivity model accounts for effects due to exposure and gadolinia content that were not considered in the previous model.

3.1.3.5 TRACG Calculated Fuel Rod Internal Gas Pressure

All TRACG models used to calculate fuel rod internal gas pressure are unchanged from their previous applications for AOO (References 8 and 9), stability (Reference 12), ATWS (References 9 and 10), and LOCA (Reference 11). Fuel rod internal gas pressure is calculated by TRACG for every rod group of every simulated channel using Equation 7.5-31 of Reference 17. Other details for how the TRACG model accounts for changes in the gas volume and temperatures are provided in Sections 7.5.3.1 and 7.5.3.2 of Reference 17. [[

]]

The PRIME fuel files provide the key inputs as is the case in all other TRACG applications that use the dynamic gap model. [[

]]

For CRDA transients in TRACG, the PRIME fuel files are applied [[

]]

Limitation 1.3 in the NRC SE contained in Reference 11 "permits modeling of competitor or co-resident fuel to the extent that TRACG LOCA can accommodate the design features of such fuel, but requires that operating constraints on such fuel remain supported by, or more conservative than, the analytic methods furnished by the vendor(s) of the fuel." [[

]]

3.1.3.6 TRACG Cladding Perforation Calculation

The local pressure difference between the calculated fuel rod internal gas pressure and local coolant pressure is used to calculate the cladding hoop stress. The details are provided in Section 7.5.3.1. of Reference 17. [[

]] (See the response to Nuclear Performance and Code Review Branch (SNPB) Request for Additional Information (RAI)-33 in NEDE-33005P-A Revision 1 (Reference 11).)

The most important features of the TRACG cladding perforation model are: [[

]] The details of the model are provided in Section 7.5.3.1. of Reference 17.

Increases in fuel rod internal gas pressures and cladding temperature depend on the power level [[

]] See Section 3.4 for a discussion of how the TRACG cladding perforation model complements the NRC CRDA acceptance criteria.

3.1.4 PRIME

The NRC-approved PRIME steady-state capability (References 22, 23, and 24) is used to produce the fuel file inputs that are needed in the TRACG CRDA calculations. [[ ]] The PRIME references cited here contain the entire contents of their corresponding NRC final SE and Technical Evaluation Report (TER).

3.1.4.1 Fission Gas

Fission gas production in the fuel pellet and release of fission gas from the fuel pellet is modeled by PRIME [[ ]] This information is passed to TRACG via the PRIME fuel files. [[

]]

In the NRC assessment (Reference 18) of PRIME, the NRC contractor Pacific Northwest National Laboratory concluded that the FGR model in PRIME is acceptable for steady-state and transient cases up to a [[ ]]. This exposure limitation is adequate for CRDA evaluations because the bundles that produce the largest enthalpy responses have lower exposures.

3.1.4.2 Gap Conductivity

Gap conductivity depends on the composition of the gases in the free volume inside the fuel rod tube and outside of the fuel pellet. Gap conductivity affects the calculated fuel rod temperatures, which in turn influence the release of fission gas from the fuel pellet. These effects are all modeled in PRIME to provide a best estimate of the amount of fission gas that has been released from the fuel into the gap.

Fission gas values obtained from PRIME are passed to TRACG via the fuel files. [[

]]

3.1.4.3 Fuel Rod Thermal-Mechanical Model

The PRIME fuel rod thermal-mechanical model together with all other PRIME models are used [[ ]] as defined in Section 3.3.1 of Reference 24. [[

]]

The PRIME fuel files [[

]] This interface is the way that PRIME thermal-mechanical limits are communicated and enforced in downstream application programs like TRACG.

For CRDA applications the PRIME fuel files are applied [[

]] These uncertainties are summarized in Section 8.6 of the SE on PRIME in Reference 24 where the reviewer concluded for each element that the treatment of uncertainty was acceptable.

3.1.5 Model Qualification Relevant to CRDA Applications

Coupling of the PANACEA 3D neutronic model with hydraulic, fuel rod, and other models in TRACG has previously been utilized for AOO (References 8 and 9), stability (Reference 12), and ATWS (Reference 9 and 10) calculations. These models are qualified for BWR applications by comparison to BWR plant data in Chapter 7 of Reference 21. This qualification against plant data when combined with the separate effects, component, and system effects testing in earlier chapters of Reference 21 provides an excellent basis for the application of TRACG to all BWR plant transients, including CRDA applications.

3.1.5.1 Description of SPERT III Core

Comparisons with experimental rod drop transients performed at the SPERT III facility in 1965 provide additional qualification of PANACEA and TRACG for rapid reactivity insertions like those that characterize the limiting BWR CRDA events. The experimental tests are very fast and of short duration so hydraulic modeling is not important except to establish the initial thermal conditions for the fuel. For this reason, the tests can be simulated well using only PANACEA. For comparison purposes, the test (Test 43) with the highest reactivity insertion of $1.21 was also simulated with TRACG.

Details of the SPERT III facility and the reactivity insertion tests that were performed there are available in Reference 26. For convenience, some of the most relevant features are briefly summarized here with additional information provided in Section 3.8 of Reference 21. The SPERT III facility was designed as a small model PWR. The fuel was sintered UO_2 (10.5 g/cm^3) enriched to 4.8%. The fuel rods were 0.466 inches (11.8 mm) in diameter and located in 4x4 and 5x5 BWR type fuel bundles. The fuel bundles were located on three-inch (7.62 cm) centers. The control rods were of two types: a single cruciform transient rod and eight box shim rods. The control rods were made of borated steel. The transient rod was located in the center of the core and was inserted into the core from the bottom. The shim rods were in the second and third fuel ring and were inserted from the top of the core. Each shim rod had a fuel leader which consisted of a sixteen-rod fuel bundle. The placement of the different components of the SPERT III core is shown in Figure 3.8-1 of Reference 21. The cruciform control rod was in the center of the core with the four sixteen-rod bundles surrounding it. Each core quadrant contained two coupled shim rods. Additional design characteristics of the SPERT III core can be found in Table 3.8-1 of Reference 21.

The cold reactivity insertion transients were run from a 294 K (70°F) condition with no coolant flow. The reactivity insertion at cold startup conditions ranged from $0.77 to $1.21. For comparison of the SPERT III transients with the methodology described above, the nuclear cross sections were generated using the TGBLA06 nuclear lattice design program with each unique 4x4 or 5x5 fuel lattice simulated separately. As in the typical BWR model, each fuel assembly was modeled as a stack of twenty-four equal-length axial nodes. Because of the shorter SPERT III active core height of 97.28 cm (38.3 inches) compared to ~366 to 381 cm (144 to 150 inches) for a BWR core, the axial mesh height used for SPERT III simulation was 4.05 cm (~1.59 inches) compared to a typical mesh height of 15.24 cm (6 inches) for a BWR core.

3.1.5.2 PANACEA Transient Experiment Comparisons

Table 3-1 summarizes for several of the SPERT III core cold startup transients the key comparisons between the PANACEA calculated and experimental results and experimental uncertainties from Reference 26. Specifically, Tests 22, 13, 17, 51, 19, 21, 41, and 43 were considered. These are arranged in Table 3-1 in order of increasing accident rod worth.

The initial critical position of the shim-transient rods for each case was different. [[

]] The cold critical experimental configuration at 294 K (70°F) and a pressure of 1.01352E5 Pa (14.7 psia) is documented in Reference 26 to occur with the shim bank position 37.08 cm (14.6 inches) above the bottom of the core, and the transient rod all the way in (i.e., at notch position 00).

[[

]] The maximum reactivities obtained for all cases along with the initial target eigenvalue are well within the experimental uncertainty.

The peak power for all of cases agrees well with 1.5 standard deviations of the experimental values. [[

]]

The power responses versus time for all eight PANACEA cases were compared to the corresponding measured responses. In all cases the agreement was very good for the time ranges over which the experimental data was recorded. Of all the tests, the power response from Test 43 with the highest reactivity insertion of $1.21 is expected to best reflect a power response from a BWR CRDA event that would result in a fuel enthalpy response closest to the acceptance criteria. For this reason, only the plots for Test 43 are shown here in Figure 3-2 and Figure 3-3.

Integration of the power response with time for the calculated and measured responses allows one to quantify the total energy deposited in the fuel. These total energies do not quantify the effects of specific bundle and fuel rod local power peaking but they do provide a quantity that is proportional to the core-averaged energy deposited in the fuel which is thus proportional to deposited fuel enthalpy.

3.1.5.3 TRACG Transient Experiment Comparisons

Only Test 43 of the SPERT III cold reactivity insertion transients was analyzed with TRACG because it was the test with the highest control blade worth of $1.21 ± 0.05. TRACG simulation of Test 43 is documented in Section 3.8 of Reference 21. For convenience, the information has been summarized here. A PANACEA wrapup file was generated for the corresponding Test 43 configuration and initial conditions. The PANACEA wrapup file provides the nuclear parameters as functions of water density, fuel temperature, control state and exposure as described in Section 9.2 of Reference 17.

[[

]]

The TRACG simulation was performed up to the time where experimental data were recorded at the nominal control rod worth of $1.21. [[

]]

The calculated powers are compared with the experimental results in Figure 3-4. [[

]]

The peak power (total core power) is very sensitive to the control rod worth as shown in Figure 3-4. [[

]] The energy deposition or integrated power during the experiment is obtained by integrating the power curves with time. For comparison purposes, the calculated energies have been shifted in time by the same amount as used to line up the calculated peak powers. The integrated powers are compared in Figure 3-5. [[

]] The comparison shows that the TRACG calculation for the nominal rod worth of $1.21 predicts the total energy within the estimated uncertainty. [[

]]

3.1.5.4 SPERT III Benchmark Conclusions

The methodologies in PANACEA and TRACG used in the analysis of reactivity insertion accidents involve the use of the time-dependent 3D diffusion equations using the modified one-and-a-half group equations from the steady-state methods and six delayed neutron groups. [[

]] Both methodologies have been qualified by comparison to the SPERT III experimental rod drop accident measurements. They show excellent agreement with these experimental data. Based on the demonstrated characteristics and the qualification of the two methodologies, it is concluded that methods in PANACEA and TRACG are valid for the purposes of the analyzing rapid reactivity insertion events.

Table 3-1: Summary Comparison of Experimental and PANACEA Calculated Results

Description	Maximum Reactivity ($)	Peak Power (MWth)	Time of Peak (s)	Energy@ TOPP (MJ)
Test 22 (experimental uncertainty)	0.77 (0.03)	2.1 (0.3)	13.7 (0.2)	6.9 (1.2)
[[				]]
Test 13 (experimental uncertainty)	0.93 (0.04)	5.6 (0.8)	3.2 (0.06)	5.1 (0.9)
[[				]]
Test 17 (experimental uncertainty)	0.99 (0.04)	8.6 (1.3)	1.2 (0.05)	3.3 (0.6)
[[				]]
Test 51 (experimental uncertainty)	1.00 (0.04)	11 (2)	1.08 (0.02)	2.4 (0.4)
[[				]]
Test 19 (experimental uncertainty)	1.03 (0.04)	18 (3)	0.63 0.01)	1.9 (0.3)
[[				]]
Test 21 (experimental uncertainty)	1.09 (0.04)	56 (8)	0.38 (0.01)	2.8 (0.5)
[[				]]
Test 41 (experimental uncertainty)	1.13 (0.05)	110 (17)	0.31 (0.02)	3.8 (0.6)
[[				]]
Test 43 (experimental uncertainty)	1.21 (0.05)	280 (42)	0.21 (0.006)	6.0 (1.0)
[[				]]

[[

]]

Figure 3-2: PANACEA Power Response Comparison to SPERT 43 Data ("EXPT" is experimental data, "ALT" is time-shifted PANACEA calculation)

[[

]]

Figure 3-3: PANACEA Integrated Energy Comparison to SPERT 43 Data ("EXPT" is experimental data, "ALT" is time-shifted PANACEA calculation)

[[

]]

Figure 3-4: TRACG Power Response Comparison to SPERT 43 Data

[[

]]

Figure 3-5: TRACG Integrated Energy Comparison to SPERT 43 Data

3.2 PCMI Enthalpy Criteria

The PCMI fuel cladding failure mechanisms for a postulated CRDA event are described in References 6 and 7. Because cladding failure may occur almost instantaneously during a prompt power rise, the PCMI cladding failure threshold is based on the prompt fuel enthalpy increase. The prompt fuel enthalpy increase is defined as the radial average fuel enthalpy increase at the time corresponding to one half-height pulse width after the peak of the prompt power pulse, referred to as one Full-Width Half-Maximum (FWHM) after the peak. Hereafter, the term “delta enthalpy” is substituted for the prompt fuel enthalpy increase for brevity.

The interim RIA acceptance criteria and guidance (SRP 4.2 Appendix B (Reference 6)) provide a definition for the PCMI cladding failure threshold, which is modified based on additional information in Reference 7. The PCMI threshold is expressed in terms of the delta enthalpy (Δcal/g) and is dependent on the excess fuel rod cladding hydrogen content. The numerical values for the PCMI failure threshold from Reference 7 are listed in Table 3-2 and the PCMI failure threshold is illustrated in Figure 3-6.

3.2.1 Hydrogen Model

The PCMI fuel cladding failure threshold is presented in terms of delta enthalpy versus fuel rod cladding hydrogen content. However, it is desirable to convert it to a form based on fuel burnup (i.e., exposure) to be more easily applied in CRDA analyses. This conversion is achieved by utilizing an exposure-dependent hydrogen concentration model.

To demonstrate the conversion of hydrogen concentration to exposure, this report applies the NRC-provided best-estimate hydrogen uptake model for modern cladding alloys (Equation 3.2-1) from Reference 27. The modern hydrogen model is chosen as it is based on fuels more representative of current fuel than the legacy model. [[ ]], a 1.40 multiplier is applied to the modern hydrogen uptake model, per Reference 27, which is approximately equal to two sigma at lower burnup.

$$C_H = 1.4 * \{22.8 + \exp[0.117 * (PPE - 20)]\} \qquad (3.2\text{-}1)$$

where:

C_H = Total hydrogen concentration (wppm)

PPE = Peak Pellet Exposure (GWd/MTU)

The term peak pellet exposure is the standard terminology for the value of the local axial burnup calculated by PANACEA. PPE is approximated as the *peak nodal pin exposure* (i.e., the peak fuel pin exposure in a calculational node). The application of this hydrogen model to the PCMI cladding failure threshold is demonstrated in Section 3.2.2.

3.2.2 PCMI Threshold Conversion

As exposure and enthalpy are readily available from the TRACG calculations, the threshold criteria from Section 3.2 are more useful as a function of exposure, rather than as a function of hydrogen content. Therefore, the hydrogen model from Section 3.2.1 is applied to the threshold criteria to convert them to a function of exposure. Equation 3.2-1 is used iteratively to determine the exposure that produces the target hydrogen concentration. Thus, the PCMI failure threshold as a function of hydrogen content is recast in terms of delta enthalpy and PPE. This then allows a direct comparison to the outcome of a CRDA with a known delta enthalpy and PPE.

The numerical values for the PCMI criteria developed with the hydrogen concentration model from Section 3.2.1 are listed in Table 3-2. The PCMI failure threshold is illustrated in Figure 3-6 and the corresponding PCMI criteria are illustrated in Figure 3-7.

Table 3-2: PCMI Failure Threshold and Enthalpy Criteria

Hydrogen Concentration (wppm)	Exposure (GWD/MTU)	Delta Enthalpy (Δcal/g)
0.0	0.0	150.0
75.0	49.3	150.0
150.0	57.9	75.0
300.0	64.9	50.0

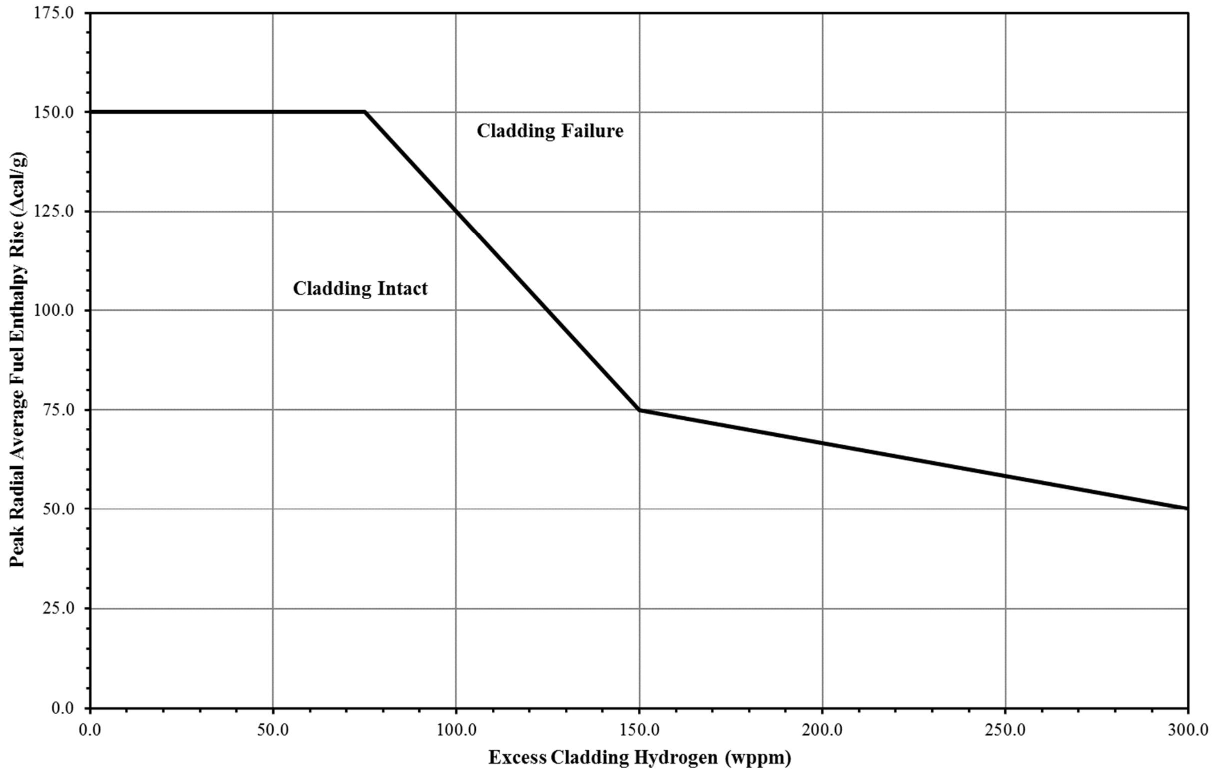

Figure 3-6: PCMI Failure Threshold

[[

]]

Figure 3-7: PCMI Enthalpy Criteria vs. Exposure

3.3 High Temperature Cladding Failure Enthalpy Criteria

The HTCF mechanisms for CRDA events are described in References 6 and 7. These events are associated with a slower increase in temperature than occurs in PCMI failure events and are based on an absolute peak radial average fuel enthalpy, not a prompt enthalpy increase (i.e., not the delta enthalpy). The interim RIA acceptance criteria and guidance (SRP 4.2 Appendix B (Reference 6)) provide a definition for the failure thresholds, which is modified based on additional information in Reference 7. The final criteria that emerge from Section 3.2.1.3 of Reference 7 provide a composite failure threshold, depicted in Figure 3-8, which is represented by the following Equation 3.3-1:

Cladding differential pressure < 1.0 MPa, (3.3-1)

Peak radial average fuel enthalpy = 170 cal/g

Cladding differential pressure > 1.0 MPa, < 4.5 MPa,

Peak radial average fuel enthalpy = 170 – ((ΔP – 1.0)*20) cal/g

Cladding differential pressure > 4.5 MPa,

Peak radial average fuel enthalpy = 100 cal/g where:

ΔP = Cladding differential pressure (MPa)

3.3.1 Fission Gas Release Model Application

The total fission gas in the gap and fuel rod plenum must be determined to calculate the cladding differential pressure. This total includes the FGR from steady-state operation, as well as the transient FGR that occurs during the CRDA event. The steady-state calculation of released and retained fission gas is performed using the approved PRIME methodology, as described in Section 3.1.4.

The only missing component is then the additional transient FGR that occurs during a postulated CRDA. This methodology applies the NRC-provided transient FGR model (Reference 7) (Equations 3.3-2A and 3.3-2B) to obtain the percent transient FGR during the CRDA event. This is then applied to the retained fission gas fraction from PRIME to obtain the combined steady-state and transient FGR in the fuel rod.

$$\text{PPE} < 50 \text{ GWd/MTU: Transient FGR (\%)} = [(0.26 * \Delta H) - 13] \quad (3.3\text{-}2A)$$

$$\text{PPE} > 50 \text{ GWd/MTU: Transient FGR (\%)} = [(0.26 * \Delta H) - 5] \quad (3.3\text{-}2B)$$

where:

FGR = Fission gas release, must be greater than or equal to zero (%)

ΔH = Radial average fuel enthalpy increase throughout the event (Δcal/g)

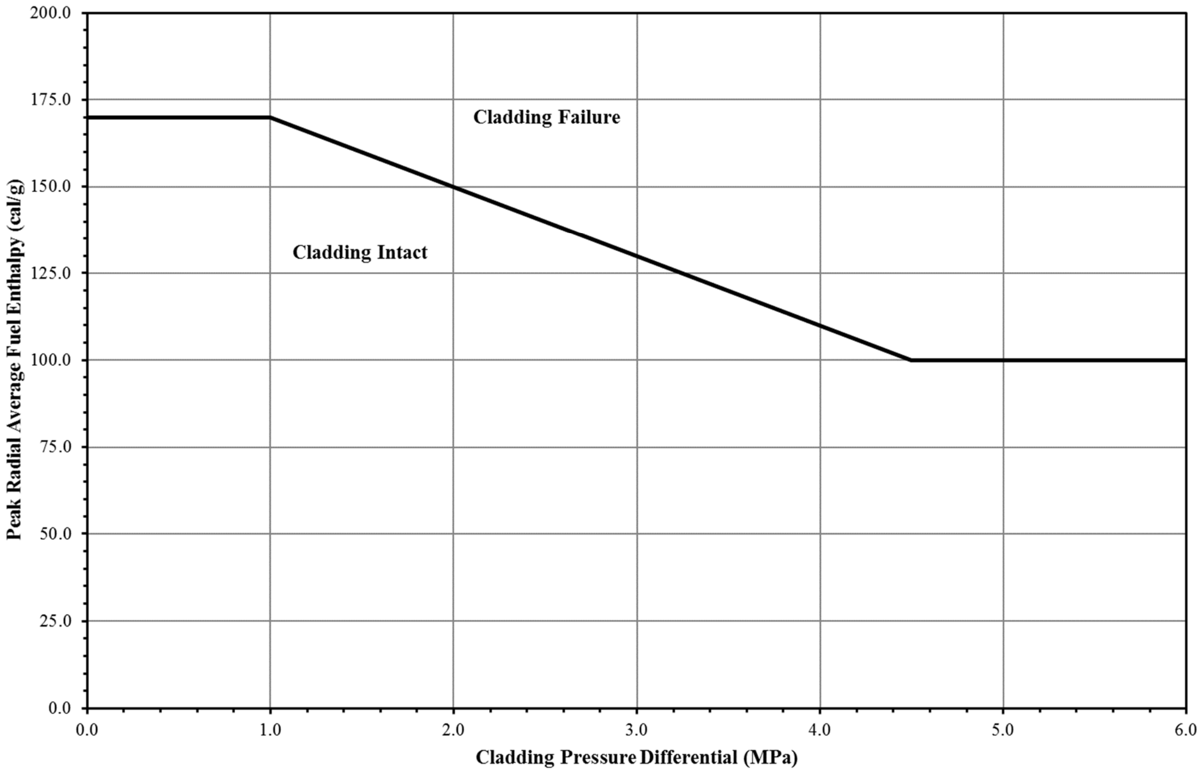

Figure 3-8: HTCF Threshold

3.4 Cladding Perforation Caused by Temperature-Induced Strain

The NRC CRDA acceptance criteria presume a relationship based on test data that relates cladding failure to (1) cladding hydrogen content, (2) fuel pellet radially average enthalpy, and (3) differential pressure between fuel rod internal gas pressure and the coolant pressure. Items 1 and 2 are supported by cladding failure data obtained from singular power pulses in test reactors. These singular power pulses were designed to produce fuel responses that are bounding of a fast reactivity insertion event like a BWR CRDA or a PWR CRE. Item 3 is an acknowledgment that differential pressure on the cladding that increases with fission gas accumulations at higher exposures will be exacerbated during a CRDA power pulse.

[[

]] The TRACG perforation model described in Section 3.1.3.6 complements the NRC CRDA acceptance criteria by evaluating the cladding hoop stress relative to the rupture hoop stress. [[

]] the CRDA mitigation strategy described in Section 4.3 will be used to also prevent cladding perforations that are predicted by the TRACG cladding perforation model.

3.5 Fuel Melting

Per Reference 7, the change to the core cooling guidance resulted in the addition of a new failure threshold: if a fuel pellet experiences any incipient melting, it is now considered a failed rod for CRDA calculations. For CRDA calculations the fuel centerline temperature can be checked against the appropriate exposure and material-dependent fuel melting temperature to determine if any melting has occurred. Alternatively, it can be demonstrated that fuel temperatures do not rise above a conservatively bounding temperature for all expected conditions in the core. If it is demonstrated that no melting occurs in the fuel pellets, then no rods are considered to have failed due to fuel melt.

3.6 [[ ]]

[[

]]

3.7 Uncertainty Evaluation

A description of the postulated BWR CRDA is described in Section 1.1 begins with an assumed complete mechanical disconnection of a control blade drive from its cruciform control blade at or near the coupling in such a way that the drive can move independently of the control blade. The probability of this initial mechanical failure is low because the design of the drive and its coupling uses high quality materials and it receives stringent quality control testing procedures appropriate to other equipment typically listed in the critical component list for a plant. Additionally, tests conducted under both simulated reactor conditions and conditions more extreme than those expected in reactor service have shown that the drive (or coupling) retains its integrity even after thousands of scram cycles. Tests also show that the drive and coupling do not fail when subjected to forces twenty times greater than that which can be achieved in a reactor (Reference 1).

For purposes of completely evaluating the consequences of the failed control drive coupling, the mechanical failure is assumed to be possible for any control blade at any time during the fuel cycle. Also for purposes of conservative evaluation, it is assumed that the decoupled control blade has stuck in the fully inserted location because this will allow for the maximum drop of the control blade when the drive is withdrawn and thus maximize the reactivity insertion. The condition of a decoupled control blade becoming stuck in its fully inserted position is highly unlikely because each blade is equipped with rollers or pads that make contact with the nearly flat channel walls. Because a control blade weighs approximately 84.4 kg (i.e., 186 pounds), even if it separates from its drive, gravity forces would tend to make the blade follow its drive movement as if it were connected (Reference 1).

At some later time after the drive has been withdrawn, the previously decoupled and stuck control blade is assumed to fall or drop at its terminal velocity from its full-in position to the position of its decoupled drive mechanism. For purposes of conservative evaluation, the drop is assumed to occur at the time in cycle and reactor conditions where the reactivity insertion would be maximized.

The probability of blade separation from its drive is already extremely low and this event does not, of itself, immediately result in a CRDA, making the probability of an impactful CRDA even more unlikely. The control rod separating from the Control Rod Drive (CRD) mechanism is not of immediate reactor safety consequence as would be a LOCA event, where the line break initiates the LOCA. In most cases, if such a blade separation were to occur, it is expected that the blade would not be stuck, but rather follow its drive movement. The separation would be detected at the next fully withdrawn stroke where the ability to withdraw to the over-travel position would signal separation, because the blade bottoms on a seat and prevents withdrawal to the over-travel position if connected. Thus, this drive could be inserted and declared inoperable in accordance with the plant Technical Specifications (TS) until the next outage where it could be repaired. However, for the analysis, it is presumed that the separated blade is somehow stuck at the fully inserted position. This assumption sets up a condition whereby, if the drive were withdrawn, the stuck blade could later fall to its drive position and cause a reactivity insertion accident defined as the CRDA. The assumption of a blade being stuck at the full-in position, while also dropping during the start-up sequence, at the maximum drop velocity, and with a worst-case blade pattern, results in an analysis which postulates an extraordinarily unlikely event. Despite the improbability of the event, this methodology addresses key uncertainties for the CRDA evaluations as described in the following section.

3.7.1 [[ ]] Addressing Uncertainties

Specific key uncertainties that are known to affect CRDA evaluations were called out in the regulatory guidance summarized in Table 2-1. Elements from Table 2-1 related to uncertainty have been listed below in Table 3-3 [[ ]]

Table 3-3: [[ ]] Used to Address Uncertainties

Uncertainty	Process Treatment	Requirement
Reactivity Coefficients (Doppler)	[[ ]] The modeling sensitivity in TRACG is provided in Section 3.7.2.	Reference 14 III.1.D
Reactivity Coefficients (Void)	Sensitivity to void reactivity is examined in Section 3.7.2, [[ ]]	Reference 14 III.1.D
Manufacturing Uncertainties	[[ ]]	Reference 6 Appendix B, C As modified by Reference 7

Uncertainty	Process Treatment	Requirement
Fuel Cladding Failure Thresholds	The fuel cladding failure criteria is a lower bound of the applicable failure data from different fuel rod designs and claddings [[ ]] Additionally, the hydrogen model (Section 3.2.1) accounts for uncertainties in the hydrogen data, [[ ]]	General
Modeling Uncertainties: Pellet Burnup Related Effects	As discussed in Section 3.1.3.5, [[ ]]	Reference 6, Appendix B, C As modified by Reference 7
Fission Gas Release	[[ ]] See Sections 3.1.4.1, 3.3.1, and 4.1.1.2 for additional information.	Reference 6 Appendix B, D As modified by Reference 7

Uncertainty	Process Treatment	Requirement
Control Rod Worths	[[ ]] Variation on worth with exposure and cycle are addressed through calculational processes discussed in Section 4.3.	Reference 14 III.1.C
Reactor Scram	[[ ]]	Reference 14 III.1.E

3.7.2 Doppler and Void Reactivity Feedback Evaluation

There are two inherent feedback mechanisms that affect the local reactivity during a postulated CRDA event. The prompt power pulse is limited by the Doppler reactivity feedback while formation of steam voids in the moderator limits the long-term power, in the absence of a scram from neutron flux or period trip signals. [[

]]

Table 3-4: [[ ]]

[[			
			]]

Table 3-5: [[ ]]

[[		
		]]

[[

]]

Figure 3-9: [[ ]]

[[

]]

Figure 3-10: [[ ]]

3.8 Post Failure Criteria Treatment

In addressing GDC 28 requirements, the methodology described herein does not calculate a number of failed fuel rods, but instead demonstrates that no failures occur for a given withdrawal sequence. Per Item III.3.B of Reference 14, the number of failed fuel rods is determined by assessing the various failure mechanisms including against the thresholds addressed in References 6 and 7. If it is determined, as in the case of this methodology, that the number of failed rods is zero, then the post-failure criteria of References 6 and 7 are no longer applicable. The post-failure criteria that are no longer applicable includes mechanical energy generated from non-molten fuel-to-coolant interaction, fuel melting considerations, fuel rod burst impacts on pressure boundary and reactor internals, and loss of coolable geometry. Additionally, without fuel cladding failures no radiological dose evaluation is required due to a postulated CRDA (References 6 and 7) and there is no need to calculate the fission product inventory. Therefore, GDC 28 is addressed by precluding cladding failures for a postulated CRDA.

4.0 TECHNICAL EVALUATION PROCESS DESCRIPTION

This section describes the application methodology using the methods and models described in the previous section. The calculations of enthalpy in TRACG provide the licensing basis for meeting the cladding failure thresholds (Sections 3.2, 3.3, 3.4, and 3.5) and confirming that no cladding failures will occur. The inputs, and assumptions, and a description of the evaluation and of how the outputs are used are provided in Section 4.1.

[[

]]

4.1 TRACG Enthalpy Evaluation

Enthalpy calculations with TRACG form the basis for demonstrating that no cladding failures occur during a postulated CRDA. The calculations are performed consistent with the models described in Section 3.1 and with the inputs and assumptions described in the following sections.

4.1.1 Inputs and Assumptions

The TRACG analyses require a number of key inputs and assumptions, which simplify the analysis. Some of the generic model assumptions and inputs are described in Section 3.1; however, the details of these and other key inputs [[ ]] are described in more detail in the following subsections. Along with these, there are a number of assumptions which are specific to the enthalpy analysis, which are not previously described.

The first of these assumptions is that the accident blade is assumed to fall from full-in to the CRD position at a rate of 3.11 ft/s. This is the maximum rate determined during velocity limiter testing from Reference 1. The accident blade is assumed to achieve this velocity instantaneously rather than accelerating from a resting position. This assumption introduces the greatest amount of positive reactivity in the shortest amount of time, resulting in a conservative calculated response from the fuel.

A second key assumption is [[

]] Additional information on key inputs is provided in the following subsections.

4.1.1.1 Basedeck and Nodalization

The general TRACG model is described in Section 3.1.3. [[

]]

The channel geometry is generated based on fuel dimensions, spacer grids, water rod locations and dimensions, and loss coefficient specifications, consistent with References 9 and 12. [[

]]

[[

]]

Figure 4-1: Example TRACG Channel Grouping

[[

]]

Figure 4-2: Example Delta Enthalpy Around Accident Blade (Δcal/g)

4.1.1.2 Transient Fission Gas Release

[[

]]

This application of the FGR model is for calculation of fuel rod internal pressure to determine the ratio to the HTCF threshold for a given CRDA analysis (Section 3.3). As the intention of the methodology is to demonstrate no fuel cladding failures due to a CRDA (Section 3.8), no considerations of dose release are made, and the process of calculating FGR above for dose considerations is not part of this methodology.

4.1.1.3 Initial Coolant Temperature

It has been observed that the consequences of a CRDA are worse at lower reactor coolant temperatures. [[

]] The results in Table 4-1 are expressed as ratios to the PCMI and HTCF enthalpy criteria from Sections 3.2 and 3.3. The ratios to the enthalpy criteria are used to determine trends in the sensitivities of this section and Section 4.1.1.4, as the failure criteria in Sections 3.4 and 3.5 are pass/fail criteria which do not provide trending information.

[[

]]

Table 4-1: Coolant Temperature Evaluation Results

[[			
			]]

Note:

1. [[ ]]

4.1.1.4 Power and Flow Evaluation

CRDA events of concern in a BWR are those postulated to occur at startup conditions, either cold zero-power or hot zero-power; however, these conditions could potentially incorporate a range of core flows, and, based on the definitions of the point at which CRDA is no longer a concern, can also include a range of very low powers. [[

]]

Table 4-2: Reactor Power and Core Flow Evaluation Inputs

Parameter	Value
[[	
	]]

Note:

1. [[ ]]

[[

]]

Table 4-3: [[ ]]

[[				
				]]

Table 4-4: [[ ]]

[[				
				]]

4.1.1.5 [[ ]]

[[

]]

[[

]]

[[

]]

Figure 4-3: [[ ]]

[[

]]

Figure 4-4: [[ ]]

4.1.2 Enthalpy Calculation

PANACEA is run to create the appropriate neutronics input to TRACG for the CRDA scenario being evaluated. [[

]]

The CRDA evaluation is then performed in TRACG using the basedeck and nodalization described in Section 4.1.1.1 and the inputs and assumptions described in Section 4.1.1. [[

]]

These peak fuel rod enthalpies are then compared to the failure criteria (Sections 3.2 and 3.3) using the exposure distribution from PANACEA and the differential rod internal pressure as calculated by TRACG (Section 3.1.3.5). Additionally, the fuel temperatures calculated for the hot rod group in each channel can be evaluated to confirm there has been no fuel pellet melting.

4.2 [[ ]]

[[

]]

[[

]]

Figure 4-5: [[ ]]

4.2.1 Out-of-Sequence Control Rods

This section presents the detailed process [[ ]], to justify a certain number of allowed out-of-sequence control rods in a given withdrawal sequence.

4.2.1.1 Background

During reactor startup and power ascension, there are various reasons why a plant operator may have to deviate from the analyzed startup sequence. For example, a CRD may become inoperable for various mechanical or electrical reasons. In such an instance, the usual operator action is to fully insert and disarm such an inoperable control rod. Plant operation may continue provided that the Limiting Condition for Operation (LCO) in the plant TS are still met.

There are also reasons why a plant operator may wish to deliberately deviate from an analyzed startup sequence by leaving a control rod fully inserted. One example is when a control rod adjacent is left inserted for power suppression of a specific fuel bundle. Another example could be that a control rod is particularly difficult to withdraw off notch position 00 and the control rod is left fully inserted and withdrawn later in the sequence.

Most plant TS permit up to eight inoperable control rods. This allowance for eight inoperable control rods has a wider basis than CRDA. It is customary, however, for the CRDA analysis to allow for eight sequence deviations. For example, the previous GE generic CRDA evaluation (BPWS (Reference 5)) assumed a maximum of eight inoperable control rods based on plant TS. In principle, more than eight sequence deviations could be allowed, but the number eight is judged to be more than adequate for actual plant operation.

[[

]]

CRDA evaluations need to account for out-of-sequence control rods. However, the number and location of these out-of-sequence control rods are not known in advance. [[

]] The standard process evaluates eight out-of-sequence control rods; however, the number of out-of-sequence control rods may be changed to support any desired flexibility.

4.2.1.2 Evaluation Process

[[

]]

4.2.2 [[ ]] Applicability Checklist

[[

]]

Table 4-5: GNF CRDA [[ ]] **Applicability Checklist**

[[		

Table 4-5: GNF CRDA [[ ]] **Applicability Checklist (continued)**

		]]

4.3 Plant- and Cycle-Specific CRDA Evaluation

This section presents an application of the methodology described in Sections 4.1 through 4.2 to the CRDA evaluation for any given plant and fuel cycle design. Control rod patterns developed in accordance with this section are referred to as "analyzed rod position sequence(s)" in the TS.

4.3.1 Inputs

The following inputs are needed to perform a cycle specific CRDA evaluation:

1. Nominal RLP

   [[

   ]]

Input 1 defines the reactor statepoints used to calculate the static reactivity worth of all dropped control rods using PANACEA. [[

]]

4.3.2 Assumptions

[[

]]

4.3.3 Process Outline (Standard Process)

A startup sequence is divided into a number of steps. A "step" is defined as the selection of a single control rod (or a single gang of control rods in a BWR/6), the movement of that rod (or gang) from an initial position to a final position, and then the deselection of that rod (or gang). The control rod(s) may be moved continuously or in single notch increments (or a mixture of both). The step is not completed until the control rod (or gang) is at its target position and is deselected for movement. A startup sequence progresses in steps until all control rods within a group are at a designated bank position. At this point, either the same group of control rods is moved to another bank position, or a different control rod group is chosen for movement.

[[

]]

Potential dropped control rods are considered over a wide range in which the reactor is assumed to be critical. The reactor is assumed to be critical at a given step in the sequence if the calculated eigenvalue is within the range of [[ ]]. The upper bound of the analysis range is provided by either the Low Power Set Point (LPSP) or the Low Dome Pressure Set Point (LDPSP) (See Section 4.4.1). [[

]]

When performing a calculation that models all the control rods [[ ]]. This means that for any given sequence [[ ]].

[[

]]

Figure 4-6: [[ ]]

4.3.4 [[ ]]`

[[

]]

[[

]]

Figure 4-7: [[ ]]

4.3.5 Control Rod Withdrawal Sequence Development

All the control rods in a BWR are divided into a certain number of groups to define the withdrawal sequence. This methodology does not stipulate what the control rod group definitions must be. The group definitions are an input to the evaluation process.

4.3.5.1 Order of Control Rod Withdrawal Within Each Group

The order of control rod withdrawal within each group is an important factor in a CRDA evaluation because it affects the worth of potential accident control rods. [[

]]

The plant operator specifies a single, fixed order of control rod withdrawal within each group. If this order is well-chosen such that it distributes the reactivity worth of the control rods evenly around the core, then there is a high probability that the withdrawal sequence will require minimal bank positions for protection against a postulated CRDA for a large portion of the fuel cycle.

[[

]]

[[

]]

Figure 4-8: [[ ]]

4.3.5.2 Order of Withdrawal of Control Rod Groups

[[

]]

4.3.6 Allowed Out-of-Sequence Control Rods

[[

]]

4.4 Range of Application of Plant- and Cycle-Specific CRDA Evaluation

This section defines the range of application of the GNF plant- and cycle-specific CRDA evaluation. This range is defined by either of two values: the LPSP, or a new pressure threshold, defined here as the LDPSP.

The LPSP has historically been used to define the reactor power level above which there are no constraints on control rod withdrawal order required to protect against a CRDA. The plant operator will constrain the withdrawal order for other reasons, but above the LPSP even the worst possible CRDA does not challenge the fuel cladding failure criteria. The LPSP has been used historically to define the range of application of CRDA evaluation. [[

]] Therefore, the reactor vessel steam dome pressure is used as a second parameter that defines the range of applicability for CRDA evaluations. The LDPSP is defined as the reactor vessel steam dome pressure for a saturated system above which there are no constraints on control rod withdrawal order intended to protect against a CRDA. Above the LDPSP, even the worst possible CRDA does not challenge the fuel cladding failure criteria, as described below.

Because only one of these setpoints needs to be reached, a plant can choose to implement both or just one of these setpoints to define the conditions above which there are no constraints on control rod withdrawal order due to CRDA considerations.

4.4.1 Determination of Numerical Values of LPSP and LDPSP

[[

]]

All of the LPSP and LDPSP results demonstrate large margins to the cladding failure criteria. [[

]] The LPSP is therefore defined as 5% of RTP.

[[

]] A more conservative value may also be chosen by the plant.

[[

]] the LDPSP is defined as 300 psig. [[

]]

Table 4-6: LPSP and LDPSP Results

[[							
							]]

4.4.2 Reactor Mode Switch Discussion

At present, TS require a plant rod pattern control system to be operational in Mode 1 (“Run”) until the LPSP is reached. Based on the values of the LPSP and the LDPSP described in the preceding sections, it is determined that plants' rod pattern control system need only be operational in Mode 2 (“Startup or Hot Standby”).

The basis for this determination is as follows: A plant enters Mode 1 only after reactor power has been raised high enough to clear the APRM downscale setpoint. A typical value for this setpoint is 3% of RTP (Reference 30). More importantly, reactor steam dome pressure must be high enough such that the Main Steam Isolation Valves (MSIVs) will not close on low main steam line pressure. A typical value for MSIV closure in Mode 1 is 825 psig (Reference 30). MSIV closure in Mode 1 results in a reactor scram. Any GE BWR plant therefore is inherently above the LDPSP when entering Mode 1. In actual startup operation, the plant is also expected to be above the LPSP.

4.4.3 Intermediate and Full-Power Application

The basis for the CRDA event in a BWR is limited in scope to startup conditions, specifically below 5% thermal power. [[

]]

4.4.4 Shutdown Insertion Process

Control rod withdrawal sequences developed in accordance with this methodology may be used in reverse order for reactor shutdown. However, an optional shutdown insertion process is available that is simpler than the banked mode that is used for control rod withdrawal. The technical basis for this insertion process is given in Reference 31 and it applies to this methodology without change.

In the optional shutdown insertion process, control rods are fully inserted, either individually or in gangs, to notch position 00 in one step, without stopping at any intermediate positions. Control rods that have not been confirmed to be coupled to their drive mechanisms (i.e., rods that were only partially withdrawn from the core prior to shutdown) must be fully inserted before reactor power falls below the LPSP and dome pressure falls below the LDPSP. The remainder of the control rods shall have had their coupling confirmed previously. Therefore, because control rods that have been confirmed to be coupled are only moved in the inward direction during shutdown, the possibility of a CRDA occurring below the LPSP and/or LDPSP is eliminated.

5.0 DEMONSTRATION APPLICATION

5.1 [[ ]]

[[

]]

5.1.1 [[ ]]

[[

]]

Table 5-1: [[ ]]

[[			
			]]

[[

]]

Figure 5-1: [[ ]]

5.1.2 [[ ]]`

[[

]] The inputs for this demonstration confirmation are summarized in Table 5-2.

Table 5-2: Demonstration Inputs

Parameter	Value
[[	
	]]

[[

]]

5.1.3 TRACG Enthalpy Demonstration

[[

]] The process for performing the TRACG cases is detailed in Section 4.1. [[

]] No cases in Table 5-3 exceed either the HTCF or PCMI criteria. No cladding perforation is calculated by TRACG, and fuel temperatures remain more than 900°F below the melting temperature.

[[

]]

Table 5-3: Out-of-Sequence Control Rods Demonstration Results

[[				
				]]

[[

]]

Figure 5-2: [[ ]]

Note: [[
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[[

]]

Figure 5-3: [[ ]]

[[

]]

Figure 5-4: [[ ]]

[[

]]

Figure 5-5: [[ ]]

5.2 Plant- and Cycle-Specific Application

This section presents a demonstration of the standard plant- and cycle-specific reload evaluation for CRDA. The plant chosen for demonstration is a BWR/5 with a core size of 764 bundles. The core loading consists of approximately one-third GNF2 and two-thirds GE14 fuel. [[

]]

For this demonstration, the following inputs are assumed to be chosen by the plant as input to the CRDA evaluation. They select a control rod group definition similar to those of Reference 5. [[

]] The plant specifies a single, fixed order of withdrawal within each group, [[

]]

Using these inputs, the standard CRDA evaluation is followed (Section 4.3). The minimum banking requirements are determined [[ ]]. Example startup withdrawal sequences that would be provided to the plant are presented in Table 5-5 and Table 5-6 [[ ]].

Table 5-4: Example Control Rod Group Definitions and Withdrawal Order

Order	Group 1	Group 2	Group 3	Group 4	Group 5	Group 6	Group 7	Group 8
1	2-39	2-31	22-59	30-59	2-35	2-43	10-35	10-43
2	58-23	58-31	38-3	30-3	58-27	58-19	50-27	50-19
3	58-39	26-55	38-59	6-35	58-35	58-43	50-35	50-43
4	2-23	34-7	22-3	54-27	2-27	2-19	10-27	10-19
5	18-55	34-55	6-43	54-35	26-59	18-59	26-51	18-51
6	42-7	26-7	54-19	6-27	34-3	42-3	34-11	42-11
7	42-55	10-39	54-43	22-51	34-59	42-59	34-51	42-51
8	18-7	50-23	6-19	38-11	26-3	18-3	26-11	18-11
9	10-47	50-39	14-51	38-51	6-47	10-51	18-43	18-35
10	50-15	10-23	46-11	22-11	54-15	50-11	42-19	42-27
11	50-47	18-47	46-51	14-43	54-47	50-51	42-43	42-35
12	10-15	42-15	14-11	46-19	6-15	10-11	18-19	18-27
13	10-31	42-47	30-51	46-43	14-55		26-35	26-43
14	50-31	18-15	30-11	14-19	46-7		34-27	34-19
15	26-47	18-31	14-35	30-43	46-55		34-35	34-43
16	34-15	42-31	46-27	30-19	14-7		26-27	26-19
17	34-47	26-39	46-35	22-35				
18	26-15	34-23	14-27	38-27				
19	18-39	34-39	22-43	38-35				
20	42-23	26-23	38-19	22-27				
21	42-39		38-43					
22	18-23		22-19					
23	26-31		30-35					
24	34-31		30-27					

Table 5-5: Example Withdrawal Sequences at [[ ]]

[[					
	]]				

Table 5-6 Example Withdrawal Sequences at [[ ]]

[[					
	]]				

In this example, as long as the plant adheres to these given sequences, [[ ]] compliance with the cladding failure thresholds of Sections 3.2, 3.3, 3.4, and 3.5 is ensured. [[

]] Once the plant reaches the LPSP or LDPSP (Section 4.4), it is no longer required to adhere to these banking requirements.

Table 5-7: Example Withdrawal Sequence

[[			
			]]

6.0 METHODOLOGY IMPLEMENTATION

6.1 GESTAR II Changes

This section describes the GESTAR II (Reference 15) changes proposed to incorporate this LTR following its approval. Plants that implement GESTAR II via Technical Specifications (TS), will not need to include a specific reference to this LTR in TS to implement this CRDA methodology.

The following GESTAR II markups include two GESTAR sections and the corresponding references sections.

The GESTAR II Main Section 1.1 changes pertain to the GESTAR II new fuel introduction process. This process is complete when the requirements of Section 1.1 have been completed and documented via the fuel product line compliance report. (This was historically termed the Amendment 22 process.) These changes describe what is required to be performed and documented in the compliance report when applying the methodology in this LTR.

The GESTAR II US Supplement Section S.2.2.3.1.5 changes reflect the application of the Rod Drop Accident Analysis methodology in this LTR. This LTR is much more self-contained than previous methods, hence the level of discussion is much abbreviated compared to the older methods. The methodology described in this LTR and referenced in GESTAR Section S.2.2.3.1.5 does not depend on the other S.2.2.3.1 subsections.

The additions are shown in a **bold blue font**.

6.1.1 Main Section 1.1 Changes

6.1.1.1 Rod Drop Accident Analysis

New fuel designs must satisfy one of the criterion below:

A. Plant cycle specific analysis results shall not exceed the licensing limit described in the country specific supplement to this base document.

B. Applicability of the bounding BPWS analysis must be confirmed.

C. The rod drop accident analysis methodology in Reference 1-16 shall be applied.

Discussions of plant specific and generic rod drop accident evaluation methodologies are presented in the country–specific supplement to this base document.

6.1.1.2 Rod Drop Accident Analysis

A. *Plant cycle specific analysis results shall not exceed the licensing limit in GESTAR–II.*

The current licensing limit of the control rod drop accident analysis is 280 cal/gm. This limit is based on a large amount of margin to reactivity–induced dispersal of the core and the demonstrated conservatism of current models. New models may result in a revision of the licensing limit. The results of this analysis are dependent upon the plant control rod pattern and the fuel loaded in the core. Plants with BPWS rod sequence control currently are covered by a generic analysis for all fuel types up to GE8x8NB. Plants with group notch rod sequence control must be analyzed each cycle to ensure compliance with the licensing criteria. This analysis is performed prior to plant startup each cycle.

B. *Applicability of the bounding BPWS analysis must be confirmed.*

The bounding rod drop accident analysis for plants with BPWS control rod withdrawal sequences is dependent upon the fuel design and must be confirmed generically for each new design. The applicability of the bounding analysis for a new fuel design is determined by comparing the local peaking, Doppler coefficient, and rod worths of the new fuel design with those used for the bounding analyses. The values of the local peaking and Doppler coefficient are obtained from the generic nuclear analyses documented in Subsection 1.2.3. This confirmation will be documented in the fuel design information report for older fuel products (Reference 1-2) and in the compliance reports for GE14 and newer fuel products (See Section 1.4).

C. *The rod drop accident analysis methodology in Reference 1-16 shall be applied.*

The rod drop accident analysis methodology documented in Reference 1-16 defines the fuel, plant, and cycle specific activities associated with the application of this methodology. [[

]] **The cycle specific control rod withdrawal sequence developed using the Reference 1-16 process is included in plant reload documentation. The plant's supplemental reload licensing report (SRLR) documentation will confirm that the Reference 1-16 processes have been applied.**

1.5 References

1-16 Global Nuclear Fuel, *GNF CRDA Application Methodology,* NEDE-33885P-A, Revision 1, March 2020.

6.1.2 US Supplement Section S.2.2.3.1 Changes

S.2.2.3.1.5 Alternate Control Rod Drop Accident Evaluation Based on Reference S-70

This section provides for the application of the Rod Drop Accident Analysis methodology in Reference S-70. This methodology demonstrates that no fuel failures will occur. Therefore, there is no need for a discussion of the number of rod failures.

The cycle specific control rod withdrawal sequence developed using the Reference S-70 process is included in plant reload documentation. The plant's supplemental reload licensing report (SRLR) documentation will confirm that the Reference S-70 processes have been applied.

S.6 References

S-70 Global Nuclear Fuel, *GNF CRDA Application Methodology,* NEDE-33885P-A, Revision 1, TBD.

6.2 Technical Specifications

The current methodology may be implemented without changes to the current TS. However, a review of the current BWR/4 Standard Technical Specifications (STS) (Reference 30), which are based on the BPWS methodology (Reference 5), reveals a number of changes which must be made to allow for the full flexibility provided by the methodology described in this LTR. Plants that do not implement changes to the TS may be limited in their application of this LTR, such that sequences may still need to follow BPWS (Reference 5) related restrictions including, separation criteria on inoperable (or out-of-sequence as defined herein) rods, no more than three inoperable rods in a group, and restrictions on order of groups pulled.

A sample set of TS and Bases markups for the BWR/4 STS (References 30 and 32) which implement the full flexibility of the methodology described in this LTR are included in Appendices A and B, respectively.

7.0 METHODOLOGY UPDATES AND SPECIAL APPLICATIONS

The methodology described in Sections 3.0 and 4.0 includes an example demonstration of the application in Section 5.0. The reasons for and restrictions on updating this methodology are described below, along with requirements on special applications.

7.1 [[ ]]

[[

]]

7.2 Special Applications

7.2.1 [[ ]]

The CRDA licensing basis to confirm no fuel cladding failures is defined as adherence to the criteria of Sections 3.2, 3.3, 3.4, and 3.5. [[

]]

7.2.2 Failure Threshold Criteria Changes

The methodology described in this LTR uses the NRC failure thresholds provided in Reference 7. However, should changes be made to these thresholds in the future, the methodology described here can be applied without requiring additional review and approval provided the new failure thresholds (if more restrictive than those evaluated in this document) can be accommodated by the methodology described in this document.

7.2.3 Hydrogen Model Changes

The hydrogen model provided in Section 3.2.1 is provided by the NRC and applicable for the fuels described herein. However, should a hydrogen model for a given fuel design or range of fuel designs that has been reviewed and approved by the NRC for CRDA applications become available, that model may be applied instead of the model in Section 3.2.1. Use of such an NRC-approved hydrogen model as described in the process of Section 3.2 may be applied without requiring additional review and approval.

7.2.4 Mixed-Cores

This methodology addresses mixed cores [[
]]. As the blade worth calculation in PANACEA is still being performed with the individually defined fuel types in the core, this adequately captures the differences in the fuel types. [[

]]

7.3 Model Updates

The PANACEA core simulator and the TRACG transient thermal hydraulics code have been used as generic terms for Engineering Computer Programs (ECPs) that implement the modeling elements and methodology described in this LTR. The TGBLA lattice physics ECP is used to provide inputs to PANACEA and the PRIME fuel thermal/mechanical ECP is used to provide inputs to TRACG. The maintenance and updating of these ECPs is performed in accordance with the GE-Hitachi Nuclear Energy Americas LLC (GEH) / GNF quality assurance program that complies to 10 CFR 50, Appendix B. Corrections, changes, and improvements in these ECPs that do not fundamentally alter the modeling capabilities required for CRDA and are within the limitations associated with the approved method may be made without prior NRC review and approval. Some examples include changes in the numerical methods to improve efficiency, the addition or enhancement of features that support effective code input/output and automation, or the porting to a new computer platform.

Modifications to the basic models in PANACEA, TRACG, PRIME, or TGBLA that require NRC review and approval for AOO transient, stability, or LOCA applications can also be used in CRDA applications once the updated model has been approved for the other application(s). This includes ongoing improvements to PRIME. Also included would be the replacement of TGBLA with LANCR and/or the replacement of PANACEA with AETNA. This approach acknowledges that the CRDA methodology described herein does not require new modeling but instead relies on applying capability that has already been reviewed and approved by the NRC.

8.0 CONCLUSION

This LTR documents a methodology for determining CRDA consequences to confirm BWRs are within the applicable licensing bases. The proposed methodology evaluates the fuel response in relation to the NRC-provided guidance on the fuel cladding failure thresholds. For BWRs that adhere to withdrawal sequences developed with the methodology described herein, compliance with the required cladding failure criteria is ensured and cladding failure from a postulated CRDA is precluded. The methodology is applicable to all BWR types and all fuel designs for which the methods described in Section 3.0 are applicable.

The technical and regulatory bases for the acceptance criteria are provided by the NRC in SRP Section 4.2 (Reference 6) and modified by a subsequent technical memorandum (Reference 7). There are two types of enthalpy criteria: HTCF criteria which vary with fuel rod internal pressure and PCMI criteria which vary with hydrogen content. The calculations are performed with TRACG, and make use of the PANACEA 3D core simulator. The use of these NRC-approved methods allows for modeling of the event feedback during a CRDA to calculate a realistic enthalpy, temperature, and cladding strain response for the given conditions. [[

]]

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APPENDIX A: EXAMPLE OF CHANGES TO BWR/4 STANDARD TECHNICAL SPECIFICATIONS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	--------------------NOTE------------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation," if required, to allow continued operation. ---	
	A.1 Verify stuck control rod separation criteria are met.	Immediately
	<u>AND</u>	
	A.2 Disarm the associated control rod drive (CRD).	2 hours
	<u>AND</u>	
	A.3 Perform SR 3.1.3.2 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	A.4 Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1 Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1 ---------------NOTE-------------- RWM may be bypassed as allowed by LCO 3.3.2.1, if required, to allow insertion of inoperable control rod and continued operation. ------------------------------------- Fully insert inoperable control rod.	3 hours
	AND	
	C.2 Disarm the associated CRD.	4 hours
~~D. NOTE Not applicable when THERMAL POWER > [10]% RTP.~~ ~~Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.~~	~~D.1 Restore compliance with BPWS.~~ ~~OR~~ ~~D.2 Restore control rod to OPERABLE status.~~	~~4 hours~~ ~~4 hours~~

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~E. NOTE [Not applicable when THERMAL POWER > [10]% RTP.~~ ~~One or more groups with four or more inoperable control rods.~~	~~E.1 Restore control rod to OPERABLE status.~~	~~4 hours]~~
~~F~~D. Required Action and associated Completion Time of Condition A~~,~~ or C~~, D, or E~~ not met. OR Nine or more control rods inoperable.	~~F~~D.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Determine the position of each control rod.	[24 hours OR In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.3.2	-------------------------------NOTE------------------------------ Not required to be performed until 31 days after the control rod is withdrawn and THERMAL POWER is greater than the LPSP of the RWM. --- Insert each withdrawn control rod at least one notch.	[31 days <u>OR</u> In accordance with the Surveillance Frequency Control Program]
SR 3.1.3.3	Verify each control rod scram time from fully withdrawn to notch position [06] is $\leq$ 7 seconds.	In accordance with SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4
SR 3.1.3.4	Verify each control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to declaring control rod OPERABLE after work on control rod or CRD System that could affect coupling

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Rod Pattern Control

LCO 3.1.6 ~~OPERABLE c~~**C**ontrol rods shall comply with the requirements of the **analyzed rod position sequence**~~[banked position withdrawal sequence (BPWS)]~~.

APPLICABILITY: MODE~~S~~ ~~1 and~~ 2 with THERMAL POWER ≤ [**5**~~10~~]% RTP **and reactor steam dome pressure ≤ [300] psig**.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~OPERABLE~~ control rods not in compliance with ~~[BPWS]~~**analyzed rod position sequence**.	A.1 ---------------NOTE-------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1, "Control Rod Block Instrumentation." ------------------------------------- Move associated control rod(s) to correct position.	 8 hours
	<u>OR</u>	
	A.2 ~~Declare associated~~**Fully insert** control rod(s)~~ inoperable~~.	8 hours
B. **[**Nine**]** or more ~~OPERABLE~~ **fully inserted** control rods not in compliance with ~~[BPWS]~~**analyzed rod position sequence**.	B.1 ---------------NOTE-------------- Rod worth minimizer (RWM) may be bypassed as allowed by LCO 3.3.2.1. ------------------------------------- Suspend withdrawal of control rods.	 Immediately
	<u>AND</u>	

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	B.2 Place the reactor mode switch in the shutdown position.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify all ~~OPERABLE~~ control rods comply with ~~[BPWS]~~**analyzed rod position sequence**.	[24 hours <u>OR</u> In accordance with the Surveillance Frequency Control Program]

3.3 INSTRUMENTATION

3.3.2.1A Control Rod Block Instrumentation (Without Setpoint Control Program)

LCO 3.3.2.1A The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.2.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One rod block monitor (RBM) channel inoperable.	A.1 Restore RBM channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two RBM channels inoperable.	B.1 Place one RBM channel in trip.	1 hour
C. Rod worth minimizer (RWM) inoperable during reactor startup.	C.1 Suspend control rod movement except by scram. <u>OR</u> C.2.1.1 Verify $\geq$ 12 rods withdrawn. <u>OR</u>	Immediately Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
	C.2.1.2 Verify by administrative methods that startup with RWM inoperable has not been performed in the last calendar year. AND C.2.2 Verify movement of control rods is in compliance with **the analyzed rod position** ~~banked position withdrawal~~ sequence ~~(BPWS)~~ by a second licensed operator or other qualified member of the technical staff.	Immediately During control rod movement
D. RWM inoperable during reactor shutdown.	D.1 Verify movement of control rods is in compliance with ~~BPWS~~ **the analyzed rod position sequence** by a second licensed operator or other qualified member of the technical staff.	During control rod movement
E. One or more Reactor Mode Switch - Shutdown Position channels inoperable.	E.1 Suspend control rod withdrawal. AND E.2 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

------------------------------NOTE------------------------------

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block Function.

2. When an RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	Perform CHANNEL FUNCTIONAL TEST.	[[92] days <u>OR</u> In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.1.2	---------------NOTE--------------- Not required to be performed until 1 hour after any control rod is withdrawn at ≤ [~~10~~5]% RTP **and ≤ [300] psig reactor steam dome pressure** in MODE 2. ------------------------------------ Perform CHANNEL FUNCTIONAL TEST.	[[92] days <u>OR</u> In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.3	-------------------------------NOTE------------------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ [~~10~~5]% RTP **and reactor steam dome pressure is ≤ [300] psig** in MODE 2~~1~~. --- Perform CHANNEL FUNCTIONAL TEST.	[[92] days <u>OR</u> In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.1.4	-------------------------------NOTE------------------------------ [Neutron detectors are excluded.] --- Verify the RBM: a. Low Power Range - Upscale Function is not bypassed when THERMAL POWER is ≥ 29% and ≤ 64% RTP. b. Intermediate Power Range - Upscale Function is not bypassed when THERMAL POWER is > 64% and ≤ 84% RTP. c. High Power Range - Upscale Function is not bypassed when THERMAL POWER is > 84% RTP.	[[18] months <u>OR</u> In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.1.5	Verify the RWM is not bypassed ~~when THERMAL POWER is ≤ [10]% RTP~~.	[18] months <u>OR</u>

	In accordance with the Surveillance Frequency Control Program]

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.6	------------------------------NOTE------------------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.1.7	------------------------------NOTE------------------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	[[18] months OR In accordance with the Surveillance Frequency Control Program]
SR 3.3.2.1.8	Verify control rod sequences input to the RWM are in conformance with ~~BPWS~~**the analyzed rod position sequence**.	Prior to declaring RWM OPERABLE following loading of sequence into RWM

Table 3.3.2.1-1 (page 1 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range - Upscale	(a)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7[(b)(c)]	≤ [115.5/125] divisions of full scale
b. Intermediate Power Range - Upscale	(d)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7[(b)(c)]	≤ [109.7/125] divisions of full scale
c. High Power Range - Upscale	(e),(f)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7[(b)(c)]	≤ [105.9/125] divisions of full scale
d. Inop	(f),(g)	[2]	SR 3.3.2.1.1	NA
e. Downscale	(f),(g)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.7	≥ [93/125] divisions of full scale
f. Bypass Time Delay	(f),(g)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.7	≤ [2.0] seconds

(a) THERMAL POWER ≥ [29]% and ≤ [64]% RTP and MCPR < 1.70.

(b) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(c) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Limiting Trip Setpoint (LTSP) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the LTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the Surveillance procedures (Nominal Trip Setpoint) to confirm channel performance. The LTSP and the methodologies used to determine the as-found and as-left tolerances are specified in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].

(d) THERMAL POWER > [64]% and ≤ [84]% RTP and MCPR < 1.70.

(e) THERMAL POWER > [84]% and < 90% RTP and MCPR < 1.70.

(f) THERMAL POWER ≥ 90% RTP and MCPR < 1.40.

(g) THERMAL POWER ≥ [64]% and < 90% RTP and MCPR < 1.70.

Table 3.3.2.1-1 (page 2 of 2)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Rod Worth Minimizer	~~1(h),~~ 2(h)	[1]	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch - Shutdown Position	(i)	[2]	SR 3.3.2.1.6	NA

(h) With THERMAL POWER ≤ [~~10~~5]% RTP **and reactor steam dome pressure ≤ [300] psig** [,except during the reactor shutdown process if the coupling of each withdrawn control rod has been confirmed].

(i) Reactor mode switch in the shutdown position.

3.3 INSTRUMENTATION

3.3.1.2B Source Range Monitor (SRM) Instrumentation (With Setpoint Control Program)

Note:

If needed, changes to 3.3.1.2B and its bases should be made consistent with the changes to 3.3.1.2A.

APPENDIX B: EXAMPLE OF CHANGES TO BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Control Rod OPERABILITY

BASES

BACKGROUND

Control rods are components of the Control Rod Drive (CRD) System, which is the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the CRD System provides the means for the reliable control of reactivity changes to ensure under conditions of normal operation, including anticipated operational occurrences, that specified acceptable fuel design limits are not exceeded. In addition, the control rods provide the capability to hold the reactor core subcritical under all conditions and to limit the potential amount and rate of reactivity increase caused by a malfunction in the CRD System. The CRD System is designed to satisfy the requirements of GDC 26, GDC 27, GDC 28, and 29 (Ref. 1).

The CRD System consists of 137 locking piston control rod drive mechanisms (CRDMs) and a hydraulic control unit for each drive mechanism. The locking piston type CRDM is a double acting hydraulic piston, which uses condensate water as the operating fluid. Accumulators provide additional energy for scram. An index tube and piston, coupled to the control rod, are locked at fixed increments by a collet mechanism. The collet fingers engage notches in the index tube to prevent unintentional withdrawal of the control rod, but without restricting insertion.

This Specification, along with LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators," ensure that the performance of the control rods in the event of a Design Basis Accident (DBA) or transient meets the assumptions used in the safety analyses of References 2, 3, and 4.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in the evaluations involving control rods are presented in References 2, 3, and 4. The control rods provide the primary means for rapid reactivity control (reactor scram), for maintaining the reactor subcritical and for limiting the potential effects of reactivity insertion events caused by malfunctions in the CRD System.

The capability to insert the control rods provides assurance that the assumptions for scram reactivity in the DBA and transient analyses are not violated. Since the SDM ensures the reactor will be subcritical with the highest worth control rod withdrawn (assumed single failure), the additional failure of a second control rod to insert, if required, could invalidate the demonstrated SDM and potentially limit the ability of the CRD System to hold the reactor subcritical. If the control rod is stuck at

BASES

APPLICABLE SAFETY ANALYSES (continued)

an inserted position and becomes decoupled from the CRD, a control rod drop accident (CRDA) can possibly occur. Therefore, the requirement that all control rods be OPERABLE ensures the CRD System can perform its intended function.

The control rods also protect the fuel from damage which could result in release of radioactivity. The limits protected are the MCPR Safety Limit (SL) (see Bases for SL 2.1.1, "Reactor Core SLs," and LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), the 1% cladding plastic strain fuel design limit (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"), and the fuel damage limit (see Bases for LCO 3.1.6, "Rod Pattern Control") during reactivity insertion events.

The negative reactivity insertion (scram) provided by the CRD System provides the analytical basis for determination of plant thermal limits and provides protection against fuel damage limits during a CRDA. The Bases for LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6 discuss in more detail how the SLs are protected by the CRD System.

Control rod OPERABILITY satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The OPERABILITY of an individual control rod is based on a combination of factors, primarily, the scram insertion times, the control rod coupling integrity, and the ability to determine the control rod position. Accumulator OPERABILITY is addressed by LCO 3.1.5. The associated scram accumulator status for a control rod only affects the scram insertion times; therefore, an inoperable accumulator does not immediately require declaring a control rod inoperable. Although not all control rods are required to be OPERABLE to satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY - Refueling."

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ACTIONS

The ACTIONS Table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, A.3, and A.4

A control rod is considered stuck if it will not insert by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note, which allows the rod worth minimizer (RWM) to be bypassed if required to allow continued operation. LCO 3.3.2.1, "Control Rod Block Instrumentation," provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, a verification that the separation criteria are met must be performed immediately. The separation criteria are not met if: a) the stuck control rod occupies a location adjacent to two "slow" control rods, b) the stuck control rod occupies a location adjacent to one "slow" control rod, and the one "slow" control rod is also adjacent to another "slow" control rod, or c) if the stuck control rod occupies a location adjacent to one "slow" control rod when there is another pair of "slow" control rods adjacent to one another. The description of "slow" control rods is provided in LCO 3.1.4, "Control Rod Scram Times." In addition, the associated control rod drive must be disarmed in 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. The control rod can be isolated from scram and normal insert and withdraw pressure, yet still maintain cooling water to the CRD.

Monitoring of the insertion capability of each withdrawn control rod must also be performed within 24 hours from discovery of Condition A concurrent with THERMAL POWER greater than the low power setpoint (LPSP) of the RWM. SR 3.1.3.2 performs periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. This Completion Time allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." The Required Action A.2 Completion Time only begins upon discovery of Condition A concurrent

BASES

ACTIONS (continued)

with THERMAL POWER greater than the actual LPSP of the RWM since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RWM (LCO 3.3.2.1). The allowed Completion Time of 24 hours from discovery of Condition A, concurrent with THERMAL POWER greater than the LPSP of the RWM, provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests.

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion, an additional control rod would have to be assumed to fail to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod alsofails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 5).

B.1

With two or more withdrawn control rods stuck, the plant must be brought to MODE 3 within 12 hours. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control

BASES

ACTIONS (continued)

rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. The control rods can be electrically disarmed by disconnecting power from all four directional control valve solenoids. Required Action C.1 is modified by a Note, which allows the RWM to be bypassed if required to allow insertion of the inoperable control rods and continued operation. LCO 3.3.2.1 provides additional requirements when the RWM is bypassed to ensure compliance with the CRDA analysis.

The allowed Completion Times are reasonable, considering the small number of allowed inoperable control rods, and provide time to insert and disarm the control rods in an orderly manner and without challenging plant systems.

~~D.1 and D.2~~

~~Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ 10% RTP, the generic banked position withdrawal sequence (BPWS) analysis (Ref. 5) requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. Condition D is modified by a Note indicating that the Condition is not applicable when > 10% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.~~

~~E.1~~

~~In addition to the separation requirements for inoperable control rods, an assumption in the CRDA analysis for ANF fuel is that no more than three inoperable control rods are allowed in any one BPWS group. Therefore, with one or more BPWS groups having four or more inoperable control rods, the control rods must be restored to OPERABLE status. Required Action E.1 is modified by a Note indicating that the Condition is not applicable when THERMAL POWER is > 10% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.6. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.~~

BASES

ACTIONS (continued)

~~F~~D.1

If any Required Action and associated Completion Time of Condition A~~,~~ or C~~, D, or E~~ are not met, or there are nine or more inoperable control rods, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within
12 hours. This ensures all insertable control rods are inserted and places the reactor in a condition that does not require the active function (i.e., scram) of the control rods. The number of control rods permitted to be inoperable when operating above **[5]**~~10~~% RTP **or [300] psig reactor steam dome pressure** (e.g., no CRDA considerations **as described in the Bases for LCO 3.1.6**) could be more than the value specified, but the occurrence of a large number of inoperable control rods could be indicative of a generic problem, and investigation and resolution of the potential problem should be undertaken. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The position of each control rod must be determined to ensure adequate information on control rod position is available to the operator for determining CRD OPERABILITY and controlling rod patterns. Control rod position may be determined by the use of OPERABLE position indicators, by moving control rods to a position with an OPERABLE indicator, or by the use of other appropriate methods. [The 24 hour Frequency of this SR is based on operating experience related to expected changes in control rod position and the availability of control rod position indications in the control room.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
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SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.3.2

Control rod insertion capability is demonstrated by inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. These Surveillances are not required when THERMAL POWER is less than or equal to the actual LPSP of the RWM, since the notch insertions may not be compatible with the requirements of the ~~Banked Position Withdrawal Sequence (BPWS)~~**analyzed rod position sequence** (LCO 3.1.6) and the RWM (LCO 3.3.2.1). [Withdrawn control rods are tested at a 31 day Frequency, based on the potential power reduction required to allow the control rod movement. Furthermore, the 31 day Frequency takes into account operating experience related to changes in CRD performance.
At any time, if a control rod is immovable, a determination of that control rod's trippability (OPERABILITY) must be made and appropriate action taken.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.1.3.3

Verifying that the scram time for each control rod to notch position 06 is ≤ 7 seconds provides reasonable assurance that the control rod will insert when required during a DBA or transient, thereby completing its shutdown function. This SR is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4. The LOGIC SYSTEM FUNCTIONAL TEST in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and the functional testing of SDV vent and drain valves in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlap this Surveillance to provide complete testing of the assumed safety function. The associated

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SURVEILLANCE REQUIREMENTS (continued)

Frequencies are acceptable, considering the more frequent testing performed to demonstrate other aspects of control rod OPERABILITY and operating experience, which shows scram times do not significantly change over an operating cycle.

SR 3.1.3.4

Coupling verification is performed to ensure the control rod is connected to the CRDM and will perform its intended function when necessary. The Surveillance requires verifying a control rod does not go to the withdrawn overtravel position. The overtravel position feature provides a positive check on the coupling integrity since only an uncoupled CRD can reach the overtravel position. The verification is required to be performed any time a control rod is withdrawn to the "full out" position (notch position 48) or prior to declaring the control rod OPERABLE after work on the control rod or CRD System that could affect coupling. This includes control rods inserted one notch and then returned to the "full out" position during the performance of SR 3.1.3.2. This Frequency is acceptable, considering the low probability that a control rod will become uncoupled when it is not being moved and operating experience related to uncoupling events.

The performance of SR 3.1.3.4 is an assumption of Reference 5.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26, GDC 27, GDC 28, and GDC 29.
2. FSAR, Section [4.2.3.2.2.4].
3. FSAR, Section [5A.4.3].
4. FSAR, Section [15.1].
5. ~~NEDO-21231, "Banked Position Withdrawal Sequence," Section 7.2, January 1977~~**[NEDE-24011-P-A, “General Electric Standard Application for Reactor Fuel,” current revision or NEDE-33885P-A, “GNF CRDA Application Methodology,” Revision 1, TBD.]**.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 Rod Pattern Control

BASES

BACKGROUND

Control rod patterns during startup conditions are controlled by the operator and the rod worth minimizer (RWM) (LCO 3.3.2.1, "Control Rod Block Instrumentation"), so that only specified control rod sequences and relative positions are allowed over the operating range of all control rods inserted to [~~10~~5]% RTP **or [300] psig reactor steam dome pressure**. The sequences limit the potential amount of reactivity addition that could occur in the event of a Control Rod Drop Accident (CRDA).

This Specification assures that the control rod patterns are consistent with the assumptions of the CRDA analyses of References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the CRDA are summarized in References 1 and 2. CRDA analyses assume that the reactor operator follows prescribed withdrawal sequences. These sequences define the potential initial conditions for the CRDA analysis. The RWM (LCO 3.3.2.1) provides backup to operator control of the withdrawal sequences to ensure that the initial conditions of the CRDA analysis are not violated.

Prevention or mitigation of positive reactivity insertion events is necessary to limit the energy deposition in the fuel, thereby preventing significant fuel damage which could result in the undue release of radioactivity. **Control rod patterns analyzed in the cycle-specific analyses are developed in accordance with Ref. 8. The Technical Specifications refer to these patterns as the "analyzed rod position sequence(s)." Per Ref. 8, use of the analyzed rod position sequence ensures** ~~Since the failure consequences for UO_2 have been shown to be insignificant below fuel energy depositions of 300 cal/gm (Ref. 3), the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity (Refs. 4 and 5). Generic evaluations (Refs. 1 and 6) of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm, then the maximum reactor pressure will be less than the required~~ ASME Code limits (Ref. 7) and the calculated offsite doses will be well within the required limits (Ref. 5).

~~Control rod patterns analyzed in Reference 1 follow the banked position withdrawal sequence (BPWS). The BPWS is applicable from the condition of all control rods fully inserted to [10]% RTP (Ref. 2). For the BPWS, the control rods are required to be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions (e.g., between notches 08 and 12). The banked positions are established to minimize the maximum incremental control~~

~~rod worth without being overly restrictive during normal plant operation.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~Generic analysis of the BPWS (Ref. 1) has demonstrated that the 280 cal/gm fuel damage limit will not be violated during a CRDA while following the BPWS mode of operation. The generic BPWS analysis (Ref. 8) also evaluates the effect of fully inserted, inoperable control rods not in compliance with the sequence, to allow a limited number (i.e., eight) and distribution of fully inserted, inoperable control rods.~~

----------------------------------REVIEWER'S NOTE----------------------------------
Adoption of the use of **the optional shutdown insertion process described in** Reference 8~~9~~ requires implementation of the following commitments:

1. Before reducing power to the low power setpoint (LPSP) **and before reactor steam dome pressure falls below [300] psig**, operators shall confirm control rod coupling integrity for all rods that are fully withdrawn. Control rods that have not been confirmed coupled and are in intermediate positions must be fully inserted prior to power reduction to the LPSP **and steam dome pressure reduction to [300] psig**. No action is required for fully-inserted control rods. If a shutdown is required and all rods, which are not confirmed coupled, cannot be fully inserted prior to the power dropping below the LPSP **and the steam dome pressure below [300] psig**, then the original **analyzed rod position sequence**~~/standard BPWS~~ must be used. ~~The original/standard BPWS can be found in Licensing Topical Report NEDO 21231, "Banked Position Withdrawal Sequence," January 1977, and is referred to in NUREG 1433 and NUREG 1434.~~

2. After reactor power drops below the LPSP **and steam dome pressure drops below [300] psig**, rods may be inserted from notch position 48 to notch position 00 without stopping at the intermediate positions. However, ~~GE Nuclear Energy~~**GNF** recommends that operators insert rods in the same order as specified for the original **analyzed rod position sequence** ~~/standard BPWS~~ as much as is reasonably possible. If a plant is in the process of shutting down following **the optional shutdown insertion process** ~~improved BPWS~~ with the power below the LPSP **and reactor steam dome pressure below [300] psig**, no control rod shall be withdrawn unless the control rod pattern is in compliance with ~~standard BPWS requirements~~**the analyzed rod position sequence**.

When performing a shutdown of the plant, an optional ~~BPWS~~ control rod **insertion** sequence (Ref. ~~9~~8) may be used provided that all withdrawn control rods have been confirmed to be coupled. The rods may be inserted without the need to stop at intermediate positions since the possibility of a CRDA is eliminated by the confirmation that withdrawn

BASES

control rods are coupled. When using the Reference 8~~9~~ control rod **insertion** sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements ofthe **optional** ~~improved BPWS~~ control rod insertion process, [or bypassed in accordance with the allowance provided in the Applicability Note for the Rod Worth Minimizer in Table 3.3.2.1-1.]

BASES

APPLICABLE SAFETY ANALYSES (continued)

In order to use the Reference ~~9~~**8** ~~BPWS~~ shutdown process, an extra check is required in order to consider a control rod to be "confirmed" to be coupled. This extra check ensures that no Single Operator Error can result in an incorrect coupling check. For purposes of this shutdown process, the method for confirming that control rods are coupled varies depending on the position of the control rod in the core. Details on this coupling confirmation requirement are provided in Reference ~~9~~**8**. If the requirements for use of the ~~BPWS~~ **optional** control rod insertion process contained in Reference ~~9~~ **8** are followed, the plant is considered to be in compliance with ~~BPWS requirements~~**the analyzed rod position sequence**, as required by LCO 3.1.6.

Rod pattern control satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Compliance with the prescribed control rod sequences minimizes the potential consequences of a CRDA by limiting the initial conditions to those consistent with the **analyzed rod position sequence**~~BPWS~~. This LCO ~~only applies to OPERABLE~~**applies to all** control rods**, whether operable or inoperable**. ~~For inoperable control rods required to be inserted, separate requirements are specified in LCO 3.1.3, "Control Rod OPERABILITY," consistent with the allowances for inoperable control rods in the BPWS.~~

APPLICABILITY

In MODE~~S 1 and~~ 2, when THERMAL POWER is ≤ [~~10~~**5**]% RTP **and reactor steam dome pressure is ≤ [300] psig**, the CRDA is a Design Basis Accident and, therefore, compliance with the assumptions of the safety analysis is required. When THERMAL POWER is > [~~10~~**5**]% RTP **or reactor steam dome pressure is > [300] psig**, there is no credible control rod configuration that results in a control rod worth that could exceed the ~~280 cal/gm fuel damage limit~~**fuel cladding failure criteria** during a CRDA (Ref. ~~2~~**8**). In MODES 3, 4, and 5, since the reactor is shut down and only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will remain subcritical with a single control rod withdrawn. **Before entering MODE 1, the reactor has completed heat up and pressurization. Reactor steam dome pressure is therefore above 300 psig, and so constraints on the control rod pattern due to CRDA are not required in MODE 1.**

BASES

ACTIONS

A.1 and A.2

With one or more ~~OPERABLE~~ control rods not in compliance with the prescribed control rod sequence, actions may be taken to either correct the control rod pattern or ~~declare the associated control rods inoperable~~**fully insert the associated control rods** within 8 hours. Noncompliance with the prescribed sequence may be the result of "double notching," drifting from a control rod drive cooling water transient, leaking scram valves, or a power reduction to ≤ [~~10~~**5**]% RTP **and ≤ [300] psig reactor steam dome pressure** before establishing the correct control rod pattern. The number of ~~OPERABLE~~ control rods not in compliance with the prescribed sequence is limited to **[**eight**]**, to prevent the operator from attempting to correct a control rod pattern that significantly deviates from the prescribed sequence. When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted.

Required Action A.1 is modified by a Note which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff. This ensures that the control rods will be moved to the correct position. A control rod not in compliance with the prescribed sequence is not **necessarily** considered inoperable ~~except as required by Required Action A.2~~. OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." The allowed Completion Time of 8 hours is reasonable, considering the restrictions on the number of allowed out of sequence control rods and the low probability of a CRDA occurring during the time the control rods are out of sequence.

B.1 and B.2

If **[**nine**]** or more ~~OPERABLE~~ control rods are out of sequence, the control rod pattern significantly deviates from the prescribed sequence. Control rod withdrawal should be suspended immediately to prevent the potential for further deviation from the prescribed sequence. Control rod insertion to correct control rods withdrawn beyond their allowed position is allowed since, in general, insertion of control rods has less impact on control rod worth than withdrawals have. Required Action B.1 is modified by a Note

BASES

ACTIONS (continued)

which allows the RWM to be bypassed to allow the affected control rods to be returned to their correct position. LCO 3.3.2.1 requires verification of control rod movement by a qualified member of the technical staff.

When [nine] or more ~~OPERABLE~~ control rods are not in compliance with **the analyzed rod position sequence**~~BPWS~~, the reactor mode switch must be placed in the shutdown position within 1 hour. With the mode switch in shutdown, the reactor is shut down, and as such, does not meet the applicability requirements of this LCO. The allowed Completion Time of 1 hour is reasonable to allow insertion of control rods to restore compliance, and is appropriate relative to the low probability of a CRDA occurring with the control rods out of sequence.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

The control rod pattern is periodically verified to be in compliance with the **analyzed rod position sequence**~~BPWS~~ to ensure the assumptions of the CRDA analyses are met. [The 24 hour Frequency was developed considering that the primary check on compliance with the **analyzed rod position sequence**~~BPWS~~ is performed by the RWM (LCO 3.3.2.1).

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

The RWM provides control rod blocks to enforce the required sequence and is required to be OPERABLE when operating at ≤ [5~~10~~]% RTP **and ≤ [300] psig reactor steam dome pressure**.

REFERENCES

1. NEDE-24011-P-A-9-US, "General Electric Standard Application for Reactor Fuel, Supplement for United States," Section 2.2.3.1, September 1988.

2. ~~"Modifications to the Requirements for Control Rod Drop Accident Mitigating System," BWR Owners Group, July 1986~~**Deleted**.

3. NUREG-0979, Section 4.2.1.3.2, April 1983.

4. NUREG-0800, Section 15.4.9, Revision 2, July 1981.

BASES

REFERENCES (continued)

5. 10 CFR 100.11.

6. NEDO-21778-A, "Transient Pressure Rises Affected Fracture Toughness Requirements for Boiling Water Reactors," December 1978.

7. ASME, Boiler and Pressure Vessel Code.

8. ~~NEDO-21231, "Banked Position Withdrawal Sequence," January 1977~~**NEDE-33885P-A, "GNF CRDA Application Methodology," Revision 1, TBD**.

9. ~~NEDO-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004~~**Deleted**.

B 3.3 INSTRUMENTATION

B 3.3.2.1A Control Rod Block Instrumentation (Without Setpoint Control Program)

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod block monitor (RBM) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod worth minimizer (RWM) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch - Shutdown Position Function ensure that all control rods remain inserted to prevent inadvertent criticalities.

The protection and monitoring functions of the control rod block instrumentation has been designed to ensure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the Reactor Protection System (RPS), as well as LCOs on other reactor system parameters and equipment performance.

Technical Specifications are required by 10 CFR 50.36 to include LSSS for variables that have significant safety functions. LSSS are defined by the regulation as "Where a LSSS is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective actions will correct the abnormal situation before a Safety Limit (SL) is exceeded." The Analytical Limit is the limit of the process variable at which a safety action is initiated, as established by the safety analysis, to ensure that a SL is not exceeded. Any automatic protection action that occurs on reaching the Analytical Limit therefore ensures that the SL is not exceeded. However, in practice, the actual settings for automatic protection channels must be chosen to be more conservative than the Analytical Limit to account for instrument loop uncertainties related to the setting at which the automatic protective action would actually occur.

------------------------------- REVIEWER'S NOTE ----------------------------------
The term "Limiting Trip Setpoint" [LTSP] is generic terminology for the calculated trip setting (setpoint) value calculated by means of the plant specific setpoint methodology documented in a document controlled under 10 CFR 50.59. The term [LTSP] indicates that no additional margin has been added between the Analytical Limit and the calculated trip setting.

BASES

BACKGROUND (continued)

"Nominal Trip Setpoint [NTSP]" is the suggested terminology for the actual setpoint implemented in the plant surveillance procedures where margin has been added to the calculated [LTSP]. The as-found and as-left tolerances will apply to the [NTSP] implemented in the Surveillance procedures to confirm channel performance.

Licensees are to insert the name of the document(s) controlled under 10 CFR 50.59 that contain the methodology for calculating the as-left and as-found tolerances, in Note c of Table 3.3.2.1-1 for the phrase "[insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]" throughout the Bases.

If the [LTSP] is not included in Table 3.3.2.1-1, the plant specific location for the [LTSP] or [NTSP] must be cited in Note c of Table 3.3.2.1-1. The brackets indicate plant specific terms may apply, as reviewed and approved by the NRC.

The [Limiting Trip Setpoint (LTSP)] specified in Table 3.3.2.1-1, is a predetermined setting for a protection channel chosen to ensure automatic actuation prior to the process variable reaching the Analytical Limit and thus ensuring that the SL would not be exceeded. As such, the [LTSP] accounts for uncertainties in setting the channel (e.g., calibration), uncertainties in how the channel might actually perform (e.g., repeatability), changes in the point of action of the channel over time (e.g., drift during surveillance intervals), and any other factors which may influence its actual performance (e.g., harsh accident environments). In this manner, the [LTSP] ensures that SLs are not exceeded. Therefore, the [LTSP] meets the definition of an LSSS (Ref. 1).

The Allowable Values specified in Table 3.3.2.1-1 serves as the LSSS such that a channel is OPERABLE if the trip setpoint is found not to exceed the Allowable Value. As such, the Allowable Value differs from the trip setpoint by an amount primarily equal to the expected instrument loop uncertainties, such as drift, during the surveillance interval. In this manner, the actual setting of the device will still meet the LSSS definition and ensure that a SL is not exceeded at any given point of time as long as the device has not drifted beyond that expected during the surveillance interval.

Technical Specifications contain values related to the OPERABILITY of equipment required for safe operation of the facility. OPERABLE is defined in Technical Specifications as "...being capable of performing its safety function(s)." Relying solely on the [LTSP] to define OPERABILITY in Technical Specifications would be an overly restrictive requirement if it

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BACKGROUND (continued)

were applied as an OPERABILITY limit for the "as found" value of a protection channel setting during a Surveillance. This would result in Technical Specification compliance problems, as well as reports and corrective actions required by the rule which are not necessary to ensure safety. For example, an automatic protection channel with a setting that has been found to be different from the [LTSP] due to some drift of the setting may still be OPERABLE because drift is to be expected. This expected drift would have been specifically accounted for in the setpoint methodology for calculating the [LTSP] and thus the automatic protective action would still have ensured that the SL would not be exceeded with the "as found" setting of the protection channel. Therefore, the channel would still be OPERABLE because it would have performed its safety function and the only corrective action required would be to reset the channel within the established as-left tolerance around [LTSP] to account for further drift during the next surveillance interval. Note that, although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance, in accordance with uncertainty assumptions stated in the referenced setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

However, there is also some point beyond which the channel would have not been able to perform its function due to, for example, greater than expected drift. This value needs to be specified in the Technical Specifications in order to define OPERABILITY of the channels and is designated as the Allowable Value.

If the actual setting (as-found setpoint) of the channel is found to be conservative with respect to the Allowable Value but is beyond the as-found tolerance band, the channel is OPERABLE, but degraded. The degraded condition will be further evaluated during performance of the SR. This evaluation will consist of resetting the channel setpoint to the [LTSP] (within the allowed tolerance), and evaluating the channel response. If the channel is functioning as required and expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition will be entered into the Corrective Action Program for further evaluation.

The purpose of the RBM is to limit control rod withdrawal if localized neutron flux exceeds a predetermined setpoint during control rod manipulations. It is assumed to function to block further control rod withdrawal to preclude a MCPR SL violation. The RBM supplies a trip signal to the Reactor Manual Control System (RMCS) to appropriately

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BACKGROUND (continued)

inhibit control rod withdrawal during power operation above the low power range setpoint. The RBM has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. One RBM channel inputs into one RMCS rod block circuit and the other RBM channel inputs into the second RMCS rod block circuit. The RBM channel signal is generated by averaging a set of local power range monitor (LPRM) signals at various core heights surrounding the control rod being withdrawn. A signal from one average power range monitor (APRM) channel assigned to each RPS trip system supplies a reference signal for the RBM channel in the same trip system. This reference signal is used to determine which RBM range setpoint (low, intermediate, or high) is enabled. If the APRM is indicating less than the low power range setpoint, the RBM is automatically bypassed. The RBM is also automatically bypassed if a peripheral control rod is selected (Ref. 2).

The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to **[5]**~~10~~% RTP **or [300] psig reactor steam dome pressure**. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. Prescribed control rod sequences are stored in the RWM, which will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the stored sequence. The RWM determines the actual sequence based position indication for each control rod. The RWM also uses feedwater flow and steam flow signals to determine when the reactor power is above the preset power level at which the RWM is automatically bypassed (Ref. 3). The RWM is a single channel system that provides input into both RMCS rod block circuits.

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This Function prevents inadvertent criticality as the result of a control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, each inputting into a separate RMCS rod block circuit. A rod block in either RMCS circuit will provide a control rod block to all control rods.

Permissive and interlock setpoints allow the blocking of trips during plant startups, and restoration of trips when the permissive conditions are not satisfied, but they are not explicitly modeled in the Safety Analyses. These permissives and interlocks ensure that the starting conditions are consistent with the safety analysis, before preventive or mitigating actions occur. Because these permissives or interlocks are only one of multiple

BASES

BACKGROUND (continued)

conservative starting assumptions for the accident analysis, they are generally considered as nominal values without regard to measurement accuracy.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Allowable Values are specified for each Rod Block Function specified in SR 3.3.2.1.7. [LTSP] and the methodologies for calculation of the as-left and as-found tolerances are described in [insert the name of a document controlled under 10 CFR 50.59 such as the Technical Requirements Manual or any document incorporated into the facility FSAR]. The [LTSPs] are selected to ensure that the actual setpoints remain conservative with respect to the as-found tolerance band between successive CHANNEL CALIBRATIONS. After each calibration the trip setpoint shall be left within the as-left band around the [LTSP].

[LTSPs] are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor vessel water level), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytical limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytical limits, corrected for calibration, process, and some of the instrument errors. The [LTSPs] are then determined accounting for the remaining instrument errors (e.g., drift). The [LTSPs] derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

1. Rod Block Monitor

The RBM is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 4. A statistical analysis of RWE events was performed to determine the RBM response for both channels for each event. From these responses, the fuel thermal performance as a function of RBM Allowable Value was determined. The Allowable Values are chosen as a function of power level. Based on the specified Allowable Values, operating limits are established.

The RBM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

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APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value for the associated power range, to ensure that no single instrument failure can preclude a rod block from this Function. The actual setpoints are calibrated consistent with applicable setpoint methodology.

The RBM is assumed to mitigate the consequences of a RWE event when operating ≥ 29% RTP. Below this power level, the consequences of a RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 4). When operating < 90% RTP, analyses (Ref. 4) have shown that with an initial MCPR ≥ 1.70, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.40, no RWE event will result in exceeding the MCPR SL (Ref. 4). Therefore, under these conditions, the RBM is also not required to be OPERABLE.

2. Rod Worth Minimizer

The RWM enforces the **analyzed rod position** ~~banked position withdrawal~~ sequence ~~(BPWS)~~ to ensure that the initial conditions of the CRDA analysis are not violated.
The analytical methods and assumptions used in evaluating the CRDA are summarized in References 5, 6, 7~~, 8~~, and 9. ~~The standard BPWS requires that control rods be moved in groups, with all control rods assigned to a specific group required to be within specified banked positions.~~ Requirements that the control rod sequence is in compliance with the **analyzed rod position sequence**~~BPWS~~ are specified in LCO 3.1.6, "Rod Pattern Control."

----------------------------------REVIEWER'S NOTE-----------------------------------
Adoption of the use of **the optional shutdown insertion process in** Reference ~~8~~**7** requires implementation of the following commitments:

1. Before reducing power to the low power setpoint (LPSP) **and before reactor steam dome pressure falls below [300] psig**, operators shall confirm control rod coupling integrity for all rods that are fully withdrawn. Control rods that have not been confirmed coupled and are in intermediate positions must be fully inserted prior to power reduction to the LPSP **and steam dome pressure reduction to [300] psig**. No action is required for fully-inserted control rods. If a shutdown is required and all rods, which are not confirmed coupled, cannot be fully inserted prior to the power dropping below the LPSP **and the steam dome pressure below [300] psig**, then the original **analyzed rod position sequence**~~/standard BPWS~~ must be used. ~~The original/standard BPWS can be found in Licensing Topical Report NEDO 21231, "Banked Position Withdrawal Sequence," January 1977, and is~~

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~~referred to in NUREG-1433 and NUREG-1434.~~

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. After reactor power drops below the LPSP **and steam dome pressure drops below [300] psig**, rods may be inserted from notch position 48 to notch position 00 without stopping at the intermediate positions. However, ~~GE Nuclear Energy~~**GNF** recommends that operators insert rods in the same order as specified for the original **analyzed rod position sequence**~~/standard BPWS~~ as much as is reasonably possible. If a plant is in the process of shutting down following **the optional shutdown insertion process** ~~improved BPWS~~ with the power below the LPSP **and reactor steam dome pressure below [300] psig**, no control rod shall be withdrawn unless the control rod pattern is in compliance with ~~standard BPWS requirements~~**the analyzed rod position sequence**.

When performing a shutdown of the plant, an optional ~~BPWS~~ control rod **insertion** sequence (Ref. **7**~~8~~) may be used if the coupling of each withdrawn control rod has been confirmed. The rods may be inserted without the need to stop at intermediate positions. When using the Reference ~~8~~ **7** control rod insertion sequence for shutdown, the rod worth minimizer may be reprogrammed to enforce the requirements of the **optional** ~~improved BPWS~~ control rod insertion process[,or it can be bypassed if it is not programmed to reflect the optional ~~BPWS~~ shutdown sequence, as permitted by the Applicability Note for the RWM in Table 3.3.2.1-1].

The RWM Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Since the RWM is a hardwired system designed to act as a backup to operator control of the rod sequences, only one channel of the RWM is available and required to be OPERABLE (Ref. 9). Special circumstances provided for in the Required Action of LCO 3.1.3, "Control Rod OPERABILITY," and LCO 3.1.6 may necessitate bypassing the RWM to allow continued operation with ~~inoperable~~ **fully inserted, out-of-sequence** control rods, or to allow correction of a control rod pattern not in compliance with the **analyzed rod position sequence**~~BPWS~~. The RWM may be bypassed as required by these conditions, but then it must be considered inoperable and the Required Actions of this LCO followed.

Compliance with the **analyzed rod position sequence**~~BPWS~~, and therefore OPERABILITY of the RWM, is required in MODE~~S 1 and~~ 2 when THERMAL POWER is **≤**~~<~~ **[5**~~10~~**]**% RTP **and reactor steam dome pressure ≤ [300] psig**. When THERMAL POWER is > **[5**~~10~~**]**% RTP **or reactor steam dome pressure > [300] psig**, there is no possible control rod configuration that results in a control rod worth that could exceed the ~~280 cal/gm fuel damage limit~~**fuel cladding failure criteria** during a CRDA (Ref~~s~~. **7**~~6 and 9~~). In
MODES 3 and 4, all control rods are required to be inserted into the core;

BASES

therefore, a CRDA cannot occur. In MODE 5, since only a single control rod can be withdrawn from a core cell containing fuel assemblies, adequate SDM ensures that the consequences of a CRDA are acceptable, since the reactor will be subcritical. **Before entering MODE 1, the reactor has completed heat up and pressurization. Reactor steam dome pressure is therefore above 300 psig, and so constraints on the control rod pattern due to CRDA are not required in MODE 1.**

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. Reactor Mode Switch - Shutdown Position

During MODES 3 and 4, and during MODE 5 when the reactor mode switch is required to be in the shutdown position, the core is assumed to be subcritical; therefore, no positive reactivity insertion events are analyzed. The Reactor Mode Switch - Shutdown Position control rod withdrawal block ensures that the reactor remains subcritical by blocking control rod withdrawal, thereby preserving the assumptions of the safety analysis.

The Reactor Mode Switch - Shutdown Position Function satisfies Criterion 3 of 10 CFR 50.36(c)(2)(ii).

Two channels are required to be OPERABLE to ensure that no single channel failure will preclude a rod block when required. There is no Allowable Value for this Function since the channels are mechanically actuated based solely on reactor mode switch position.

During shutdown conditions (MODE 3, 4, or 5), no positive reactivity insertion events are analyzed because assumptions are that control rod withdrawal blocks are provided to prevent criticality. Therefore, when the reactor mode switch is in the shutdown position, the control rod withdrawal block is required to be OPERABLE. During MODE 5 with the reactor mode switch in the refueling position, the refuel position one-rod-out interlock (LCO 3.9.2) provides the required control rod withdrawal blocks.

BASES

ACTIONS

-----------------------------------REVIEWER'S NOTE-----------------------------------
Certain LCO Completion Times are based on approved topical reports. In order for the licensee to use the times, the licensee must justify the Completion Times as required by the staff Safety Evaluation Report (SER) for the topical report.

A.1

With one RBM channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod block function; however, overall reliability is reduced because a single failure in the remaining OPERABLE channel can result in no control rod block capability for the RBM. For this reason, Required Action A.1 requires restoration of the inoperable channel to OPERABLE status. The Completion Time of 24 hours is based on the low probability of an event occurring coincident with a failure in the remaining OPERABLE channel.

B.1

If Required Action A.1 is not met and the associated Completion Time has expired, the inoperable channel must be placed in trip within 1 hour. If both RBM channels are inoperable, the RBM is not capable of performing its intended function; thus, one channel must also be placed in trip. This initiates a control rod withdrawal block, thereby ensuring that the RBM function is met.

The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping of inoperable channels.

C.1, C.2.1.1, C.2.1.2, and C.2.2

With the RWM inoperable during a reactor startup, the operator is still capable of enforcing the prescribed control rod sequence. However, the overall reliability is reduced because a single operator error can result in violating the control rod sequence. Therefore, control rod movement must be immediately suspended except by scram. Alternatively, startup may continue if at least 12 control rods have already been withdrawn, or a reactor startup with an inoperable RWM was not performed in the last 12 months. Required Actions C.2.1.1 and C.2.1.2 require verification of these conditions by review of plant logs and control room indications. Once Required Action C.2.1.1 or C.2.1.2 is satisfactorily completed,

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ACTIONS (continued)

control rod withdrawal may proceed in accordance with the restrictions imposed by Required Action C.2.2. Required Action C.2.2 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff.

The RWM may be bypassed under these conditions to allow continued operations. In addition, Required Actions of LCO 3.1.3 and LCO 3.1.6 may require bypassing the RWM, during which time the RWM must be considered inoperable with Condition C entered and its Required Actions taken.

D.1

With the RWM inoperable during a reactor shutdown, the operator is still capable of enforcing the prescribed control rod sequence. Required Action D.1 allows for the RWM Function to be performed manually and requires a double check of compliance with the prescribed rod sequence by a second licensed operator (Reactor Operator or Senior Reactor Operator) or other qualified member of the technical staff. The RWM may be bypassed under these conditions to allow the reactor shutdown to continue.

E.1 and E.2

With one Reactor Mode Switch - Shutdown Position control rod withdrawal block channel inoperable, the remaining OPERABLE channel is adequate to perform the control rod withdrawal block function. However, since the Required Actions are consistent with the normal action of an OPERABLE Reactor Mode Switch - Shutdown Position Function (i.e., maintaining all control rods inserted), there is no distinction between having one or two channels inoperable.

In both cases (one or both channels inoperable), suspending all control rod withdrawal and initiating action to fully insert all insertable control rods in core cells containing one or more fuel assemblies will ensure that the core is subcritical with adequate SDM ensured by LCO 3.1.1. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and are therefore not required to be inserted. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies are fully inserted.

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SURVEILLANCE REQUIREMENTS

-----------------------------------REVIEWER'S NOTE-----------------------------------
Certain Frequencies are based on approved topical reports. In order for a licensee to use these Frequencies, the licensee must justify the Frequencies as required by the staff SER for the topical report.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Notes b and c are applied to the setpoint verification Surveillances for the Control Rod Block Instrumentation Functions in Table 3.3.2.1-1 unless one or more of the following exclusions apply:

1. Manual actuation circuits, automatic actuation logic circuits or instrument functions that derive input from contacts which have no associated sensor or adjustable device, e.g., limit switches, breaker position switches, manual actuation switches, float switches, proximity detectors, etc. are excluded. In addition, those permissives and interlocks that derive input from a sensor or adjustable device that is tested as part of another TS function are excluded.

2. Settings associated with safety relief valves are excluded. The performance of these components is already controlled (i.e., trended with as-left and as-found limits) under the ASME Code for Operation and Maintenance of Nuclear Power Plants testing program.

3. Functions and Surveillance Requirements which test only digital components are normally excluded. There is no expected change in result between SR performances for these components. Where separate as-left and as-found tolerance is established for digital component SRs, the requirements would apply.

As noted at the beginning of the SRs, the SRs for each Control Rod Block instrumentation Function are found in the SRs column of Table 3.3.2.1-1.

The Surveillances are modified by a Note to indicate that when a RBM channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains control rod block capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Ref. 11) assumption of the average time required to perform channel Surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that a control rod block will be initiated when necessary.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.2.1.1

A CHANNEL FUNCTIONAL TEST is performed for each RBM channel to ensure that the entire channel will perform the intended function. It includes the Reactor Manual Control Multiplexing System input. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. [The Frequency of 92 days is based on reliability analyses (Ref. 10).

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.3.2.1.2 and SR 3.3.2.1.3

A CHANNEL FUNCTIONAL TEST is performed for the RWM to ensure that the entire system will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the RWM is performed by attempting to withdraw a control rod not in compliance with the prescribed sequence and verifying a control rod block occurs. As noted in the SRs, SR 3.3.2.1.2 is not required to be performed until 1 hour after any control rod is withdrawn **at ≤ [5]% RTP and ≤ [300] psig reactor steam dome pressure** in MODE 2. As noted, SR 3.3.2.1.3 is

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SURVEILLANCE REQUIREMENTS (continued)

not required to be performed until 1 hour after THERMAL POWER is ≤ ~~10~~**[5]**% RTP **and steam dome pressure is ≤ [300] psig** in MODE ~~1~~**2**. This allows entry into MODE 2 **during a startup** for SR 3.3.2.1.2, and entry into MODE ~~1~~ **2 during a shutdown** when THERMAL POWER is ≤ ~~10~~**[5]**% RTP **and steam dome pressure is ≤ [300] psig** for SR 3.3.2.1.3, to perform the required Surveillance if the Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs. [The Frequencies are based on reliability analysis (Ref. 10).

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.3.2.1.4

The RBM setpoints are automatically varied as a function of power. Three Allowable Values are specified in Table 3.3.2.1-1, each within a specific power range. The power at which the control rod block Allowable Values automatically change are based on the APRM signal's input to each RBM channel. Below the minimum power setpoint, the RBM is automatically bypassed. These power Allowable Values must be verified periodically to be less than or equal to the specified values. If any power range setpoint is nonconservative, then the affected RBM channel is considered inoperable. Alternatively, the power range channel can be placed in the conservative condition (i.e., enabling the proper RBM setpoint). If placed in this condition, the SR is met and the RBM channel is not considered inoperable. As noted, neutron detectors are excluded from the Surveillance because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6. [The 18 month Frequency is based on the actual trip setpoint methodology utilized for these channels.

OR

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SURVEILLANCE REQUIREMENTS (continued)

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.3.2.1.5

The RWM is automatically bypassed when power is above a specified value. The power level is determined from feedwater flow and steam flow signals. The automatic bypass setpoint must be verified periodically to be ≤ [~~10~~5]% RTP. If the RWM low power setpoint is nonconservative, then the RWM is considered inoperable. Alternately, the low power setpoint channel can be placed in the conservative condition (nonbypass). If placed in the nonbypassed condition, the SR is met and the RWM is not considered inoperable. [The Frequency is based on the trip setpoint methodology utilized for the low power setpoint channel.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
---]

SR 3.3.2.1.6

A CHANNEL FUNCTIONAL TEST is performed for the Reactor Mode Switch - Shutdown Position Function to ensure that the entire channel will perform the intended function. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL FUNCTIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other

BASES

SURVEILLANCE REQUIREMENTS (continued)

Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The CHANNEL FUNCTIONAL TEST for the Reactor Mode Switch - Shutdown Position Function is performed by attempting to withdraw any control rod with the reactor mode switch in the shutdown position and verifying a control rod block occurs.

As noted in the SR, the Surveillance is not required to be performed until 1 hour after the reactor mode switch is in the shutdown position, since testing of this interlock with the reactor mode switch in any other position cannot be performed without using jumpers, lifted leads, or movable links. This allows entry into MODES 3 and 4 if the Frequency is not met per SR 3.0.2. The 1 hour allowance is based on operating experience and in consideration of providing a reasonable time in which to complete the SRs.

[The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

<u>SR 3.3.2.1.7</u>

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

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SURVEILLANCE REQUIREMENTS (continued)

As noted, neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Neutron detectors are adequately tested in SR 3.3.1.1.2 and SR 3.3.1.1.6.

[The Frequency is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

OR

The Surveillance Frequency is controlled under the Surveillance Frequency Control Program.

-----------------------------------REVIEWER'S NOTE-----------------------------------
Plants controlling Surveillance Frequencies under a Surveillance Frequency Control Program should utilize the appropriate Frequency description, given above, and the appropriate choice of Frequency in the Surveillance Requirement.
--]

SR 3.3.2.1.7 for Functions [3.3.2.1-1.1.a, 3.3.2.1-1.1.b, and 3.3.2.1-1.1.c] is modified by two Notes as identified in Table 3.3.2.1-1. The first Note requires evaluation of channel performance for the condition where the as-found setting for the channel setpoint is outside its as-found tolerance but conservative with respect to the Allowable Value. Evaluation of channel performance will verify that the channel will continue to behave in accordance with safety analysis assumptions and the channel performance assumptions in the setpoint methodology. The purpose of the assessment is to ensure confidence in the channel performance prior to returning the channel to service. For channels determined to be OPERABLE but degraded, after returning the channel to service the performance of these channels will be evaluated under the plant Corrective Action Program. Entry into the Corrective Action Program will ensure required review and documentation of the condition. The second Note requires that the as-left setting for the channel be within the as-left tolerance of the [LTSP]. Where a setpoint more conservative than the [LTSP] is used in the plant surveillance procedures [Nominal Trip Setpoint (NTSP)], the as-left and as-found tolerances, as applicable, will be applied to the surveillance procedure setpoint. This will ensure that sufficient margin to the Safety Limit and/or Analytical Limit is maintained. If the as-left channel setting cannot be returned to a setting within the as- left tolerance of the [LTSP], then the channel shall be declared inoperable. The second Note also

BASES

SURVEILLANCE REQUIREMENTS (continued)

requires that [LTSP] and the methodologies for calculating the as-left and the as-found tolerances be in [insert the facility FSAR reference or the name of any document incorporated into the facility FSAR by reference].

SR 3.3.2.1.8

The RWM will only enforce the proper control rod sequence if the rod sequence is properly input into the RWM computer. This SR ensures that the proper sequence is loaded into the RWM so that it can perform its intended function. The Surveillance is performed once prior to declaring RWM OPERABLE following loading of sequence into RWM, since this is when rod sequence input errors are possible.

REFERENCES

1. Regulatory Guide 1.105, "Setpoints for Safety-Related Instrumentation," Revision 3.

2. FSAR, Section [7.6.2.2.5].

3. FSAR, Section [7.6.8.2.6].

4. NEDC-30474-P, "Average Power Range Monitor, Rod Block Monitor, and Technical Specification Improvements (ARTS) Program for Edwin I. Hatch Nuclear Plants," December 1983.

5. NEDE-24011-P-A-9-US, "General Electrical Standard Application for Reload Fuel," Supplement for United States, Section S 2.2.3.1, September 1988.

6. "Modifications to the Requirements for Control Rod Drop Accident Mitigating Systems," BWR Owners' Group, July 1986.

7. ~~NEDO-21231, "Banked Position Withdrawal Sequence," January 1977~~**NEDE-33885P-A, "GNF CRDA Application Methodology," Revision 1, TBD**.

8. ~~NEDO-33091-A, Revision 2, "Improved BPWS Control Rod Insertion Process," July 2004~~**Deleted**.

9. NRC SER, "Acceptance of Referencing of Licensing Topical Report NEDE-24011-P-A," "General Electric Standard Application for Reactor Fuel, Revision 8, Amendment 17," December 27, 1987.

BASES

REFERENCES (continued)

10. NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988.

11. GENE-770-06-1, "Addendum to Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications," February 1991.

APPENDIX C: NRC REQUESTS FOR ADDITIONAL INFORMATION (RAIS) WITH GNF RESPONSES

NRC RAI #1

The primary mitigation mechanism for the CRDA event is the Doppler reactivity effect as the fuel temperature increases. This effect is determined by [[

]]. More information is needed to fully understand [[

]]:

a. Provide a breakdown of the data from Figure 4-3 [[

b.

]].

GNF Response

As currently implemented for most other processes, [[

]] These lattice definitions in order of increasing height in the bundle are listed [[
]] in Table1. [[

]]

The range of lattice enrichments, range of pin enrichments, range of gadolinium rod numbers, and range of gadolinium enrichments are presented for all lattices in Table 2. This significant range of parameters shows that [[ ]], while changes to gadolinium and enrichment are expected to have minimal impact.

[[

]]

The text shown below, [[ ]], is added to the end of Section 4.1.1.5 of the submitted LTR (Reference 1).

[[

]]

References

1. NEDE-33885P, "GNF CRDA Application Methodology," Revision 0, February 2018.

Table 1: Lattice Types of Consideration [[
]]

[[	
	]]

Table 2: Variation in Lattice Parameters [[
]]

[[	
	]]

[[

]]

Figure 1: [[ ]]

[[

]]

Figure 2: [[ ]]

[[

]]

Figure 3: [[ ]]

[[

]]

Figure 4: Doppler Coefficient Comparison [[

]]

NRC RAI #2

The TR indicates [[

]]. Figure 4-2 in the TR shows a peak pin enthalpy distribution that suggests that some of the fuel assemblies [[

]] may have peak enthalpy rises that approach that experienced by the fuel assemblies located next to the control rod. This may be especially true for core configurations where control rods have been withdrawn such that there are additional neutronic coupling considerations. Provide justification [[

]].

GNF Response

[[

]]

The methodology just described is a deliberate transformation [[

]]

It is true that neutronic coupling needs to be taken into consideration. This is achieved by means of [[

]]

A secondary consideration [[

]]

NRC RAI #3

[[ ]] Provide an extended discussion regarding the extent [[ ]] are validated as acceptable [[ ]]. Discuss how the area of applicability [[ ]].

GNF Response

There are [[ ]] parameters that define the area of applicability [[ ]]. Table 1 lists these parameters and describes how they are defined.

Table 1: GNF CRDA [[ ]] Applicability Checklist

[[	

[[	
	]]

The applicability [[

]] This checklist appears in Section 4.2.2 of the TR along with instructions for its use.

Discussion on Core Design and Operating Strategy

The CRDA evaluation that is performed [[ ]] accounts for variation in both core design and operating strategy by:

[[

]]

[[

]]

Furthermore, a plant's core design and operating strategy is constrained by the fuel product line definition [[ ]], as well as by the plant's Technical Specifications pertaining to reactivity control and power distribution limits.

[[

]]

Example Scenario [[

]]

The LTR is modified as follows:

4.2.2 [[ ]] Applicability Checklist

[[

]]

Table 4-5: GNF CRDA [[ ]] **Applicability Checklist**

[[		

[[		
		]]

References:

1. NEDE-33885P, "GNF CRDA Application Methodology," Revision 0, February 2018.

NRC RAI #4

Section 4.3.4 of the TR describes several options [[

]]

Describe why this requirement would be sufficient [[

]].

GNF Response

GNF has opted to [[

]] without introducing any new safety concerns.

[[

]]

The LTR is modified as follows.

[[

]]

NRC RAI 5

The guidance provided in the TR does not discuss any specific constraints or recommendations that may be necessary for the time step size or explain why the recommended inputs from prior NRC approved applications of TRACG are acceptable for the CRDA event. The CRDA event is a very rapid prompt power excursion that occurs on a very short time scale, so the predicted neutronic response due to heatup of the fuel and moderator may be sensitive to the time step size. Provide a discussion regarding what time step inputs are to be used for the CRDA analysis, and why these inputs are acceptable for this intended application.

GNF Response

TRACG automatically determines the time step size in an attempt to maximize the accuracy of the calculation and minimize the computer time. As described in Section 8.2.4 of Reference [1], two basic criteria are used for this purpose: convergence and Rate-Of-Change (ROC).

Solutions to the thermal-hydraulic equations are required to converge within a prescribed set of convergence criteria. The calculations are aborted if convergence cannot be achieved by decreasing the time step size. If convergence is obtained within a low number of iterations, the time step size is allowed to increase provided the ROC criteria regulating temporal discretization are also satisfied. Conversely if many iterations are required, the time step size is reduced. TRACG will reduce the time step size before failure of the outer iterations. This allows the time step controller to maximize the time step size and computational efficiency while preventing excessive backtracking because of failures to converge in the hydraulics solution.

TRACG examines the ROC for the primary dependent variables in all cells and all nodes of all components. If the maximum ROC is low a quasi-steady-state condition exists, and the time step size increases. Conversely if the ROC is high, the time step size is reduced. The ROC criteria serve to prevent excessive changes in each of the dependent variables during a time step and thereby prevent excessive temporal discretization errors. The ROC dependent variables are: (1) total pressure, (2) void fraction, (3) gas temperature, (4) liquid temperature, (5) total non-condensable gas pressure ("air"), (6) vessel slab temperature, and (7) fuel rod temperature. The eighth ROC ratio is a complex quantity from the 3D kinetics model [[

]]. The time step size is automatically reduced to accommodate any rapid ROCs [[ ]] as they occur in the control rod drop simulation(s). Because of this automatic feature, the calculated results are not sensitive to the minimum and maximum limits that are input to the code.

Two examples are provided using information taken from Section 5.1.3 of Reference [2] [[
]]. Time

step information is extracted from the TRACG cases that make up the Table 5-3 results. The minimum allowable time step size used for the TRACG cases is 1.0E-08 seconds and the maximum allowable time step size is 5.0E-03 seconds. The purpose of the maximum allowable value is to prevent the code from running at a pace that will not allow it to *see* the changing responses as they begin. This maximum limit comes into play primarily prior to the reactivity insertion and after the reactivity induced pulse has begun to settle back into a new quasi-steady condition.

Figure 1 presents the minimum and average time step size for each case compared to the minimum and maximum allowable time step size. The minimum time step size actually used was 8.07E-06 seconds which is roughly two orders of magnitude from the 1.0E-08 second minimum allowable time step size. The average time step size is approximately 3.76E-03 seconds. This example demonstrates that the specified lower limit on allowable time step size is sufficiently low that its value will not affect the results.

Figure 2 presents the individual time step sizes for the case with the highest overall ratio to either the pellet-cladding mechanical interaction or the high temperature cladding failure acceptance criteria in Table 5-3. [[

]] The minimum time step size actually used was 3.46E-05 seconds and the average time step size is approximately 2.40E-03 seconds. This example also demonstrates that the specified lower limit on allowable time step size is sufficiently low that its value will not affect the results.

References

[1] GE Hitachi Nuclear Energy, "TRACG Model Description," NEDE-32176P, Revision 4, January 2008.

[2] Global Nuclear Fuel, "GNF CRDA Application Methodology," NEDE-33885P, Revision 0, February 2018.

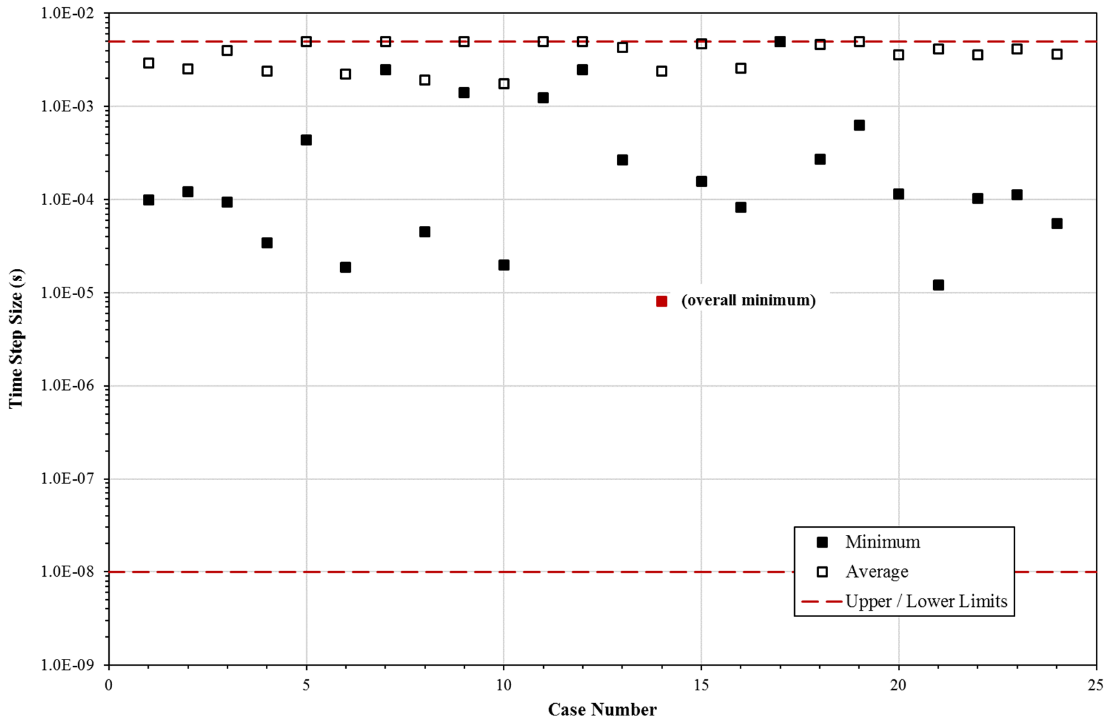

Figure 1: Time Step Sizes for Table 5-3 Cases

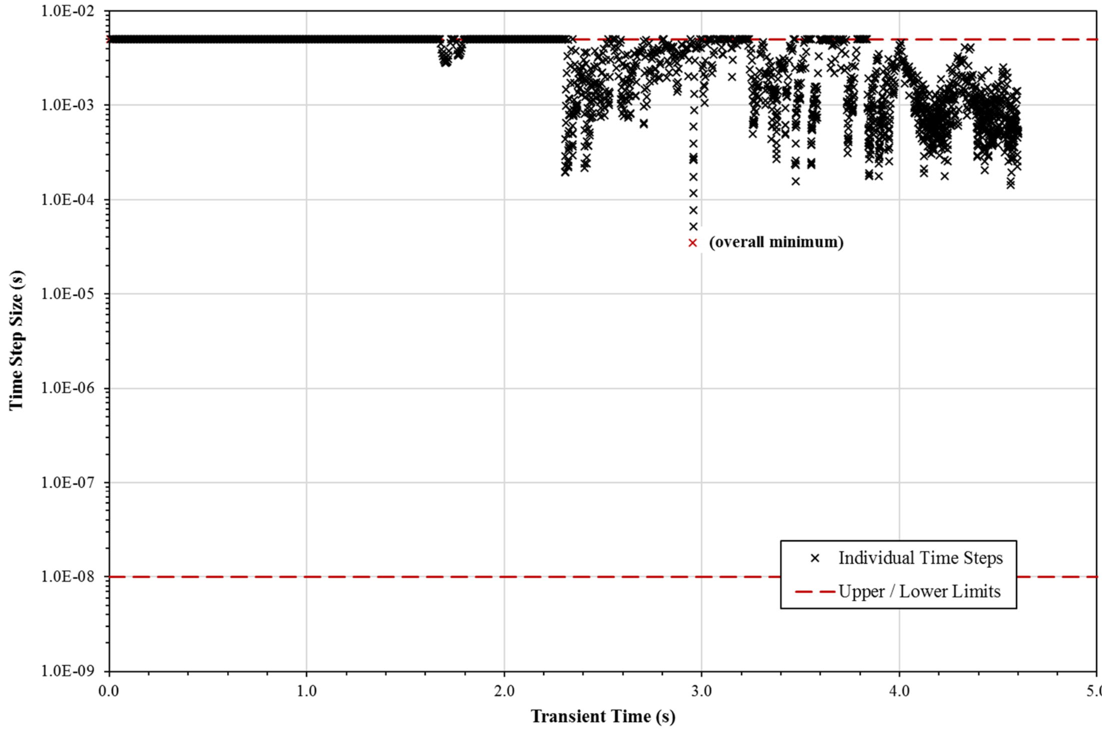

Figure 2: Time Step Sizes for Table 5-3 Limiting Case

NRC RAI #6

The guidance provided in the TR indicates that the control rods are to be modeled [[

]] provide justification [[

]].

GNF Response

When performing a calculation that models all the control rods [[
]]. This means that for any given sequence [[
]].

The following text is added to Section 4.3.3 of the LTR.

When performing a calculation that models all the control rods [[
]]. This means that for any given sequence [[
]].

NRC RAI-7

A significant characteristic of the expected limiting CRDA event is that it involves a prompt critical excursion. In addition to the worth of the dropped control rod, another critical quantity to define the magnitude of the prompt critical excursion is the delayed neutron fraction. Provide a discussion of the uncertainty in the delayed neutron fraction as calculated by the neutronics model in PANACEA/TRACG, and explain how it is accommodated by the proposed CRDA analysis methodology.

GNF Response

The total delayed neutron fraction (β) is calculated by PANACEA [[

]] Values of β depend primarily on exposure because of how the proportions of fissionable isotopes change with exposure. Initial lattice U235 enrichment also plays a less-significant roll. In addition to these two effects, the lattice calculations account for the changing neutron spectrum with exposure via the dependencies listed above. The capturing of the key elements influencing β together with the SPERT qualification and all the AOO transient qualification for TRACG using the PANAC 3D neutron kinetics supports the conclusion that there is no bias [[

]].

Uncertainty in β values are documented in Tables 1-13, 1-14 and 1-16 of Reference [7-1]. Text at the end of Section 1.4.1 of Reference [7-1] states the "more useful β values are given in Table 1-16" so that is the information used in this response. The first three rows of Table 7-3 of this response show the inputs that were used. Fission fractions for the key U235, U238, and Pu239 isotopes shown in columns 2, 3, and 4 were estimated using the correlations documented in Equations (9.3-14) through (9.3-16) of the TRACG Model Description Licensing Topical Report (LTR) (Reference [7-2]). Corresponding calculated weighted β values are shown in column 5. The 1-σ uncertainty values in column 6 were calculated as $\sigma = \sqrt{\sum_i w_i \sigma_i^2}$ where w_i is the fission fraction weighting fraction as given in columns 2, 3, and 4 of Table 7-3. The rightmost column (7) shows how the fractional uncertainty in the total value of β increases with exposure. This increase is caused by the increasing contribution of Pu239 for which the uncertainty fraction σ(β)/β is larger than for U235 and U238. Note that the weighted composite values for σ(β)/β can be larger than the values for individual isotopes because of how the weighted variances were used to calculate the composite value of σ(β).

Table 7-3 Determination of Uncertainty in Total [[ ]] β

[[

]]

The largest value of σ(β)/β=0.120 in the lower right corner of Table 7-3 will conservatively bound the uncertainty in total [[ ]] β regardless of [[ ]] exposures. Control Rod Drop Accident (CRDA) evaluations have demonstrated that the responses for fuel enthalpy rise, fuel absolute enthalpy, and peak cladding temperature (PCT) are [[ ]].

The TRACG code was used to evaluate the impacts of a 1-σ uncertainty of 12% in [[ ]] total β. These impacts were assessed for a representative CRDA calculation and include the impacts on calculated values for the reactor power, fuel rod enthalpies for the peak node of the peak rod, and PCT. [[

]]

Reactor total powers [[ ]] are shown in Figure 7-1. [[

]] When β is reduced the power pulse occurs slightly earlier, rises faster, and reaches a higher peak value. The key observation is that a larger power pulse associated with a reduced β also has a narrower pulse width so that the total energy of the pulse is essentially the same. This observation is consistent with what was observed for the SPERT qualification cases.

[[

]]

Figure 7-1 Total Reactor Power

The calculated fuel enthalpies for the peak rod in the peak node are shown in Figure 7-2. [[

]] The increase [[
]] is judged to be negligible [[
]].

[[

]]

Figure 7-2 Peak Hot Rod Enthalpy for Limiting Channel, Rod Group, Heated Node

The calculated PCT values are plotted in Figure 7-3. [[

]] This difference is judged to be negligible [[

]].

[[

]]

Figure 7-3 Peak Clad Temperature (from all rods in the simulated core)

The potential impacts of uncertainty in total β have been quantified. These impacts were assessed for a representative CRDA calculation and include the impacts on calculated values for the total reactor power, fuel rod enthalpy for the peak axial node of the peak rod, and PCT for all locations of all rods in the core. The insignificant impacts on these quantities supports the conclusion that the CRDA methodology described in the LTR to calculate the margin to the CRDA acceptance criteria that depend on these quantities is reasonably accurate [[
]].

References

[7-1] Reactor Physics Constants, ANL-5800, Argonne National Laboratory, July 1963.

[7-2] TRACG Model Description LTR, NEDE-32176P, Revision 4, January 2008.

NRC RAI 8

The enthalpy for the limiting rod in each fuel assembly of interest is computed [[ ]] enthalpies used to compare against the acceptance criteria are sensitive to uncertainties in the calculated rod and assembly powers [[ ]]. Provide a discussion of the uncertainty in the rod and assembly power distribution as calculated [[ ]].

GNF Response

Reference [1] established the nuclear modeling uncertainties in the Safety Limit Minimum Critical Power Ratio (SLMCPR) and Linear Heat Generation Rate (LHGR) calculations. The pin power uncertainties total [[ ]] and the overall bundle power uncertainty is [[ ]]. Consequently, the SLMCPR and LHGR calculations incorporate a [[ ]] uncertainty. [[ ]] This uncertainty is [[ ]].

As described in Sections 3.1.3.5 and 3.1.4.3 of Reference [2], [[ ]]. These thermal-mechanical limits are developed with [[ ]] uncertainty [[ ]], see Section 8.6 of Reference [3]. [[ ]] Thus, the uncertainty [[ ]]

References

[1] GE Nuclear Energy, "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC–32694P-A, Revision 0, August 1999.

[2] Global Nuclear Fuel, "GNF CRDA Application Methodology," NEDE-33885P, Revision 0, February 2018.

[3] Global Nuclear Fuel, "The PRIME Model for Analysis of Fuel Rod Thermal – Mechanical Performance Part 1 – Technical Bases," NEDC-33256P-A Revision 1, September 2010.