DG-1327 Comment Resolution Table

Comment Breakdown

Comment submissions received from 12 members of the public and industry with a total 124 comments.

DG-1327 PUBLIC COMMENTS

- General / Editorial
- Related Guidance
- Range of Aplicability
- Analytical Methods
- High Temp Failure Thrersholds
- PCMI Failue Thresholds
- Molten Fuel Failure Threshold
- Fission Product Release Fractions
- Allowable Dose Limits
- Allowable Pressure Limits
- Allowable Coolability Limits
- Implementation

Pie chart of DG-1327 Public Comments: General/Editorial 2%, Related Guidance 2%, Range of Applicability 11%, Analytical Methods 44%, High Temp Failure Thresholds 7%, PCMI Failure Thresholds 10%, Molten Fuel Failure Threshold 2%, Fission Product Release Fractions 10%, Allowable Dose Limits 2%, Allowable Pressure Limits 2%, Allowable Coolability Limits 2%, Implementation 6%.
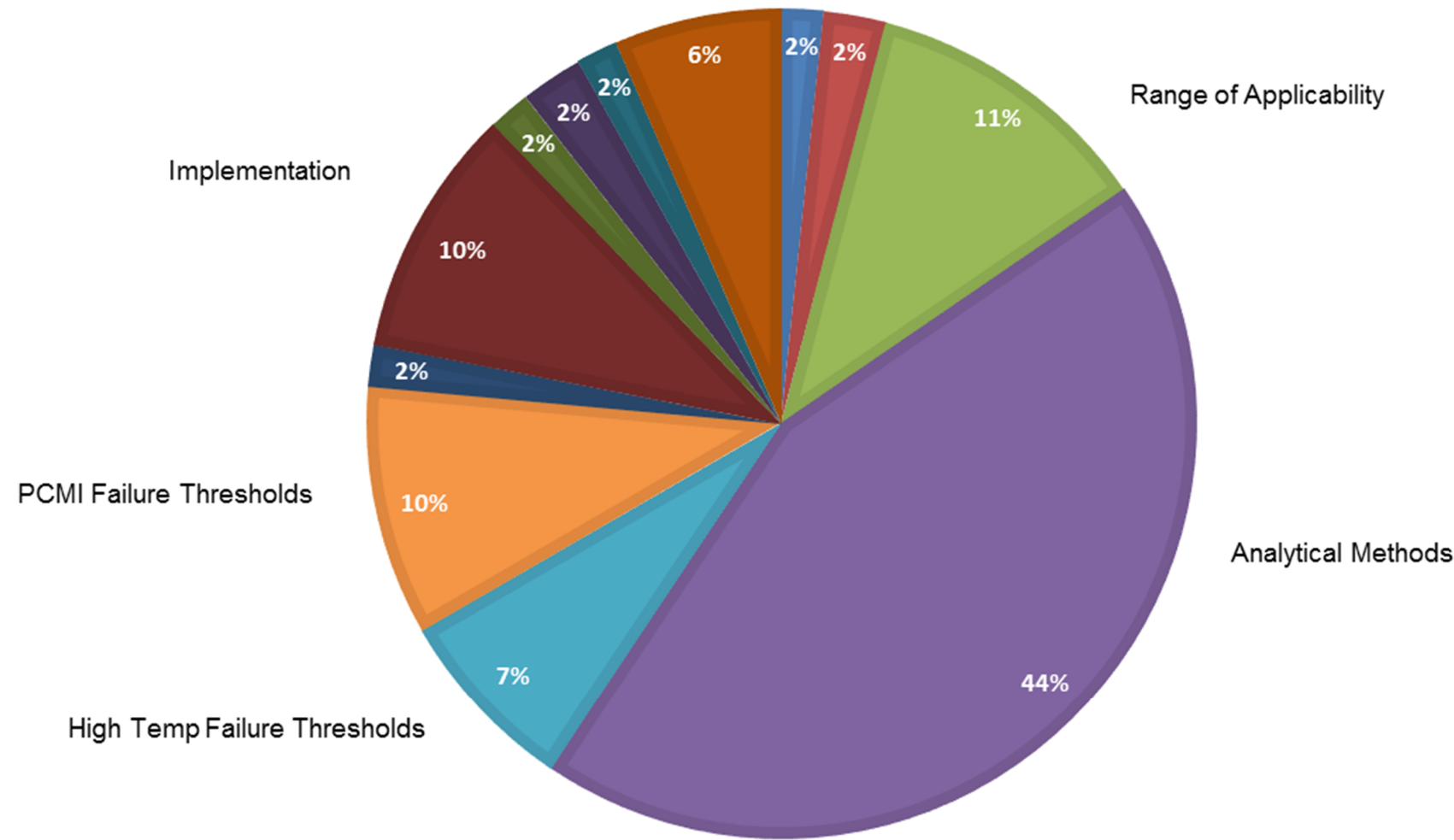
Comment Submissions*

1. NEI letter, "Request for Extension of Public Comment Period on Draft Regulatory Guide DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents' (Federal Register 81FR83288, dated November 21, 2016, Docket ID NRC-2016-0233)," January 12, 2017.

2. Public submission, Pedro Perez, "Comment on FR Doc # 2016-27903," January 25, 2017.

3. GEH letter, "Comments: Draft Regulatory Guide DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents'," M170067, March 23, 2017.

4. Public submission, Virgil C. Summer Nuclear Station Unit 1, "Comment on FR Doc# 2017-02073," April 14, 2017.

5. AREVA letter, "AREVA Inc. Response to Request for Public Comment on the Draft Regulatory Guide DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents' (Federal Register Vol. 81, No. 224, 83288, dated November 21, 2016; Docket ID NRC-2016-0233)," NRC:17:019, April 18, 2017.

6. Dominion letter, "Comments on DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents' (Docket ID NRC-2016-0233) (Federal Register Notice 81 FR 83288)," GL 16-019, April 20, 2017.

7. NEI letter, "Industry Comments on Draft Regulatory Guide DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents' (Federal Register 81FR83288, dated November 21, 2016 and 7590-01-P, dated January 26, 2017, Docket ID NRC-2016-0233)," April 21, 2017.

8. EPRI letter, "DRAFT REGULATORY GUIDE DG-1327, 'PRESSURIZED WATER REACOR CONTROL ROD EJECTION AND BOILING WATER REACTOR CONTROL ROD DROP ACCIDENTS'," FRP 2017-02, April 20, 2017.

9. SNC letter, "Comments on Draft Regulatory Guide DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents'," NL-17-0691, April 21, 2017.

10. Westinghouse letter, "Transmittal of Comments on DG-1327 [Docket ID NRC-2016-0233]," LTR-NRC-17-36, April 19, 2017.

11. NuScale Power letter, "NuScale Power, LLC Submittal of Comments on NRC DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents'," NRC-2016-0233, April 21, 2017.

12. APS letter, "Arizona Public Service (APS) Comments on Draft Regulatory Guide DG- 1327, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, Federal Register 81 FR 83288; Docket ID NRC-2016-0233," 102-07494-TNW/MDD, April 21, 2017.

13. Exelon letter, "Comments on Draft Regulatory Guide (RG) DG-1327, 'Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents' (Federal Register 81 FR83288, dated November 21, 2016, Docket ID NRC-2016-0233)," April 20, 2017.

* All 13 comment submissions received were annotated and compiled in ADAMS ML XXXXXXX

TOPIC: **General and Editorial Comments**

DG-1327 Draft Text:

N/A

Number of Comments: 1

Comment:	NRC Response:
a) To facilitate common understanding and provide a consistent interpretation of the guidance, a section devoted to nomenclature and definitions is suggested. [A-5]	a) The NRC agrees with the comment. Appendix A will be added to define acronyms and additional clarification will be added to the text.

Resolution / Path Forward / Action Item:
Appendix A added to define acronyms. Additional clarification added throughout, especially the definitions proposed by the commenter.

Revised RG Text:
See Appendix A.

TOPIC: **Purpose**

DG-1327 Draft Text:

Purpose

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when analyzing a postulated control rod ejection (CRE) accident for pressurized-water reactors (PWRs) and a postulated control rod drop (CRD) accident for boiling-water reactors (BWRs). It defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet-clad mechanical interaction (PCMI) and provides radionuclide release fractions for use in assessing radiological consequences. It also describes analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits.

Number of Comments: 1

Comment:	NRC Response:
a) The guidance does not provide specific radionuclide release fractions, but rather an algorithm to use in calculating radionuclide release fractions. [A-25]	a) The NRC agrees with the comment. The proposed language will be incorporated.

Resolution / Path Forward / Action Item:
Revised text.

Revised RG Text:
… It defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet-clad mechanical interaction (PCMI) and provides an algorithm for calculating radionuclide release fractions for use in assessing radiological consequences. …
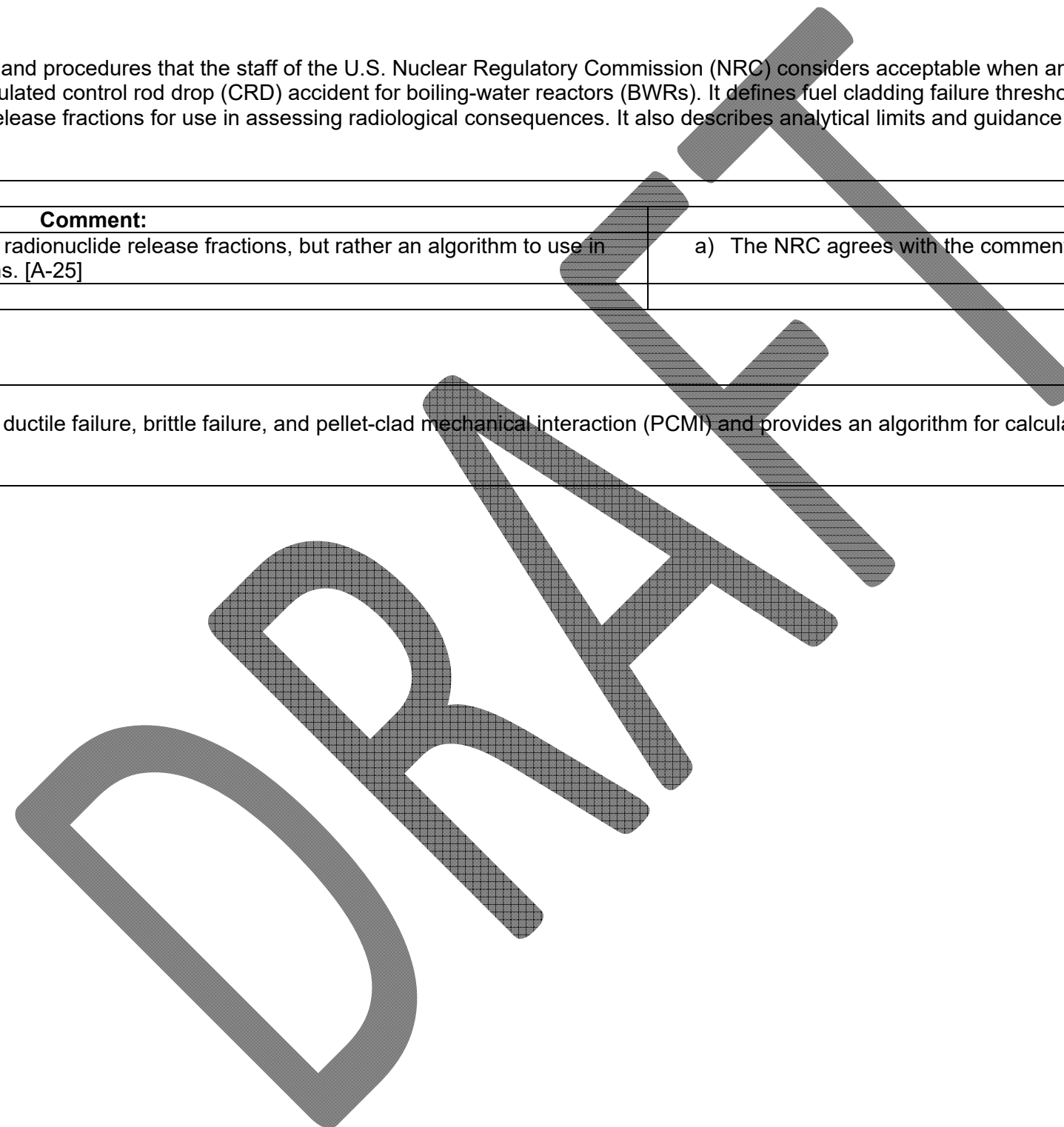
TOPIC: **Related Guidance**

DG-1327 Draft Text:

Related Guidance
- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP), (Ref. 3) provides guidance to the NRC staff for review of safety analysis reports submitted as part of license applications for nuclear power plants.
    - SRP Section 15.4.8 provides guidance to the NRC staff for reviewing PWR CRE accidents.
    - SRP Section 15.4.9 provides guidance to the NRC staff for reviewing BWR CRD accidents.
    - SRP Section 4.2 provides guidance to the NRC staff for reviewing reactor fuel designs.
    - SRP Section 4.2, Appendix B provides guidance to the NRC staff in reviewing both PWR CRE and BWR CRD accidents.
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (Ref. 4) provides guidance for calculating radiological consequences for design basis accidents.
- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," (Ref. 5) provides guidance for calculating radiological consequences for design-basis accidents.

Number of Comments: 3

Comment:	NRC Response:
a) Add RG 1.224 to the Related Guidance if it continues to be referenced. If it is not continued as a reference then disregard this comment. [NEI-A21]	a) The NRC agrees with this comment. The preliminary draft RG 1.224 will be added.
b) Section A; Related Guidance (pg. 2): It appears that RG 1.77 should be included in the Related Guidance section. DG-1327 should clarify its relationship to RG 1.77 (replacement of RG 1.77, supplement to RG 1.77, or no relationship). [APS-1]	b) The NRC agrees with this comment. RG 1.77 will be added to this section and the text will clearly identify that this guidance document replaces RG 1.77.
c) Listing of guidance is incomplete and should include RG 1.77 and NUREG-1465. [A-26]	c) The NRC agrees in part with the comment. RG 1.77 will be added as related guidance. NUREG-1465 addresses accidents involving severe core damage. Core damage is limited under the guidance within this document.

Resolution / Path Forward / Action Item:
Text added.

Revised RG Text:
- RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," (Ref. 6) provides guidance for evaluating PWR CRE.
- Preliminary draft RG 1.224, "Establishing Analytical Limits for Zirconium-Alloy Cladding Material," (Ref. 7) includes guidance for estimating cladding hydrogen content.
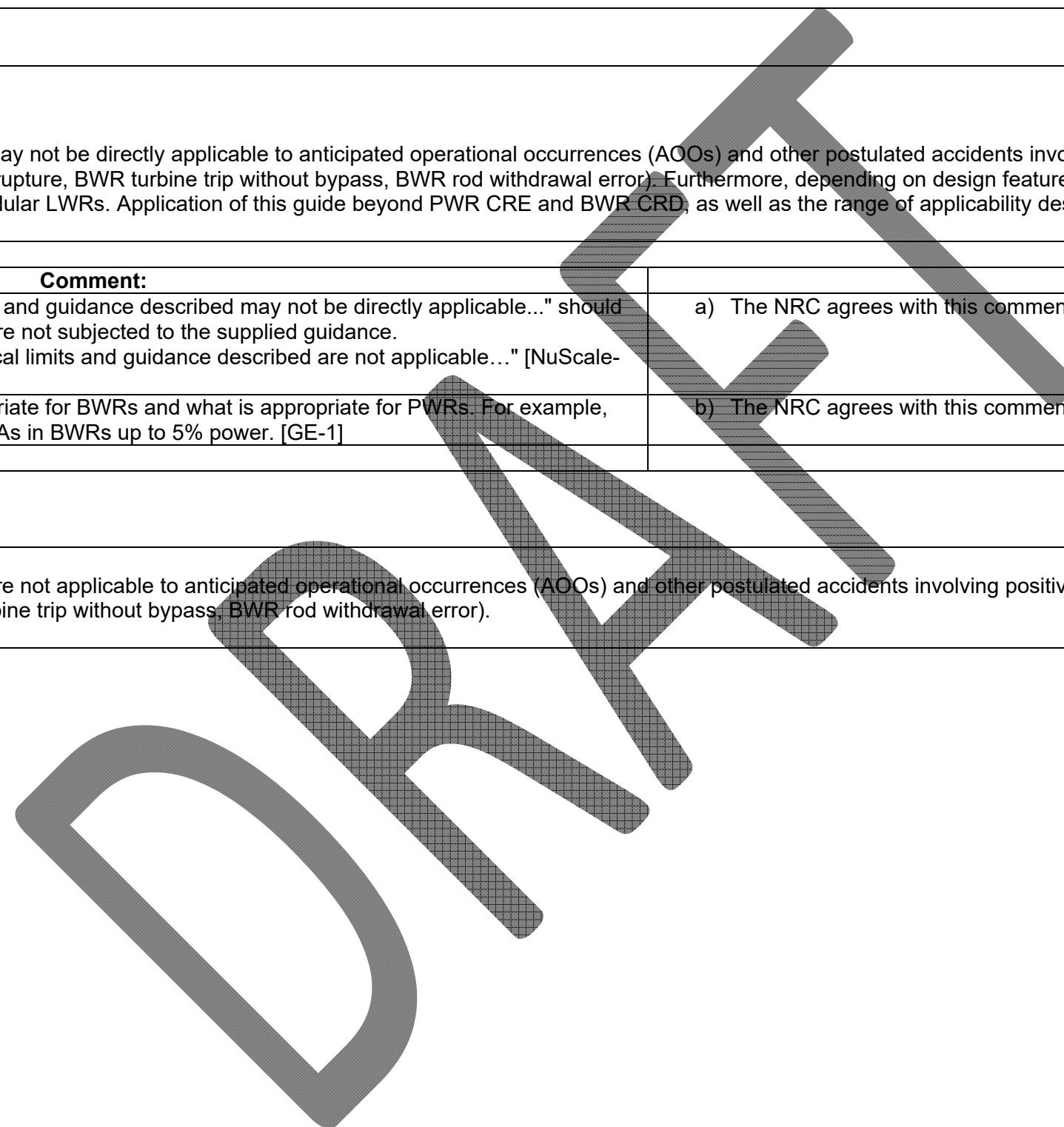
Reason for Issuance

This guide incorporates empirical data from in-pile, prompt power pulse test programs and analyses from several international publications on fuel rod performance under prompt power excursion conditions to provide guidance on acceptable analytical methods, assumptions, and limits for evaluating a postulated PWR CRE and a postulated BWR CRD accident. This guide replaces the PWR CRE analytical guidance provided RG 1.77.

TOPIC: **Applicability**

DG-1327 Draft Text:

Applicability
This guide applies to applicants and reactor licensees subject to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1) and 10 CFR Part 52 "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Limits on Applicability**

DG-1327 Draft Text:

C.1. Limits on Applicability
The analytical limits and guidance described may not be directly applicable to anticipated operational occurrences (AOOs) and other postulated accidents involving positive reactivity insertion (e.g., PWR excess load, PWR inadvertent bank withdrawal, PWR steam line rupture, BWR turbine trip without bypass, BWR rod withdrawal error). Furthermore, depending on design features, reactor kinetics, and accident progression, this guide may not be directly applicable to advanced LWRs and modular LWRs. Application of this guide beyond PWR CRE and BWR CRD, as well as the range of applicability described below, will be considered on a case-by-case basis.

Number of Comments: 2

Comment:	NRC Response:
a) The first sentence "The analytical limits and guidance described may not be directly applicable..." should be more definitive to state that AOOs are not subjected to the supplied guidance. Change sentence to read, "The analytical limits and guidance described are not applicable…" [NuScale-1]	a) The NRC agrees with this comment. Sentence will be revised to be more definitive.
b) The DG needs to clarify what is appropriate for BWRs and what is appropriate for PWRs. For example, this DG should only be applicable to RIAs in BWRs up to 5% power. [GE-1]	b) The NRC agrees with this comment. PWR/BWRs will be separated where appropriate.

Resolution / Path Forward / Action Item:
Revise text.

Revised RG Text:
The analytical limits and guidance described are not applicable to anticipated operational occurrences (AOOs) and other postulated accidents involving positive reactivity insertion (e.g., PWR excess load, PWR inadvertent bank withdrawal, PWR steam line rupture, BWR turbine trip without bypass, BWR rod withdrawal error).

TOPIC: **Limits on Applicability**

DG-1327 Draft Text:

C.1.1   LWR fuel rod designs comprised of slightly enriched UO2 ceramic pellets (up to 5.0 wt% 235U) within cylindrical zirconium-based cladding, including designs with or without barrier lined cladding, integral fuel burnable absorber (e.g., gadolinium), or a pellet central annulus. This guidance is not applicable to mixed oxide (MOX) fuel rod designs.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Limits on Applicability**

DG-1327 Draft Text:

C.1.2 The high temperature cladding failure threshold described in Section 3.1 is applicable to reactor startup, zero power, and low power operations (i.e., < 5% rated power) and covers the entire initial reactor coolant temperature range (i.e., room temperature to operating temperatures). For all other operating conditions up to full power (i.e., Mode 1), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

Number of Comments: 9

Comment:	NRC Response:
a) There may not be meaningful thermal limits below the Tech Spec power monitoring level. Licensed DNBR/CPR correlations may not have been developed to address the range of applicability consistent between 5% and TS monitoring required power levels. [NEI-A2]	a) The NRC agrees with this comment. The licensed DNBR/CPR correlations developed to estimate thermal margins at steady-state operating conditions may not be applicable to transient conditions for events initiated at lower power levels. Similar to other transient analyses (e.g., PWR return-to-power steam line break), the applicant must justify the existing correlation or propose an alternative correlation.
b) The limits in Items 1.2 and 3.1 should both be made consistent with the description of the fuel rod cladding failure thresholds in Section 3, specifically with the following text: "During a prompt critical reactivity insertion (i.e., Δρ/βeff > 1.0), fuel temperatures may approach melting temperatures, and rapid fuel pellet thermal expansion may promote PCMI cladding failure. During more benign power excursions, local heat flux may exceed critical heat flux conditions, prompting fuel cladding temperatures to rise." [W-1, SNC-1]	b) The NRC agrees with this comment. Section C.1.2 is repetitive, and not entirely consistent with Section 3.1 and therefore will be deleted.
c) There is no evidence of any fuel rod cladding failure due solely to the local heat flux exceeding the thermal design limit (e.g., departure from nucleate boiling and critical power ratios) for a prompt critical reactivity insertion. [W-2, SNC-1]	c) The NRC disagrees with this comment. High temperature cladding failures were observed at several prompt test programs. This is the bases of the cladding failure thresholds in Section 3.1. The NRC staff acknowledges that time-in-DNB (or boiling transition) is necessary for cladding failure and that prompt critical accidents will likely only momentarily experience DNB conditions. In response to a different comment, the staff has added guidance allowing an alternative failure threshold. See below.
d) Requested change: The high temperature cladding failure threshold described in Section 3.1 is applicable to ~~reactor startup, zero power, and low power operations (i.e., < 5% rated power)~~ prompt-critical reactivity insertions and covers the entire initial reactor coolant temperature range (i.e., room temperature to operating temperatures). For ~~all other operating conditions up to full power (i.e., Mode 1)~~ at-power (i.e., > 5% rated power up to full power) non-prompt critical power excursions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios). [W-3, D-2]	d) The NRC disagrees with this comment. The proposed text does not cover a scenario involving a non-prompt power excursion initiated from below 5% power. For these events where the initial heat flux is extremely low, use of the at-power thermal design limits becomes questionable. Application of Figure 1 is appropriate for these cases, and the calculated fuel enthalpy will reflect the actual power excursion (prompt or non-prompt).
e) For the CRDA (BWR) and CRE (PWR) events the fuel cladding failure criteria is specified in this DG. Acceptance criteria for other events is extraneous and should be removed from this DG. Remove the sentence: "For all other operating conditions up to full power (i.e., Mode 1), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios)." Note boiling transition is not a presumed failure mechanism for BWR fuel. [GE-2]	e) The NRC agrees with this comment. Section C.1.2 was deleted in response to comment (b) above.
f) It has been demonstrated by several tests that DNB/CPR does not systematically lead to rod failure. This is specifically mentioned in NUREG 0800 Section 4.2 and alternate cladding failure criteria are permitted with adequate technical justification. This text could be interpreted to preclude alternate cladding failure criteria. This would contradict the guidance provided in NUREG-0800 Section 4.2. [A-6]	f) The NRC agrees with this comment. The proposed text allowing an alternate approach will be added.

Resolution / Path Forward / Action Item:
Section C.1.2 deleted.

Revised RG Text:
Alternative fuel rod cladding failure criteria may be used if they are adequately justified by analytical methods and supported by sufficient experimental data. Alternative cladding failure criteria will be addressed on a case-by-case basis.

TOPIC: **Limits on Applicability**

DG-1327 Draft Text:

C.1.3 As described in Section 3.2, separate PCMI cladding failure thresholds are provided for different initial reactor coolant temperatures and different cladding thermal annealing treatments. The high temperature PCMI cladding failure threshold curves are applicable to reactor coolant temperatures at or above 500 °F. Below 500 °F, the low temperature PCMI cladding failure threshold curves are applicable. The fully recrystallized annealed (RXA) PCMI cladding failure threshold curves are applicable to cladding which has undergone thermal treatment to remove all residual stresses and is in an RXA state. For all other stages of thermal treatments, the stress relief annealed (SRA) PCMI cladding failure threshold curves are applicable.

Number of Comments: 3

Comment:	NRC Response:
a) Section C.1.3 states to use RXA cladding failure thresholds only for cladding that is in an RXA state, and to use the SRA cladding failure thresholds otherwise. Section C.2.3.5.1 states use RXA if >10% of hydrides are radial, otherwise use SRA. This guidance is inconsistent. [NEI-A18]	a) The NRC agrees with this comment. Text revised to remove inconsistency.
b) No technical basis is provided to support using SRA PCMI failure threshold curves for the entire range of metallurgical conditions between fully RXA and SRA. Recommended Change: "The recrystallization annealed (RXA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces RXA metallurgical state, while the stress relief annealed (SRA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces SRA metallurgical state. For any other metallurgical condition, the applicant should provide justification for similarity with either SRA or RXA metallurgical condition." [A-1]	b) The NRC agrees with this comment. Proposed text adopted.
c) 500°F is above the ductile-to-brittle transition temperature (DBTT) that was determined by mechanical tests on both SRA and RXA materials in the as-irradiated condition and thus leads to undue conservatism. Basis for concern: An engineering solution for BWR CRDA is to keep coolant temperature above the DBTT during "cold" shutdown state and the 500°F limit may impose unnecessary economic penalties. [A-2]	c) The NRC disagrees with this comment. This guidance does not define a minimal measure of cladding ductility such as a DBTT. But instead, addresses the changing degrees of ductility necessary to avoid cladding failure as a function of increasing fuel enthalpy (and associated pellet thermal expansion). Since zirconium hydrides have a dominant effect on cladding ductility, the cladding failure threshold is provided as a function of excess hydrogen. The NRC's investigation found that the impact of initial cladding temperature on PCMI failure threshold was only 18 cal/g between cold (room temperature) testing and hot (above 500°F) testing. The NRC would consider, on a case-by-case basis, further scaling between 500°F and a lower temperature (corresponding to plant-specific BWR startup conditions).

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.1.3 As described in Section 3.2, separate PCMI cladding failure thresholds are provided for different initial reactor coolant temperatures and different cladding thermal annealing treatments.

1.3.1   The high temperature PCMI cladding failure threshold curves are applicable to reactor coolant temperatures at or above 500 ˚F. Below 500 ˚F, the low temperature PCMI cladding failure threshold curves are applicable.

1.3.2   The recrystallization annealed (RXA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces RXA metallurgical state, while the stress relief annealed (SRA) PCMI cladding failure threshold curves are applicable to cladding which has undergone final thermal treatment that produces SRA metallurgical state. For any other metallurgical condition, the applicant should provide justification for similarity with either SRA or RXA metallurgical condition.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.1.1 Accident analyses should be performed using NRC approved analytical models and application methodologies that account for calculational uncertainties. The analytical models and computer codes used should be documented and justified, and the conservatism of the models and codes should be evaluated both by comparison with experiment and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

Number of Comments: 11

Comment:	NRC Response:
a) The cladding failure thresholds are conservative since they are a lower bound on the failure data. The details regarding uncertainties are not applicable. Furthermore, improbable events have historically been licensed using best estimate nominal calculations. [NEI-A7, GE-3]	a) The NRC disagrees with this comment. The PCMI cladding failure thresholds are a best-fit to the reported fuel enthalpy values. No additional conservatism nor application of experimental uncertainties was applied to the development of the failure curves. Analytical uncertainties need to be considered, either deterministically or statistically.
b) RG 1.203 is not mentioned in DG-1327. To clarify non-applicability of RG 1.203 some clarification should be added. [NEI-A9]	b) The NRC disagrees with this comment. The applicability and utilization of RG 1.203 to a particular vendor's methods are beyond the scope of this RG.
c) Subjects discussed in DG-1327 include PWR-specific and BWR-specific topics. Separating or designating PWR-specific and BWR-specific content would avoid confusion. [NEI-A10, D-4]	c) The NRC agrees with this comment. BWR/PWR specific topics will be separated as appropriate.
d) There should not be a requirement to compare to more sophisticated spatial kinetics codes if the submitted methodology includes a 3D spatial kinetics code. [NEI-A15].	d) The NRC agrees in part with this comment. Credit for comparison in an approved 3D kinetics code may be used and justified as part of the analysis. The important part is that justification is also required as comparison/benchmarks that were done as part the approval for 3D kinetics code may or may not be sufficient for RIA.
e) Requirement to compare to more sophisticated codes needs clarification. States that methods must be compared to more sophisticated codes, but if the method uses a sophisticated code this would be unnecessary and difficult. Consider rewording to state that if point-kinetics or 1-D kinetics are the methodology, then this comparison is required. [APS-2]	e) See comment (d).
f) In the past, a suite of codes may have been used to perform different aspects of the analysis. Uncertainty for discrete aspects may have been quantified, but often, the overall approach was deemed "conservative enough." Exelon believes that further clarification is needed to determine whether it will be sufficient to qualitatively state, "conservative enough" or if something additional and specific will be required. [Exelon-1]	f) The NRC disagrees with this comment. This is inherent to the review and approval process for any method. DG are meant to provide guidance, not a 100% boiler plate methodology approach to the analysis
g) More "sophisticated spatial kinetics codes" may exist today, but may not be available for a particular plant/vendor combination. Exelon is requesting further clarification as to whether a vendor will be required to develop the more sophisticated codes and methods. [Exelon-2]	g) The NRC does not believe that vendors will need to develop new methods for application to specific plant designs as the majority of the analyses key variables are fuel dependent and not plant dependent. The plant specific inputs are not detailed enough or nuanced that would prevent easy justification of a vendor's methodology to an existing licensee. New plant designs may require new methods development. It is possible that the vendors may need to develop new methods to meet the new, lower acceptance criteria
h) A footnote should be added to the guidance provided in Item 2, Analytical Methods and Assumptions, of Section C, Staff Regulatory Guidance, to indicate that the analytical inputs, assumptions, and methods described herein are specific and sufficient for analyzing postulated reactivity-initiated accidents and that RG 1.203 need not be applied when this regulatory guidance is employed. [D-3]	h) See comment (a).
i) The majority of experimental data ·relates to thermal mechanics and there are almost no RIA experiment dedicated to neutronic aspects. International benchmarks between codes being at the state of the art should be an alternative to comparisons to experiment. "More sophisticated spatial kinetics codes" may not necessarily be available. [A-7]	i) The NRC agrees with this comment and clarification that code-to-code comparison may be acceptable alternative to direct benchmarks and visa versa.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.1.1　　　Accident analyses should be performed using NRC approved analytical models and application methodologies. The analytical models and computer codes used should be documented and justified, and the conservatism of the models and codes should be evaluated. Comparison with experiment and/or with more sophisticated spatial kinetics codes should be performed. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

When performing statistically based accident analyses, analytical uncertainties should be quantified and their application fully justified.
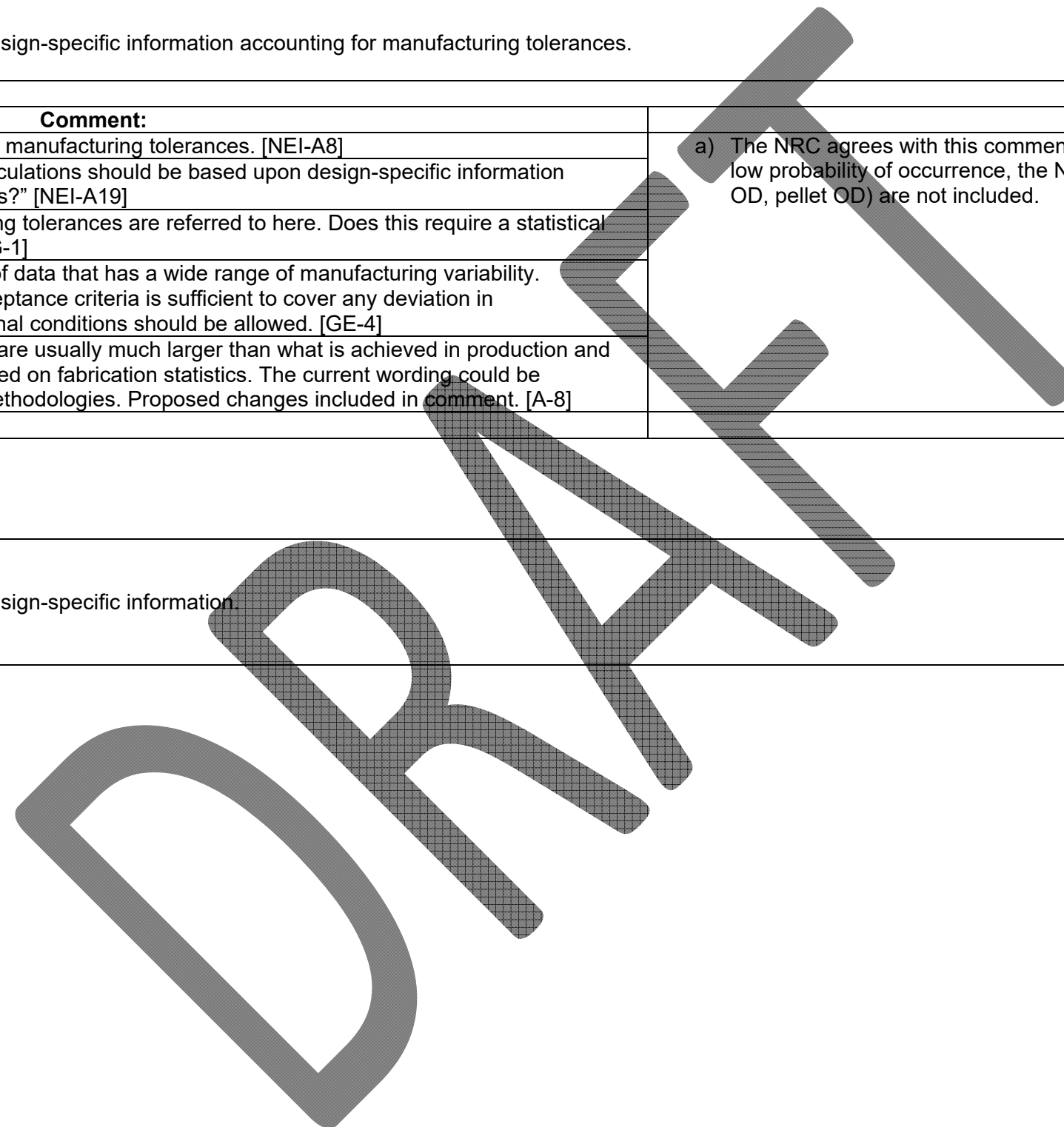
TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.1.2 The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (a) incorporation of all major reactivity feedback mechanisms, (b) at least six delayed neutron groups, (c) both axial and radial segmentation of the fuel element, (d) coolant flow provision, and (e) control rod scram initiation.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:
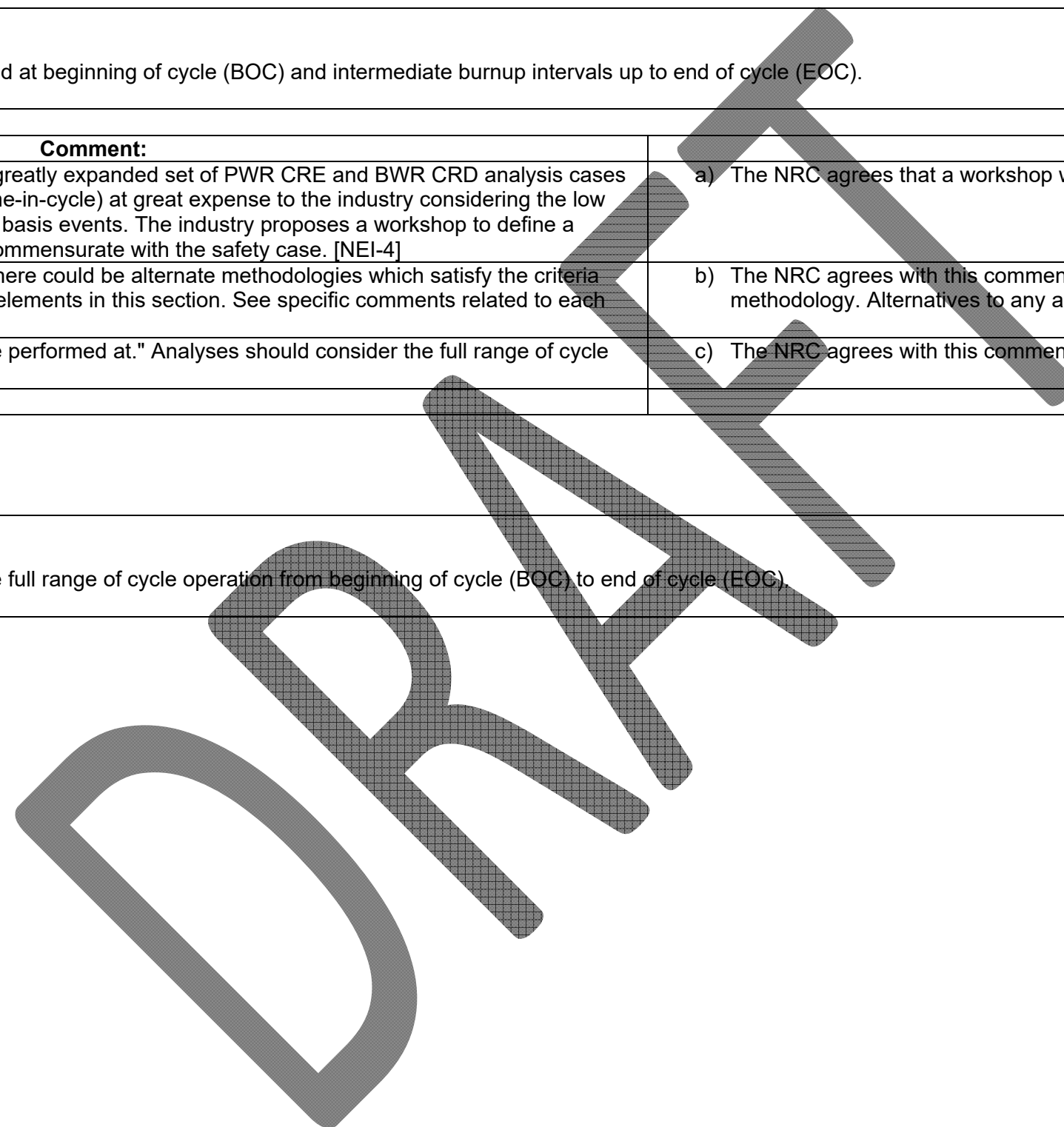
Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.1.3 Calculations should be based upon design-specific information accounting for manufacturing tolerances.

Number of Comments: 5

Comment:	NRC Response:
a) Calculations should not need to include manufacturing tolerances. [NEI-A8] b) What is intended by the statement "Calculations should be based upon design-specific information accounting for manufacturing tolerances?" [NEI-A19] c) Please clarify what kind of manufacturing tolerances are referred to here. Does this require a statistical analysis with 95/95 uncertainty? [SCEG-1] d) The failure threshold is a lower bound of data that has a wide range of manufacturing variability. Therefore, use of the conservative acceptance criteria is sufficient to cover any deviation in manufacturing. Evaluations using nominal conditions should be allowed. [GE-4] e) Manufacturing specification tolerances are usually much larger than what is achieved in production and realistic methodologies use ranges based on fabrication statistics. The current wording could be interpreted to prohibit these types of methodologies. Proposed changes included in comment. [A-8]	a) The NRC agrees with this comment. Given the level of conservatism in the analytical methods and the low probability of occurrence, the NRC finds it acceptable that manufacturing tolerances (e.g., cladding OD, pellet OD) are not included.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.1.3 Calculations should be based upon design-specific information.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.1.4 Burnup-related effects on reactor kinetics (e.g., β_{eff}, l*, rod worth, Doppler effect) and fuel performance (e.g., pellet radial power distribution, fuel thermal conductivity, fuel-clad gap conductivity, fuel melting temperature) should be accounted for in fuel enthalpy calculations.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**	
DG-1327 Draft Text: C.2.2.1 Accident analyses should be performed at beginning of cycle (BOC) and intermediate burnup intervals up to end of cycle (EOC).	
Number of Comments: 3	
Comment:	**NRC Response:**
a) The proposed guidance will result in a greatly expanded set of PWR CRE and BWR CRD analysis cases (i.e. range of initial power levels and time-in-cycle) at great expense to the industry considering the low probability of these hypothetical design basis events. The industry proposes a workshop to define a sufficient and consistent set of cases commensurate with the safety case. [NEI-4]	a) The NRC agrees that a workshop would help identify the appropriate level of analytical detail.
b) The guidance is very prescriptive and there could be alternate methodologies which satisfy the criteria that do not include all of these specific elements in this section. See specific comments related to each sub section of 2.2. [GE-5]	b) The NRC agrees with this comment. It is not the intent of this RG to cover the nuances of each vendor's methodology. Alternatives to any and all sections of a RG are acceptable if adequately justified.
c) Remove the words "analyses should be performed at." Analyses should consider the full range of cycle operation from BOC to EOC. [GE-6]	c) The NRC agrees with this comment. Text revised.
Resolution / Path Forward / Action Item: Text revised.	
Revised RG Text: C.2.2.1 Accident analyses should consider the full range of cycle operation from beginning of cycle (BOC) to end of cycle (EOC).	

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.2 Accident analyses at cold zero power (CZP) and hot zero power (HZP) conditions should encompass both (1) BOC following core reload and (2) re-start following recent power operation.

Number of Comments: 2

Comment:	NRC Response:
a) This section discusses accident analyses at Cold Zero Power (CZP) and Hot Zero Power (HZP) conditions. Exelon suggests that the NRC clarify that CZP is required for BWRs only. [Exelon-5]	a) The NRC agrees with this comment. The analytical methods section will be divided between PWR/BWR specifics as appropriate.
b) PWR operation at CZP is not possible. This is only relevant for BWRs. The current wording could be misinterpreted to require PWR analysis at CZP conditions. Proposed changes included in comment. [A-9]	

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.3 Accident analyses should be performed at intermediate power levels up to hot full power (HFP) conditions. These calculations should confirm power-dependent core operating limits (e.g., control rod insertion limits, rod power peaking limits, axial and azimuthal power distribution limits). At lower-power conditions where certain core operating limits do not apply, the analysis must consider the potential for wider operating conditions due to xenon oscillations or plant maneuvering.

Number of Comments: 7

Comment:	NRC Response:
a) Initial conditions for the BWR CRDA are from CZP to some low power level. Above a low power level, the transient response to a dropped rod is non-limiting due to void reactivity feedback.[NEI-A4]	a) The NRC disagrees with this comment. It is the responsibility of the applicant to demonstrate that CRD scenarios initiated at-power are not limiting.
b) Guidance for initial conditions requires clarification to avoid inconsistencies with ANSI/ANS N51.1 and N52.1. [NEI-A25]	b) See response to comment (c).
c) The guidance provided for selection of initial conditions for analysis requires clarification, to prevent misinterpretations that may inadvertently create new or modified nuclear safety criteria for the design of stationary light water reactors. The current criteria for many operating plants are based on ANSI/ANS standards such as N51.1 (formerly N18.2) for PWRs and N52.1 (formerly N212) for BWRs. These standards define conditions for design (e.g., limiting faults) and address the selection of initial conditions for safety analyses. Although these standards support the consideration of a wide range of possible initial conditions, they do not require the deterministic treatment of every possible set of initial conditions. For example, a plant would not necessarily be required to postulate an event combination involving a limiting fault CRE/CRD, while recovering from a transient caused by an independent moderate frequency event such as an inadvertent control rod withdrawal or drop (which may or may not have forced that plant into a lower mode of operation). Thus many safety analyses are initiated from steady-state or quasi-steady-state conditions, rather than transient initial conditions involving certain plant maneuvers or xenon oscillations. [APS-3]	c) The NRC agrees in part with this comment. It is not the intent to require licensees to analyze an event combination involving a limiting fault CRE/CRD, while recovering from a transient caused by an independent moderate frequency event such as an inadvertent control rod withdrawal or drop (which may or may not have forced that plant into a lower mode of operation). However, power plants are allowed to maneuver and have defined allowable operating limits (e.g., axial power distribution). The analysis needs to encompass the entire allowable range of initial conditions within those allowable operating limits. At lower power conditions where control room alarms on core operating limits may not exist, the applicant must justify the initial conditions with proper consideration for plant maneuvering.
d) The combination of initial conditions for a CRE or CRD event from Items 2.2.1, 2.2.2 and 2.2.3 lead to a large matrix of cases needing to be evaluated. The current guidance of Items 2.2.1, 2.2.2 and 2.2.3 could lead one to consider CZP, HZP, 5%, 10%, … HFP conditions at each burnup step (BOC, 2000 MWD/MTU, 4000 MWD/MTU, …, EOC) for a cycle design. The guidance is unclear as to the need for the expanded case matrix over the current evaluated case matrix for a PWR plant. Additional guidance is needed to understand the need for the expanded or the combination of conditions over which the NRC is concerned. [APS-4]	d) The NRC disagrees with this comment. TS/COLR define power-dependent core operating limits which require a license bases in safety analysis. Furthermore, PWRs are allowed deeper control rod insertion and wider operating ranges at powers levels below HFP. Hence, CRE events initiated from these conditions may be more limiting.
e) Given that a large majority of the time each reactor spends at power is near 100%, can low power conditions be excluded from the analysis? Many transient analyses are perfoll1ed at zero power .and full power based on probability. It would be very time-consuming to determine if intermediate power levels are more limiting at each bumup interval. It would seem that even for a load-following plant, examinations of 0, 80%, 90%, and 100% would be sufficient to cover 99% of the probability distribution. [SCEG-2]	
f) This section should be removed or the DG should state that this is applicable only to PWRs. The phrase "at intermediate power levels up to hot full power (HFP) conditions" is not applicable to BWR CRDA analyses. For BWRs the only applicable conditions are at startup and zero power up to 5%. [GE-7]	f) See comment (a).
g) The use of the word "confirm" in relation to the power dependent operating limits could be misinterpreted. Proposed changes included in comment. [A-10]	g) The NRC agrees with this comment. Text revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.3 Accident analyses should consider the full range of power operation up to hot full power (HFP) conditions. These calculations should consider power-dependent core operating limits (e.g., control rod insertion limits, rod power peaking limits, axial and azimuthal power distribution limits). At lower-power conditions where certain core operating limits do not apply, the analysis must consider the potential for wider operating conditions due to xenon oscillations or plant maneuvering.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.4 Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on damaged core coolability during fuel rod lifetime, the limiting initial conditions may involve the uncontrolled movement of lower-worth control rods or partially inserted control rods (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). As such, a more comprehensive search for the limiting conditions may be necessary to ensure that the total number of fuel rod failures is not underestimated and allowable limits are satisfied. Applicants may need to survey a larger population of BWR blade drop and PWR ejected rod core locations and exposure points to identify the limiting scenarios.

Number of Comments: 2

Comment:	NRC Response:
a) Exelon is unsure that a bounding type analysis can be successful, or how it can be performed. Section 2.2.4 seems to imply that burnup/corrosion/failure threshold/rod worth combinations need to be exhaustively searched. Exelon believes that further clarification is necessary to provide some assurance that fuel management changes do not result in a combination that was not previously evaluated. Due to the event being an accident, and due to what appears to be an expensive analysis to perform, Exelon believes that there needs to be a way to perform an acceptable analysis once that covers the extremes. Perhaps some objectives to consider might be: "How much fuel can be failed, and can more fuel be failed than that with a CEA ejection event?" and "Is a coolable geometry (230 cal/gm) maintained?" [Exelon-6]	a) The NRC agrees with this comment. Text has been added to describe a bounding approach to simplify analysis. With proper justification, a licensee may be able to demonstrate that given the localization of the power excursion within the core that the number of fuel rod cladding failures will remain below that assumed in the dose calculations. The licensee would also define a maximum reload ejected rod worth which maintains fuel enthalpy below the damaged core coolability limit.
b) There are several clarifications needed in these two subsections. They are bulleted: • The words "expected and unexpected" are not defined. • There are approximations using point kinetics that are no longer applicable with new methods. • Range of rod insertions to investigate is limited to maximum insertion when a less inserted position could result in higher worths. • Section 2.2.4 and 2.2.5 should be reversed for clarity. Section 2.2.5 says only look at maximum worths whereas Section 2.2.4 says to look at other worths. Clarification is needed to define what is required to meet this guidance. [A-11]	b) The NRC agrees with this comment. Text has been revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

Because of burnup-dependent and corrosion-dependent factors that tend to reduce cladding failure thresholds and allowable limits on core coolability during fuel rod lifetime, the limiting initial conditions may involve locations other than maximum uncontrolled rod worth defined in section 2.2.4 (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). As such, a more comprehensive search for the limiting conditions may be necessary to ensure that the total number of fuel rod failures is not underestimated and allowable limits are satisfied. Applicants may need to survey a larger population of BWR blade drop and PWR ejected rod core locations and exposure points to identify the limiting scenarios.

When properly justified, combining burnup-dependent parameters to create an artificial, composite worst time-in-life (e.g., end-of-life cladding hydrogen content combined with maximum ejected worth) is an acceptable analytical approach to reduce the number of cases analyzed.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.5 The maximum rod worth (or differential worth) should be calculated based on the following conditions: (a) all control rods at positions corresponding to values for maximum allowable insertions at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod in each rod group for different control rod configurations, both expected and unexpected. The value of rod worths should be increased, if necessary, to account for calculational uncertainties in parameters (e.g., neutron cross sections) and power asymmetries due to xenon oscillations.

Number of Comments: 2

Comment:	NRC Response:
a) Previous methods may have used an ejected rod worth that was much higher than those realized during plant operation. Smaller ejected rod worths cause the accident to behave more like an uncontrolled Control Element Assembly (CEA) withdrawal. Exelon is requesting further clarification regarding what method is applicable for this event. Should. artificially high ejected rod worths be employed to assess the inherent fuel reactivity feedbacks in a near prompt critical situation, or should more realistic and seemingly less limiting ejected rod worths be employed to assess the possible fuel failure mechanisms, with an associated fuel failure that is likely to be less than that previously analyzed to? [Exelon-3]	a) The NRC agrees with this comment. For assessing the damaged core coolability limit on maximum radial average fuel enthalpy, the maximum possible ejected rod worth needs to be evaluated. However, for assessing radiological consequences, the maximum ejected rod worth may not yield the highest number of failed fuel rods.
b) Exelon recommends deleting the phrase " ... both expected and unexpected ... " in order to avoid confusion. The rod worth calculation requirement is already defined at beginning of Section 2.2.5. In addition, Exelon requests clarification as to whether calculational uncertainties need to be applied beyond those required by approved neutronics methods.[Exelon-7]	b) The NRC agrees with this comment. Text revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

The maximum uncontrolled rod worth (the worth of an ejected rod in a PWR or a dropped blade in a BWR) should be calculated based on the following conditions: (a) the range of control rod positions allowed at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod of all inserted control rods for the allowed configurations highlighted above. The evaluation methodology should account for (1) calculation uncertainties in neutronic parameters (e.g., neutron cross sections) and (2) allowed power asymmetries

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:
C.2.2.6 The reactivity insertion rate should be determined from differential control rod worth curves and calculated transient rod position versus time curves.

Number of Comments: 0

Comment:	NRC Response:

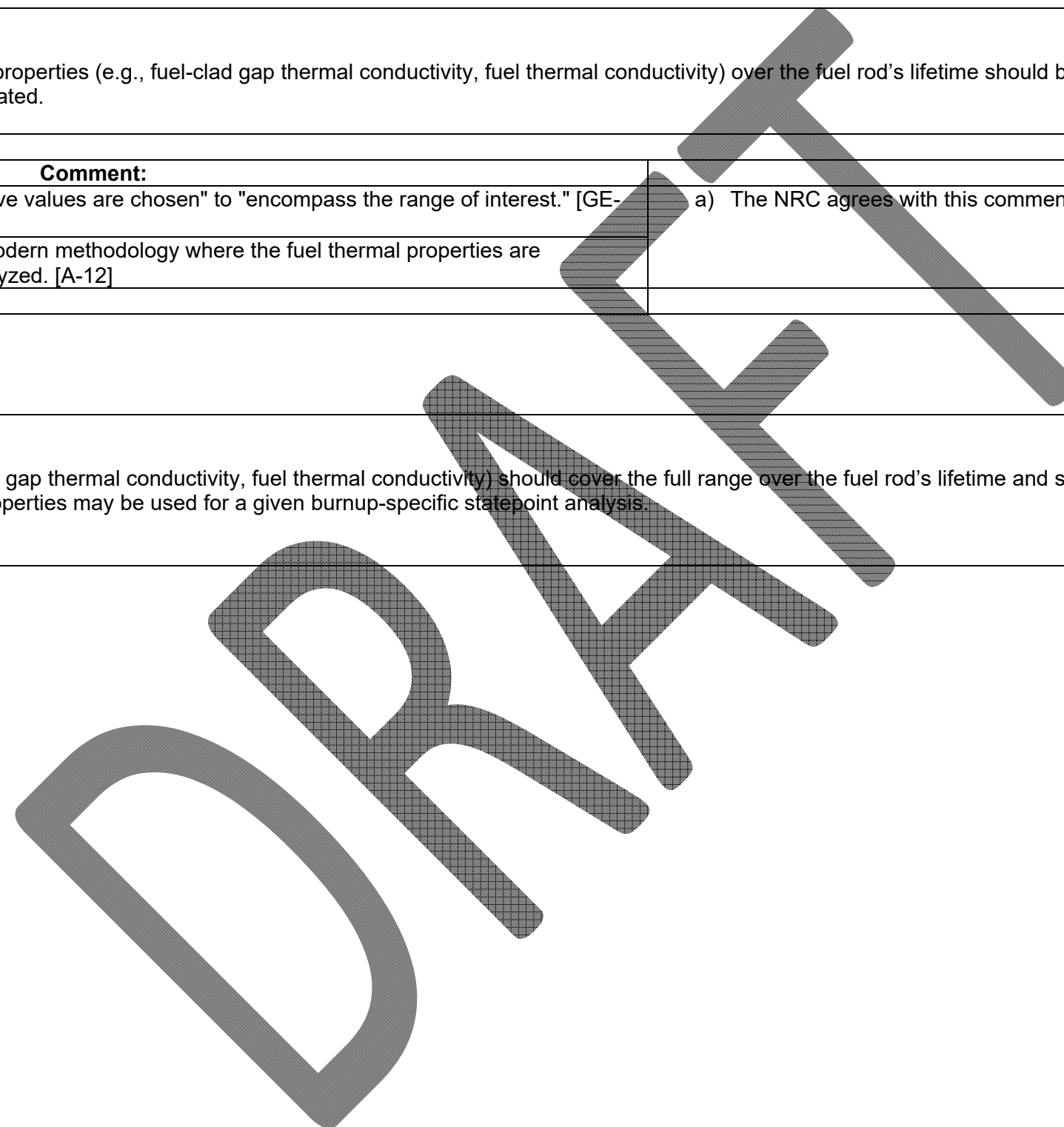| **Resolution / Path Forward / Action Item:** |
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Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.7 For CRE, the rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction. For CRD, credit may be taken for the velocity limiter when determining the rate of withdrawal due to gravitational forces.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:
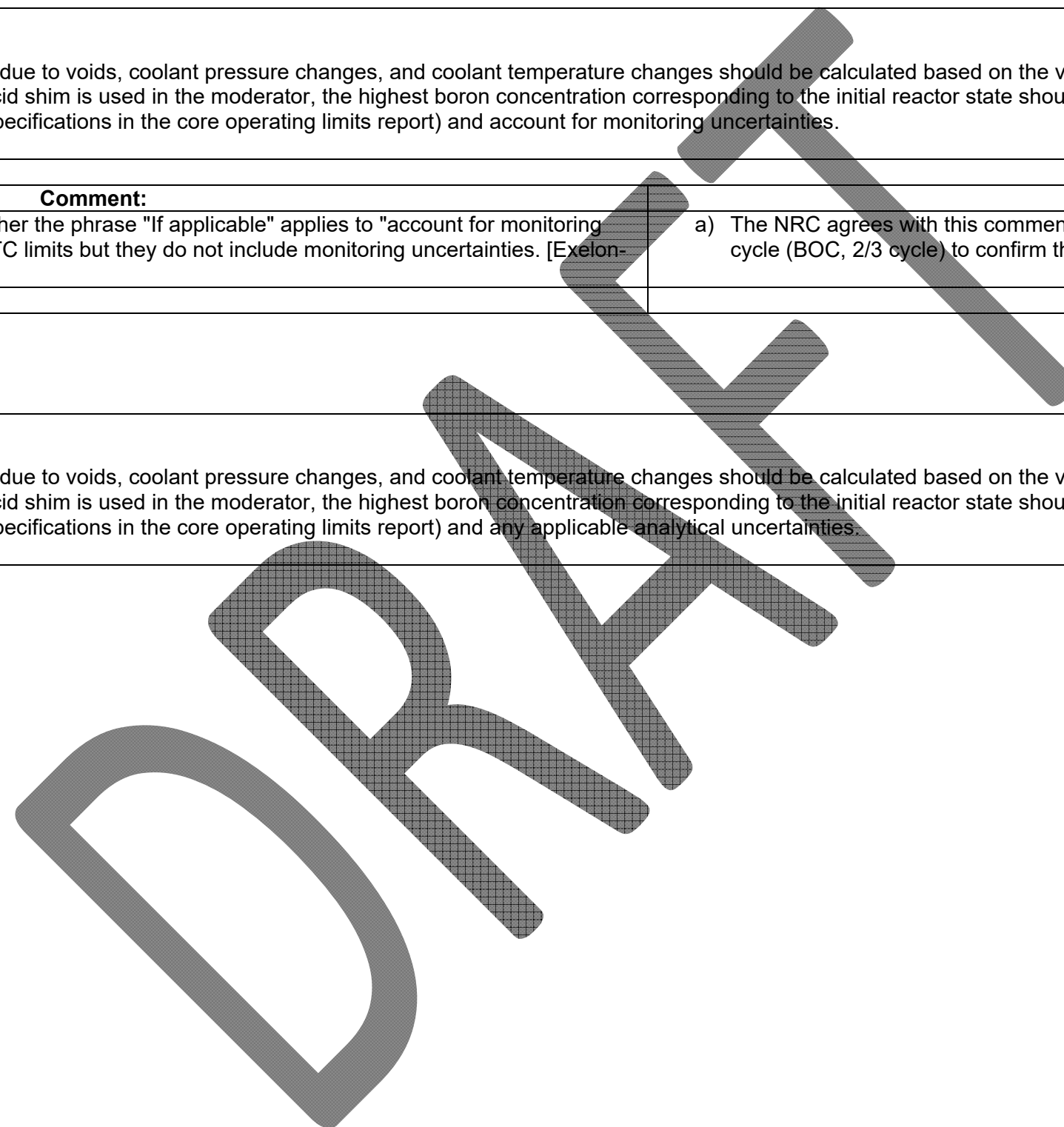
Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.8 The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending upon the transient phenomenon being investigated. Range of values should encompass the allowable operating range and monitoring uncertainties.

Number of Comments: 1

Comment:	NRC Response:
a) The phrase "conservatively chosen" should be replaced with "demonstrated to encompass the range of interest." The sentence "Range of values should encompass the allowable operating range and monitoring uncertainties" should be deleted. [GE-8]	a) The NRC agrees in part with this comment. Ranges in initial coolant temperature, pressure, and flow become important to assess thermal design limits and not for almost instantaneous deposited energy. Furthermore, allowable ranges and monitoring uncertainties may be large at CZP/HZP conditions. Text revised to stipulate at-power scenarios.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.8 For at-power scenarios, the initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen, depending upon the transient phenomenon being investigated. Range of values should encompass the allowable operating range and monitoring uncertainties.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.9 The anticipated range of fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) over the fuel rod's lifetime should be investigated to ensure conservative values are chosen, depending upon the transient phenomenon being investigated.

Number of Comments: 2

Comment:	NRC Response:
a) Replace the phrase "ensure conservative values are chosen" to "encompass the range of interest." [GE-9]	a) The NRC agrees with this comment. Text revised.
b) This guidance is outdated relative to modern methodology where the fuel thermal properties are calculated for the time in life being analyzed. [A-12]	

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.9 Fuel thermal properties (e.g., fuel-clad gap thermal conductivity, fuel thermal conductivity) should cover the full range over the fuel rod's lifetime and should be conservatively selected based on the transient phenomenon being investigated. Time-in-life specific fuel properties may be used for a given burnup-specific statepoint analysis.
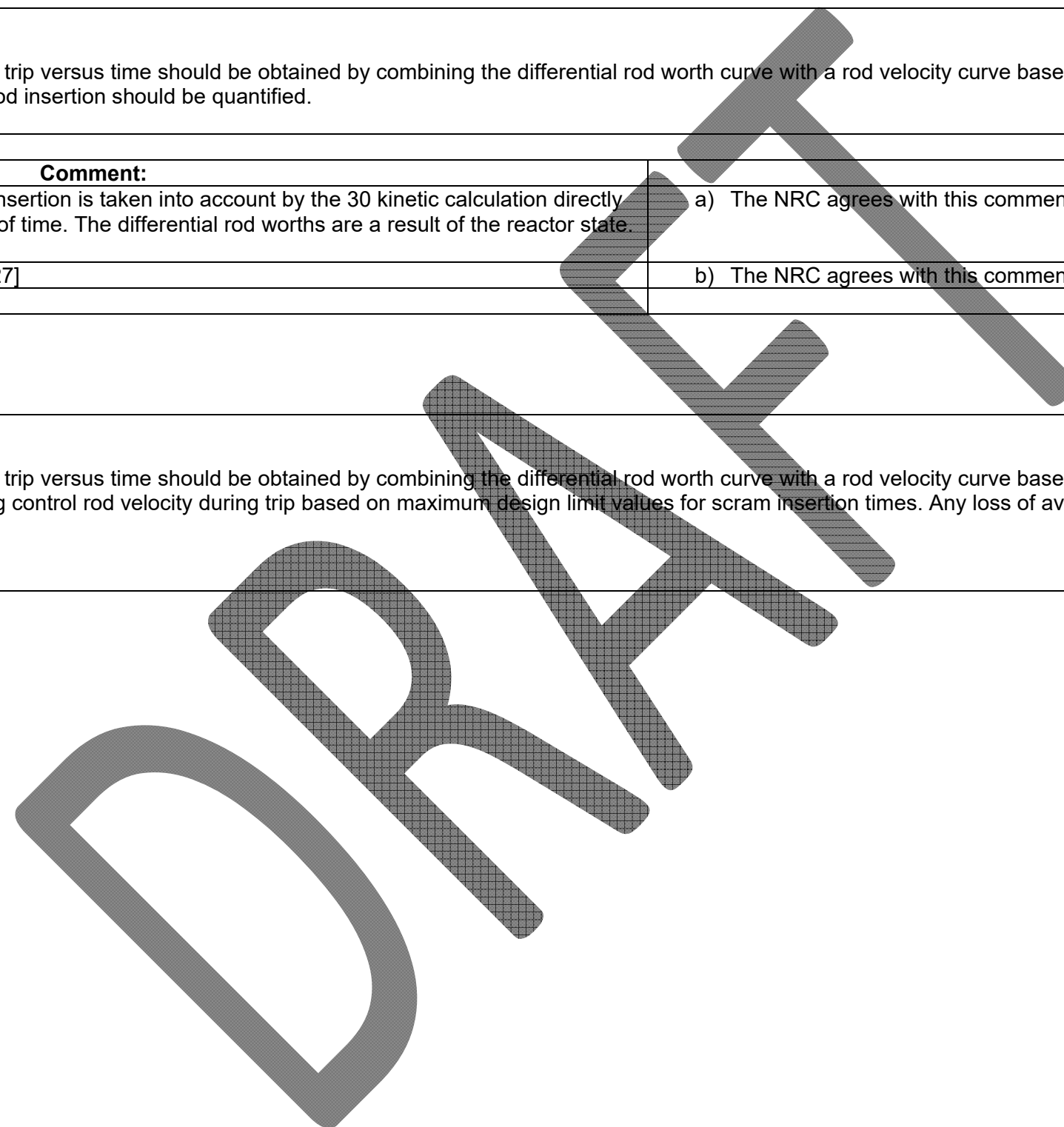
TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.10 The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed. If applicable, the range of values should encompass the allowable operating range (i.e., technical specifications in the core operating limits report) and account for monitoring uncertainties.

Number of Comments: 1

Comment:	NRC Response:
a) Exelon requests clarification as to whether the phrase "If applicable" applies to "account for monitoring uncertainties" by itself. PW Rs have MTC limits but they do not include monitoring uncertainties. [Exelon-8]	a) The NRC agrees with this comment. Some PWRs continue to measure MTC at different points within the cycle (BOC, 2/3 cycle) to confirm the predicted values and compliance to TS/COLR limits. Text revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.10 The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed. If applicable, the range of values should encompass the allowable operating range (i.e., technical specifications in the core operating limits report) and any applicable analytical uncertainties.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.11 Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively with available experimental data. Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied value of the Doppler coefficient.

Number of Comments: 1

Comment:	NRC Response:
a) The text appears to indicate that a single Doppler coefficient value will be used. It also does not allow for a best estimate Doppler model. Basis for concern: Modern codes utilize 3D cross sections to produce Doppler feedback and provide a best estimate model. A conservatively low Doppler model would not necessarily predict a conservative power defect. [A-13]	a) The NRC agrees in part with this comment. Doppler feedback has a significant impact on this analysis and uncertainties in this reactivity component need to be considered. Text revised for clarification.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.11 Calculations of the Doppler coefficient of reactivity should be based on and compared with available experimental data. Since the Doppler feedback reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting the coefficient as well as predicting fuel temperatures at different power levels should be reflected by conservatism in the application of Doppler feedback.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.12 Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. Any loss of available scram reactivity due to allowable rod insertion should be quantified.

Number of Comments: 2

Comment:	NRC Response:
a) In modern methodology, the reactivity insertion is taken into account by the 30 kinetic calculation directly from control rod insertion as a function of time. The differential rod worths are a result of the reactor state. [A-14]	a) The NRC agrees with this comment. Text revised.
b) Section numbering is not accurate. [A-27]	b) The NRC agrees with this comment. Text revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.2.12 Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. Alternatively, reactivity may be calculated using control rod velocity during trip based on maximum design limit values for scram insertion times. Any loss of available scram reactivity due to allowable rod insertion should be quantified.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.2.13 The reactor trip delay time, or the amount of time that elapses between the instant the sensed parameter (e.g., pressure, neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (a) time required for instrument channel to produce a signal, (b) time for the trip breaker to open, (c) time for the control rod motion to initiate, and (d) time required before control rods enter the core if the tips lie outside the core. Allowances for inoperable or out-of-service components and single failures should be included in the response of the reactor protection system.

Number of Comments: 1

Comment:	NRC Response:
a) Editorial changes. (1) Section C.2.2.12: There are two sections labeled C.2.2.12, (2) Section C.2.3.5.1: In the following sentence, "Otherwise, the SRA PCMI failure curves in Figures 4 and 5 should be applied.", change "Figure 4" to "Figure 3", and (3) Reference 10 should be "Terminal..." not "Thermal..." [NEI-A22]	a) The NRC agrees with this comment. Text revised.

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.3.1 At each initial state point, the total number of failed rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing each of the cladding failure thresholds described in Section C.3, "Fuel Rod Cladding Failure Thresholds," of this guide. Applicants do not need to double count fuel rods that are predicted to fail more than one of these thresholds.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.3.2 Figure 1 provides an acceptable high temperature cladding failure threshold as a function of cladding differential pressure. When applying Figure 1, the cladding differential pressure must include both the initial, pre-transient rod internal gas pressure plus any increase associated with transient fission gas release (FGR). An approved fuel rod thermal-mechanical performance code should be used to predict the initial, pre-transient rod internal conditions (e.g., moles of fission gas, void volume, FGR, rod internal pressure). The amount of transient FGR may be calculated using the burnup-dependent correlations provided in Figure 6.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.3.3 Due to the large variation in predicted radial average fuel enthalpy rise along the axial length of a fuel rod, the applicant may elect to (1) calculate transient FGR for several axial regions and (2) combine each axial contribution, along with the pre-transient gas inventory, within the calculation of total rod internal pressure.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

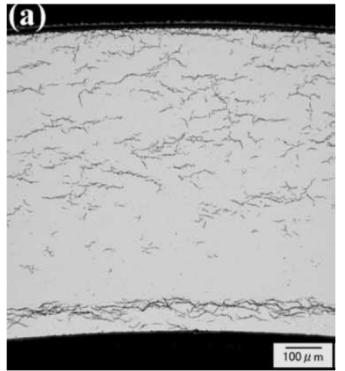| TOPIC: **Analytical Models and Assumptions** |
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DG-1327 Draft Text:

C.2.3.4 When applying the PCMI cladding failure thresholds, an approved alloy-specific cladding corrosion and hydrogen uptake model must be used to predict the initial, pre-transient cladding hydrogen content. The influence of (1) time-at-temperature (e.g., residence time, operating temperatures, steaming rate), (2) cladding fluence (e.g.dissolution of second phase precipitates), (3) enhanced hydrogen uptake mechanisms (e.g., shadow corrosion, proximity to dissimilar metal), and (4) crud deposition must be accounted for in these approved models.

2.3.4.1 Alloy-specific hydrogen uptake models in RG 1.224, "Establishing Analytical Limits for Zirconium-Based Cladding," (Ref. 9) may be used to estimate the pre-transient cladding hydrogen content.

2.3.4.2 The cladding average (e.g., mid-wall) temperature at the start of the transient should be used to define the excess hydrogen in the cladding. Use of the Kearns solubility correlation (Ref. 10) is acceptable.

2.3.4.3 Due to the large variation in predicted radial average fuel enthalpy rise along the axial length of a fuel rod along with potential axial variability in cladding hydrogen content, the applicant may need to perform multiple calculations to identify the limiting axial position. Alternatively, the PCMI cladding threshold corresponding to the predicted peak axial hydrogen content may be used to bound the entire fuel rod.

Number of Comments: 5

Comment:	NRC Response:
a) Due to the unknown future publication date of RG 1.224, the NRC-approved hydrogen uptake models need to be included in the final Regulatory Guide. [NEI-A16]	a) The NRC agrees with this comment. Appendix C was added to document the draft RG 1.224 hydrogen uptake models.
b) Clarify that the hydrogen uptake models in RG 1.224 are acceptable to NRC. [NEI-A20]	b) The NRC agrees with this comment. Clarification added.
c) RG 1.224 does not say that the recommended modern correlation is acceptable, even though the model is included in the RG. Please clarify that the recommended modern correlation is also acceptable. [GE-10]	c) The NRC disagrees with this comment. Draft RG 1.224 states, "Given the allowable range in composition within the Zircaloy-2 ASTM specification (ASTM B351/B351M, "Standard Specification for Hot-Rolled and Cold-Finished Zirconium and Zirconium Alloy Bars, Rod, and Wire for Nuclear Application," Ref. A-3) and the degree of flexibility and variability in manufacturing procedures between the fuel vendors, the staff has elected to adopt the more conservative legacy hydrogen uptake model." This logic is retained. Applicants may elect to use an alternative cladding hydrogen uptake model, with appropriate justification.
d) The second sentence of the above paragraph, with items (1) to (4) is not necessary in the context of 2.3.4 and stipulates technical requirements for the hydrogen uptake model that are not necessary. [A-15]	d) The NRC agrees with this comment. Proposed language adopted.
e) Fuel vendors may have a validated hydrogen pick-up model which differs from the one in RG 1.224. The choice to use an alternative model should be explicitly given. The PCMI thresholds are stated versus excess hydrogen content as defined in Ref. 10, but it is not explained whether the hydrogen content is that of the metal meat only or includes the hydrogen content in the oxide layer, or the liner. [A-16]	e) The NRC agrees with this comment. Paragraph C.2.3.4 states that an approved hydrogen uptake model (i.e., vendor supplied model) is required. As an alternative, Paragraph C.2.3.4.1 allows the use of models provided in the new Appendix C. Hence both options are available to the applicant. The measured and estimated cladding hydrogen content in the empirical database used to develop the PCMI failure curves are based on total hydrogen content, including any hydrogen present in the oxide layer. Therefore, total hydrogen content should be used to implement these curves. If an applicant elects to use their own approved alloy-specific hydrogen model which separates out hydrogen in the oxide layer, then these curves would no longer be applicable. Text will be added to the guidance to reflect this important clarification. With respect to the BWR liner fuel, the natural or low alloy zirconium liner acts as a sponge for hydrogen. As shown in the micrograph, hydrogen absorbed via waterside corrosion diffuses to the barrier liner on the cladding ID. Hydrogen, which migrates down the temperature gradient, may even form a hydride rim near the interface between the liner and Zry-2 cladding. Hence, the presence of a liner depletes the base metal of hydrogen and the detrimental effects of hydrides.

As described in Ref. 9 of the technical basis document (M. Aomi, et.al.), the Zr liner has 'high ductility, even after irradiation and the precipitation of a high content of hydrides.' Since the liner depletes the base metal of hydrogen and remains ductile, it is likely that liner fuel would exhibit more overall ductility than non-liner fuel at the same excess hydrogen level. Unfortunately, there are no failed test results for liner and non-liner fuel rod segments near the same excess hydrogen content to prove this point.

The NSRR FK series test results set the inflection points in the RXA PCMI failure curves. FK-1,3,4 (non-failed) set the initial drop from 150 Δcal/g at 75 wppm excess hydrogen. FK-9 (failed) sets the slope and inflection point at 75 Δcal/g with 150 wppm excess hydrogen. FK-6,7,10,12 (failed) set the slope of the line for higher concentrations up to 300 wppm (line was subsequently extended based on comment, See Attachment 1). All of these tests were conducted on Zry-2 lined cladding. Based on the discussion above, application of this RXA PCMI failure threshold to non-lined cladding may be non-conservative. Scaling this line for non-liner fuel would involve shifting the line in the direction of lower excess hydrogen to account for hydrides present in the ductile liner. Since an established relationship between hydrogen distribution in the liner and base metal is not readily available, any scaling is beyond the scope of this guidance document. Note that application of test result from non-lined RXA cladding to lined RXA cladding would be conservative since it ignores the hydrides present in the ductile liner.

There are many non-failed tests on non-lined RXA cladding at or below 70 wppm excess hydrogen. In addition, there are two failed tests on non-lined RXA cladding (LS-1 at 300 wppm and VA-6 at 708 wppm). However, the slope of the failure threshold is difficult to predict between 70 and 300 wppm. Based upon the limited database, the staff finds the RXA PCMI failure threshold acceptable for non-lined cladding up to 70 wppm excess hydrogen.

For the domestic BWR fleet, plants are currently fueled with either lined RXA Zry-2 cladding or lined (maybe a few non-lined) SRA Zry-2 cladding. There are no plants fueled with non-lined RXA cladding. So, this limitation on the applicability of the RXA PCMI curve does not introduce undue burden.

For the domestic PWR fleet, plants are currently fueled with either SRA ZIRLO, pRXA Optimized ZIRLO, or RXA M5 cladding. SRA Zry-4 is no longer being loaded in batch quantities (re-inserts still available, but fewer each year). The hydride morphology of pRXA Optimized ZIRLO has not yet been demonstrated, but is believed to be leaning toward SRA material. The Framatome M5 cladding has beneficial corrosion properties and unlikely to absorbed more than 100 wppm of hydrogen at end-of-life. This translates into approximately 25 wppm excess hydrogen at PWR operating conditions. Hence, limiting the RXA PCMI failure applicability to 70 wppm excess hydrogen will not impact PWRs operating with M5 cladding.

Text will be added to the guidance limiting the applicability of the RXA PCMI failure curve to 70 wppm for non-lined Zry-2 cladding.

Resolution / Path Forward / Action Item:
Text revised. Appendix C added.

Revised RG Text:

1.2.3   Due to the dominant role liner fuel test results played in the development of the RXA PCMI cladding failure threshold curves and the influence of the natural of low alloy liner on the initial hydride distribution, the applicability of these failure threshold curves for non-liner cladding designs is limited to cladding with less than 70 wppm excess hydrogen.

C.2.3.4 When applying the PCMI cladding failure thresholds, an NRC-approved alloy-specific cladding corrosion and hydrogen uptake model must be used to predict the initial, pre-transient cladding hydrogen content. The influence of (1) time-at-temperature (e.g., residence time, operating temperatures, steaming rate), (2) cladding fluence (e.g. dissolution of second phase precipitates), (3) enhanced hydrogen uptake mechanisms (e.g., shadow corrosion, proximity to dissimilar metal), and (4) crud deposition should be accounted for in these approved models either directly or implicitly through the supporting database.

2.3.4.1 As an alternative, acceptable alloy-specific hydrogen uptake models to estimate pre-transient cladding hydrogen content are provided in Appendix C.

2.3.4.2 The measured and estimated cladding hydrogen content in the empirical database used to develop the PCMI failure curves are based on total hydrogen content, including any hydrogen present in the oxide layer. Therefore, total hydrogen content should be used to implement these curves. If an applicant elects to use their own approved alloy-specific hydrogen model which separates out hydrogen in the oxide layer, then these curves would no longer be applicable.

2.3.4.3 The mid-wall cladding temperature at the start of the transient should be used to define the excess hydrogen in the cladding. Use of the Kearns solubility correlation (Ref. 10) is acceptable.

2.3.4.4 Due to the large variation in predicted radial average fuel enthalpy rise along the axial length of a fuel rod along with potential axial variability in cladding hydrogen content, the applicant may need to perform multiple calculations to identify the limiting axial position. Alternatively, the PCMI cladding threshold corresponding to the predicted peak axial hydrogen content may be used to bound the entire fuel rod.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.3.5 Because of the thermo-mechanical treatment of the cladding material under fabrication and its effect on the final cladding microstructure, zirconium hydride platelets will precipitate in a preferential orientation. Usually, SRA cladding exhibits circumferentially orientated zirconium hydride platelets, whereas RXA cladding tends to exhibit randomly oriented zirconium hydride platelets. In addition to fabrication-related effects, the hydride orientation is also affected by the stress state prevailing during hydride precipitation (Ref. 11). As described in References 11 and 12, hydride reorientation from the circumferential direction to the radial direction is possible when the fuel rod is heated and subsequently cooled under an applied tensile load (e.g., high rod internal pressure).

2.3.5.1 The RXA PCMI failure curves in Figures 2 and 4 should be applied to any zirconium alloy cladding material that exhibits more than 10 percent of the zirconium hydrides aligned in the radial direction. Otherwise, the SRA PCMI failure curves in Figures 4 and 5 should be applied.

2.3.5.2 Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown.

Number of Comments: 6

Comment:	NRC Response:
a) Guidance to use the RXA cladding failure thresholds for SRA cladding with >10% radial hydrides is not workable with excess hydrogen concentration limited to 300 ppm. [NEI-A11]	a) The NRC agrees with this comment. RXA PCMI cladding failure threshold expanded based on separate comment.
b) Guidance is needed regarding an acceptable method for evaluating the possibility of hydride reorientation during power maneuvers or a reactor shutdown prior to an RIA event. [NEI-A23, APS-5]	b) The NRC disagrees with this comment. Rod internal pressure limits established as part of the fuel rods mechanical design should consider the possibility of hydride reorientation. NUREG-0800 SRP 4.2, Fuel System Design, Section II.1.A.vi states that rod internal pressure should be limited to preclude (in addition to other criteria) reorientation of the hydrides in the radial direction in the cladding.
c) The incorrect Figures are referenced. SRA is covered in Figures 3 and 5, rather than Figures 4 and 5. [NuScale-2, A-28]	d) The NRC agrees with this comment. Typo corrected.
e) The 10% threshold is too restrictive and in contradiction with industry standards in ASTM B811-13. Section 2.3.5.1 also contradicts Section 1.3 with regards to material classification: Section 1.3 defines two categories, SRA and RXA, whereas Section 2.3.5.1 introduces another classification criterion based on hydride orientation. If the two classification criteria were equivalent then they would also be redundant and only one of them should be kept, namely the original criterion based on metallurgical state as defined in 1.3. **Basis for concern:** In the context of sub-section 2.3.5.1, the 10 percent fraction of hydrides that can be radially oriented represents the radial hydride orientation fraction (Fn) that is routinely measured during fabrication. The ASTM standard B811-13 defines the acceptance criteria for this Fn fraction to be: no more than 30% for SRA and not greater than 50% for RXA; therefore, the threshold between SRA and RXA is an Fn value of 30%. Proposed changes included. [A-3]	e) The NRC agrees in part with this comment. Text revised to eliminate separate classification based on percentage of hydrides aligned in the radial direction. The relevant section from ASTM B811-13 is provided below. The commenter stated that the proposed 10% threshold on radial hydrides is too restrictive and in contradiction with industry standards. In accordance with ASTM B811-13, the commenter proposed a threshold Fn value of 0.30 between SRA and RXA. M. Aomi et.al. (Reference 9 of technical basis document) describes testing on irradiated cladding segments. As part of this investigation, the initial, as-irradiated zirconium hydride orientation was measured. As shown in Figure 4 of this journal article, Fn(40) (fraction of number of hydrides in radial direction $\pm$ 40°) was reported as approximately 0.08 and 0.12 for irradiated RXA Zry-2 at 50 and 55 GWd/MTU respectively. The corresponding values of Fl(45) (fraction of length of hydrides in radial direction $\pm$ 45°) were approximately 0.18 and 0.28. In contrast, the reported Fl(45) was approximately 0.18 and 0.19 for SRA Zry-4 at 39 and 48 GWd/MTU respectively. Fn(40) values not reported for SRA Zry-4 cladding. This independent data suggests that (1) the as-fabricated ASTM testing methods and results may not be valid to characterize hydride precipitation in irradiated cladding and (2) the as-fabricated ASTM threshold Fn value of 0.30 is not valid to distinguish RXA and SRA performance. Given that measured values of Fn(40) were approximately 0.08 and 0.12 for irradiated RXA Zry-2, maybe the originally proposed 10% threshold was reasonable. Nevertheless, to expedite publication of this guidance and recognizing that further work in this area is necessary, the text will be revised and the burden of proof of applicability placed on the applicant.

	ASTM B811, Standard Specification for Wrought Zirconium Alloy Seamless Tubes for Nuclear Reactor Fuel Cladding
	8.3 Hydride Orientation Fraction:
	8.3.1 Hydride orientation fraction, Fn, shall be determined on samples taken from mill finished tubes.
	8.3.2 The hydride orientation shall be determined in accordance with Annex A2.
	8.3.3 Acceptance Criteria—Stress relief annealed specimens shall have an Fn value not more than 0.30. Recrystallization annealed specimens shall have an Fn value not greater than 0.50.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

C.2.3.5 Because of the thermo-mechanical treatment of the cladding material under fabrication and its effect on the final cladding microstructure, zirconium hydride platelets will precipitate in a preferential orientation. For SRA cladding, a majority of zirconium hydride platelets will precipitate in the circumferentially orientation. Whereas RXA cladding tends to exhibit randomly oriented zirconium hydride platelets. In addition to fabrication-related effects, the hydride orientation is also affected by the stress state prevailing during hydride precipitation (Ref. 11). As described in References 11 and 12, hydride reorientation from the circumferential direction to the radial direction is possible when the fuel rod cladding is loaded in tension beyond the hydride reorientation stress threshold. Each applicant should address the possibility of hydride reorientation because of power maneuvering or reactor shutdown consistent with the requirements in NUREG-0800 Section 4.2, Fuel System Design, Section II.1.A.vi, page 4.2-7, Revision 3, March 2007.

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.4.1 Because of the large variation in predicted fuel radial average enthalpy rise along the axial length of a fuel rod, the applicant may elect to (1) calculate the transient fission product release fraction for each radionuclide for several axial regions and (2) combine each axial contribution, along with the pre-transient, steady-state inventories, to obtain the total radiological source term for dose calculations.

Number of Comments: 0

Comment:	NRC Response:

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Analytical Models and Assumptions**

DG-1327 Draft Text:

C.2.5.1 The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves, as appropriate. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

Number of Comments: 3

Comment:	NRC Response:
a) For control rod ejection (CRE), since the reactivity-initiated accident (RIA) transient is caused by the pressure boundary breach, the analysis should be able to credit the pressure boundary breach in the peak RCS pressure analysis. [SCEG-3]	a) The NRC agrees, in part, with this comment. Depending on its design, the CEA/RCCA may partially block the CEDM nozzle rupture. An applicant would need to justify the dynamic break flowrate based on plant-specific design characteristics. However, given the short time duration of the power excursion and resulting pressure surge, credit for the relatively small break area would not likely significantly impact peak pressure calculations. The text will remain. Applicants may justify an alternative treatment of the pressure boundary breach.
b) This section was written from the perspective of a PWR. [GE-11]	b) The NRC agrees with this comment. Need to add text related to CRD.
c) This guidance does not specify the treatment of the potential pressure reduction caused by the assumed failure of the control rod pressure housing on other criteria specified in the document, only for RCS peak pressure. This leads to uncertainty in how to treat the pressure impact of the assumed failure of the control rod pressure housing on the other criteria specified in the document such as: Figure 1 High Temperature Cladding Failure Threshold, DNB and CPR. [A-17]	c) The NRC agrees with this comment. This discussion relates solely to PWR CRE. Still under development. First draft. Specific modeling of a small break in the reactor vessel upper head will promote a gradual loss of inventory and depressurization of the RCS. For the simulation of a high worth CRE scenario with its rapid (prompt critical, or close to prompt critical) power excursion, the predicted accident progression and consequences would not be significantly impacted. The power excursion, inherent doppler feedback, and response of safety-related SSCs (e.g., high power reactor trip, scram, pressure relief values) occur in a short time frame. For the simulation of a low worth CRE scenario, initial conditions could be orchestrated to produce a slowly evolving transient which delays or even avoids a reactor trip. For this scenario, the gradual loss of inventory and depressurization become more important. RCS depressurization would promote DNB cladding failures and cladding burst. This may potentially increase the predicted number of failed rods which is input to the dose calculation. The CRE event originates from concerns associated with rapid, or prompt, power excursions which could significantly disturb the fuel bundle array, pulverize or melt fuel pellets, disperse fuel particles, promote a rapid generation of steam, and challenge the integrity of the reactor vessel and its internals and the ability to cool the core. The CRE serves as a design basis accident for the reactor pressure boundary, safety-related pressure relief functions, excore detector high-flux reactor trip function, control rod design and insertion limits, and fuel design and core loading pattern. Unlike other UFSAR design basis accidents, operator actions are not credited, EOP actions are not designated, and the long-term performance of safety-related SCC to mitigate the consequences and active diminishing core temperatures and shut-down cooling is not demonstrated. The CRE design basis has evolved with the inclusion of accident progressions dictated by RCS depressurization. Safety-related SCCs never before associated with CRE now become important. For example, Low RCS Pressure trip function, variable overpower trip function, Thot saturation trip function, containment trip and isolation functions, and even ECCS safety injection. Furthermore, the accident progression and response of control room operators to such a scenario begin to mimic a SBLOCA.

	From a public safety perspective, the low worth CRE scenario is not limiting. A breach in the reactor vessel would provide a radiological pathway similar to LOCA (i.e., RCS activity released into large containment structure with small leakage to atmosphere). Any additional fuel rod cladding failures associated with the RCS depressurization would be insignificant with respect to the LOCA source term which assumes significant core wide damage and fuel melt. For these reasons, the NRC staff believes that the original CRE design basis should be preserved.

Resolution / Path Forward / Action Item:

Revised RG Text:

TOPIC: **Fuel Rod Cladding Failure Thresholds**

DG-1327 Draft Text:

3. Fuel Rod Cladding Failure Thresholds

Depending on the amount and rate of reactivity insertion, fuel rods may experience several degradation mechanisms and failure modes. During a prompt critical reactivity insertion (i.e., $\Delta\rho/\beta$eff > 1.0), fuel temperatures may approach melting temperatures, and rapid fuel pellet thermal expansion may promote PCMI cladding failure. During more benign power excursions, local heat flux may exceed critical heat flux conditions, prompting fuel cladding temperatures to rise. Fuel cladding may fail because of oxygen-induced embrittlement (i.e., brittle failure) or fuel rod ballooning and rupture (i.e., ductile failure). To ensure a conservative assessment of onsite and offsite radiological consequences, each of these failure modes must be quantified, and the sum total number of failed fuel rods must not be underestimated.

Number of Comments: 1

Comment:	NRC Response:
a) This introduction should remain neutral on cause and effect as to what criteria are analyzed for the type of event. [A-18]	a) The NRC agrees with this comment. The proposed text was adopted.

Resolution / Path Forward / Action Item:
Text revised.

Revised RG Text:

3. **Fuel Rod Cladding Failure Thresholds**

Depending on the energy deposition level and the heat transfer from the rod, the following phenomena can occur: fuel temperatures increase and may approach melting temperatures (both rim and/or centerline), rapid fuel pellet thermal expansion may promote PCMI cladding failure, and local heat flux may exceed critical heat flux conditions, prompting fuel cladding temperatures to rise leading to other potential fuel failure mechanisms.

The following sections define acceptable fuel rod cladding failure thresholds which encompass each degradation mechanism and failure mode. To ensure a conservative assessment of onsite and offsite radiological consequences, each of these failure modes must be quantified, and the sum total number of failed fuel rods must not be underestimated.

Alternative fuel rod cladding failure criteria may be used if they are adequately justified by analytical methods and supported by sufficient experimental data. Alternative cladding failure criteria will be addressed on a case-by-case basis.

TOPIC: **Fuel Rod Cladding Failure Thresholds**

DG-1327 Draft Text:

3.1 High Temperature Cladding Failure Threshold
The empirically based high temperature cladding failure threshold is shown in Figure 1. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events initiated from lower operating modes (e.g., Mode 2, less than 5 percent reactor power). Because ductile failure depends on both cladding temperature and differential pressure (i.e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in total peak radial average fuel enthalpy (cal/g) versus fuel cladding differential pressure (MPa).

For all other operating conditions up to full power (i.e., Mode 1), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

Number of Comments: 8

Comment:	NRC Response:
a) Specifically include language allowing submittal of alternate cladding failure thresholds. [NEI-A3]	a) The staff agrees with this comment. Text added in Section 3 to allow an alternative.
b) For reactor designs where Mode 2 does not coincide with low power conditions it does not make sense to refer to Mode 2 while addressing low power conditions (<5%). Remove reference to Mode 2. [NuScale-3]	b) The staff agrees with this comment. Text revised to remove operating modes.
c) Recommended change: The empirically based high temperature cladding failure threshold is shown in Figure 1. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events ~~initiated from lower operating modes (e.g., Mode 2, less than 5 percent reactor power)~~ to prompt- critical power excursions. Because ductile failure depends on both cladding temperature and differential pressure (i.e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in total peak radial average fuel enthalpy (cal/g) versus fuel cladding differential pressure (MPa). For ~~all other operating conditions up to full power (i.e., Mode 1)~~ at-power (i.e., < 5% rated power up to full power) non-prompt critical power excursions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios). [W-4]	c) The NRC disagrees with this comment. The proposed text does not cover a scenario involving a non-prompt power excursion initiated from below 5% power. For these events where the initial heat flux is extremely low, use of the at-power thermal design limits becomes questionable. Application of Figure 1 is appropriate for these cases, and the calculated fuel enthalpy will reflect the actual power excursion (prompt or non-prompt).
d) Regulatory Guide 1.77 established the presumption of cladding failure at the onset of DNB. However, RG 1.77 also included the following provision: "Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data." Alternative cladding failure criteria will be addressed on a case-by-case basis. Westinghouse believes this alternative should be added back into the final regulatory guide such that alternative failure criteria other than DNB can be used in the future as different experimental data or improved methods become available. This provision has been in the previous two versions of DG-1327. [W-5, SNC-2, A-19]	d) The staff agrees with this comment. Text added in Section 3 to allow an alternative.
e) Remove the sentence: "For all other operating conditions up to full power (i.e., Mode 1), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios)." Note boiling transition is not a presumed failure mechanism for BWR fuel. [GE-12]	e) The NRC disagrees with this comment. Boiling transition is a figure of merit to define cladding failure. Alternate failure thresholds are allowed (See (a) above). It is not the intent of this RG to cover the nuances of each vendor's methodology as it current stands and how it may develop in the future. If a vendor's methodology can show that the post event heat up is bounded by another event or non-limiting than that is an acceptable approach to disposition that part of the accident. The intent of including post event heat up it to establish that it must be considered in future fuel designs even if the current state of analysis indicates that it is not limiting. It is conceivable that future fuel designs and changes to methodology could make this part of the event limiting.
f) The use of "total peak radial average fuel enthalpy" is not consistent with the terminology in Figure 1. [A-29]	f) The staff agrees with this comment. Text revised to remove "total."

Resolution / Path Forward / Action Item:

Revised text.

Revised RG Text:

3.1 High Temperature Cladding Failure Threshold

The empirically based high temperature cladding failure threshold is shown in Figure 1. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events initiated from reactor startup conditions up to 5 percent reactor power operating conditions. Because ductile failure depends on both cladding temperature and differential pressure (i.e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (cal/g) versus fuel cladding differential pressure (MPa).

For at-power operating conditions (i.e., above 5 percent reactor power), fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

TOPIC: **Fuel Rod Cladding Failure Thresholds**

DG-1327 Draft Text:
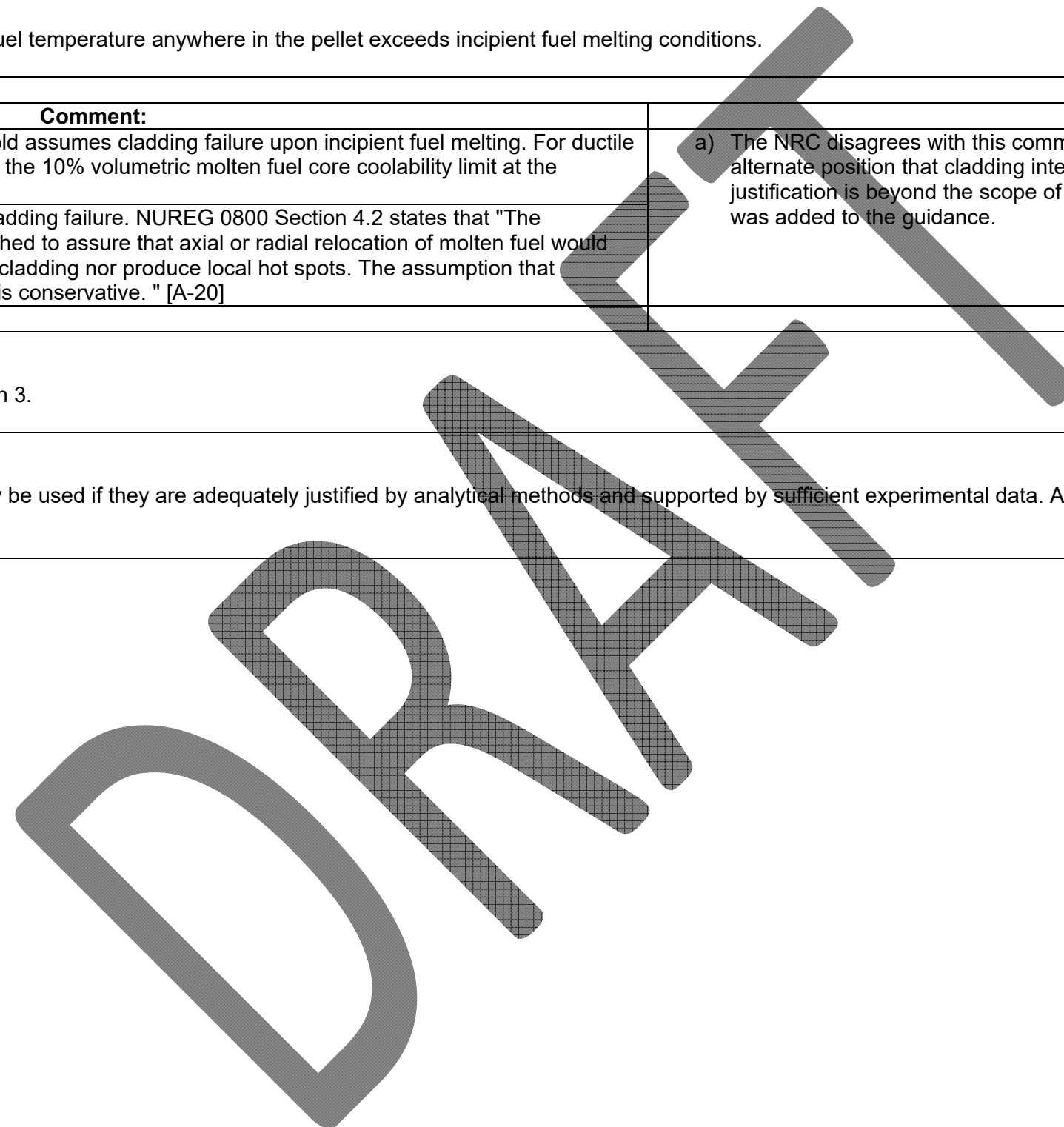
3.2 PCMI Cladding Failure Threshold
The empirically based PCMI cladding failure thresholds are shown in Figures 2 through 5. Because fuel cladding ductility is sensitive to initial temperature, hydrogen content, and zirconium hydride orientation, separate PCMI failure curves are provided for RXA and SRA cladding types at both low temperature reactor coolant conditions (e.g., BWR cold startup) and high temperature reactor coolant conditions (e.g., PWR hot zero power). The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content (weight parts per million [wppm]). Excess cladding hydrogen content means the portion of total hydrogen content in the form of zirconium hydrides (i.e., does not include hydrogen in solution).

Number of Comments: 12

Comment:	NRC Response:
a) The RIA test facility data used by the NRC to develop the cladding failure thresholds due to pellet-to-cladding mechanical interaction (PCMI) do not represent the conditions that are simulated for the hypothetical PWR control rod ejection and BWR control rod drop design basis accidents. The coolant temperature, the cladding temperature response, and the power pulse width resulting from the reactivity excursion are atypical and result in the overly conservative cladding failure thresholds proposed by the NRC. [NEI-1]	a) The NRC agrees with this comment. However, attempts have been made to understand the influence of non-typical experimental conditions and scale, as appropriate, the experimental results.
b) The NRC research was limited to 300 ppm hydrogen content in the proposed failure threshold for recrystallized annealed Zircaloy-2 cladding. This hydrogen content does not support current industry use of Zircaloy-2 for BWRs. EPRI has performed test programs to extend the hydrogen content to 593 ppm. [NEI-2]	b) The NRC agrees, in part, with this comment. Based upon recent test results on higher hydrogen content RXA cladding, the PCMI failure curves were extended. See response to comment (g) below.
c) PNNL Report 22549 predates the corrections to the NSRR test data. This report is part of the technical basis for the PCMI cladding failure thresholds in DG-1327. The impact of the corrections to the NSRR data needs to be verified as addressed in the use of the PNNL Report 22549 content in the final Regulatory Guide. [NEI-A12]	c) The NRC agrees with this comment. While the PNNL was not revised, the staff's technical bases document was revised to account for corrections to the NSRR test data. The guidance provided in DG-1327 already accounts for these corrections.
d) For the BWR control rod drop accident only the energy deposited during the prompt part of the pulse is to be included when comparing to the PCMI cladding failure threshold. The energy deposited in the tail is beyond the timescale of the challenge to cladding integrity from PCMI. [NEI-A14]	d) The NRC agrees with this comment. Text from the Interim RIA Guidance (NUREG-0800, SRP-4.2 Appendix B) was added to Section 2.3.
e) Will extrapolation of the cladding failure thresholds in Figures 2-5 to high excess hydrogen values be allowed? [NEI-A17]	e) See response to comment (g) below.
f) The DG-1327 PCMI guidance is not applicable for BWR Zr-2 RXA cladding if criticality is restricted to ≥ 100°C (212°F) as the cladding will be ductile. [NEI-A28]	f) The NRC staff does not agree with this comment. While material ductility is enhanced and zirconium hydrides will dissolve (into solution) at increased temperatures, there is no dramatic step change in cladding properties or performance under RIA conditions at 100°C.
g) The fuel enthalpy rise curves for RXA cladding material end at an excess hydrogen concentration of 300 wppm. Higher hydrogen concentration was measured for RXA Zry-2 cladding. Proposed changes included in comment. [A-4]	g) The staff agrees with this comment. The PCMI failure curves have been re-drawn based upon the information provided in the comment. See attachment 1 for further information.
h) Based on fuel cladding outer surface temperature calculations and observed pulse width dependence, under typical BWR condition, the RXA BWR cladding would not be susceptible to PCMI type of failure as indicated in Figure 12 by the failure line normalized to 85°C and a 15 ms pulse width scenario. Furthermore, in most of the cases the starting cladding temperature would be higher than 25°C since credible RIA events for most of the plants won't occur until well above 70°C, see Figure 13. Therefore, EPRI proposes a constant 150 cal/g energy deposition limit for the BWR RXA cladding PCMI failure threshold up to 593 ppm hydrogen. [EPRI-1]	h) The NRC disagrees with this comment. The NRC has elected to develop cladding failure thresholds based on in-pile testing. Attempts have been made to understand the influence of non-typical experimental conditions and scale, as appropriate, the experimental results. Out-of-pile tests are useful to further understand the phenomena and may be able to demonstrate equivalent performance of future zirconium-based cladding alloys. The RG provides one acceptable set of PCMI cladding failure threshold curves. An applicant is able to propose alternatives with proper justification.
i) Irradiated BWR RXA cladding test samples used in the EPRI MBT program were sectioned from four different fuel rods irradiated in different plants. Sample hydrogen concentrations range from 100 to 593 ppm and much of the test data were generated with the high hydrogen content cladding. The test results show the burst strain is not dependent on the hydrogen concentration at both room and elevated temperatures and no anomalous behavior was observed. A possible explanation for lack of hydrogen	

	dependence is described in reference 1. In the most recent test campaign with low hydrogen concentration cladding (~100 ppm) one test sample displayed higher ductility. Since there may be variation in the hydrogen concentration and hydride distribution, the actual hydrogen content of the test sample may be close to the solubility limit. Therefore, it is possible the hydride effect (lower ductility relative to SRA cladding at below 400 ppm hydrogen) only manifested at the solubility limit and saturates thereafter. The RXA fuel cladding is well behaved up to 593 ppm hydrogen concentration and therefore EPRI proposes the regulatory limit be extended to 593 ppm. [EPRI-2]	
i)	The test data show SRA cladding burst strain below 450 ppm of hydrogen is greater than 2%. This level of ductility is confirmed by the CABRI REPNa3 test where the hydrogen was calculated, based on 15% pickup fraction, to be approximately 400 ppm. At an energy deposition of 138 cal/g, the calculated thermal expansion is 2%. A residual hoop strain of 2.2% was reported, this likely included cladding expansion due to internal pressure later on as the internal gas pressured increased. The PWR SRA PCMI failure limits were re-evaluated using an updated CSED failure model based on the MBT data. The results of the evaluation using the Falcon fuel performance code is plotted in Figure 14 and are consistent with the hand calculation described previously. Based on the technical basis provided, EPRI proposes the 280°C and 10 ms pulse width curve for the SRA PWR cladding PCMI limit, as shown in Figure 14. [EPRI-3]	
j)	Although PWR RXA cladding was not directly tested in the EPRI program, the impact of potential radial hydride formation would be the same for both alloy types (BWR and PWR RXA). Test results show the ductility of SRA and RXA cladding increase with increasing temperature, with both cladding types having greater than 2% burst strain up to 450 ppm hydrogen at hot zero power condition and at pulse widths of ~3 ms. The RXA cladding ductility is fully recovered above 85°C for pulse widths longer than 13 ms up to 593 ppm of hydrogen. A key difference between the BWR RXA and PWR RXA cladding is the presence of a liner in the BWR RXA cladding. The hydrogen concentration in the liner is typically higher than the main cladding, see Figure 15(a), and contain a significant portion of the total cladding hydrogen concentration. BWR RXA cladding with up to 593 ppm of hydrogen was tested in the EPRI program and the hydride morphology/distribution for this material is shown in Figure 15(a) and is compared to a typical PWR RXA cladding hydride morphology in Figure 15(b). Assuming 1/2 of the cladding hydrogen of the 593 ppm test samples reside in the liner, the main cladding hydrogen concentration should be 296 ppm. Therefore, the PWR RXA cladding would be ductile up to a minimum hydrogen concentration of 300 ppm. EPRI proposes a constant 150 cal/g limit up to 300 ppm of hydrogen. [EPRI-4]	
k)	The effects of pulse-width and temperature on cladding ductility observed with PWR SRA cladding would be applicable to SRA cladding in BWR application. This approach is consistent with PNNL/NRC's use of PWR SRA data as a basis for BWR SRA CZP limit [10-12]. Testing at intermediate temperatures were not conducted for the PWR SRA cladding and therefore adjustment to the NSRR test results only considers the pulse-width effect. However, BWR channel test results shown in Figure 5 indicate the ductility does improve at slightly elevated temperatures. Test data plotted in Figure 11 indicates SRA cladding at room temperature to have a stronger pulse width dependence than RXA cladding, but because of the test data pulse width range, the lower slope room temperature RXA cladding pulse width dependence is used to adjust the NSRR test data. The BWR RXA cladding room temperature pulse width dependence, 0.056%/ms, is used to adjust the proposed limit to 20 ms pulse width and is shown in Figure 16. Based on the technical basis provided, EPRI proposes the loading rate effect adjusted curve for the SRA BWR cladding PCMI limit, as shown in Figure 16. [EPRI-5]	

Resolution / Path Forward / Action Item:

RXA PCMI cladding failure curves revised.

Revised RG Text:

TOPIC: **Fuel Rod Cladding Failure Thresholds**

DG-1327 Draft Text:

3.3 Molten Fuel Cladding Failure Threshold
Fuel cladding failure is presumed if predicted fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions.

Number of Comments: 2

Comment:	NRC Response:
a) The molten fuel cladding failure threshold assumes cladding failure upon incipient fuel melting. For ductile cladding this assumption is not valid for the 10% volumetric molten fuel core coolability limit at the centerline. [NEI-A13]	a) The NRC disagrees with this comment. The applicant is free to provide justification to support an alternate position that cladding integrity is preserved with limited fuel centerline melting. But that justification is beyond the scope of this guidance. An allowance for alternate cladding failure thresholds was added to the guidance.
b) Fuel melting does not always induce cladding failure. NUREG 0800 Section 4.2 states that "The centerline melting criterion was established to assure that axial or radial relocation of molten fuel would neither allow molten fuel to contact the cladding nor produce local hot spots. The assumption that centerline melting results in fuel failure is conservative. " [A-20]	

Resolution / Path Forward / Action Item:

Alternate failure threshold text added to Section 3.

Revised RG Text:

Alternative fuel rod cladding failure criteria may be used if they are adequately justified by analytical methods and supported by sufficient experimental data. Alternative cladding failure criteria will be addressed on a case-by-case basis.

TOPIC: **Fission Product Release Fractions**

DG-1327 Draft Text:

4. Fission Product Release Fractions
The total fission product fraction available for release following any event would include the steady-state fission product gap inventory (present before the event) plus any fission gas released during the transient. Whereas FGR (into the rod plenum) during normal operation is governed by diffusion, pellet fracturing and grain boundary separation are the primary mechanisms for FGR during the transient.

The empirically based transient FGR correlation is shown in Figure 6. The empirical database suggests that transient FGR is sensitive to both local fuel burnup and peak radial average fuel enthalpy rise. As a result, separate low burnup and high burnup transient FGR correlations are provided as a function of peak radial average fuel enthalpy rise (Δcal/g).

An investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release (Ref. 13) concluded that adjustments to the empirically based correlations are needed for different radionuclides.

4.1 For stable, long-lived isotopes (e.g., Kr-85), the transient fission product release is equivalent to the burnup-dependent correlations provided in Figure 6.

4.2 For Cs-134 and Cs-137, the transient fission product release correlations provided in Figure 6 should be multiplied by a factor of 1.414.

4.3 For volatile, short-lived radioactive isotopes such as iodine (i.e., I-131, I-132, I-133, I-135) and xenon and krypton noble gases except Kr-85 (i.e., Xe-133, Xe-135, Kr-85m, Kr-87, Kr-88), the transient fission product release correlations provided in Figure 6 should be multiplied by a factor of 0.333.

4.4 The transient fission product release fractions must be added to the steady-state fission product inventory for each radionuclide (present before the event) to obtain the total radiological source term for dose calculations. Additional fission product releases from fuel melting may need to be included in total radiological source term. See RG 1.183 for steady-state fission product gap inventories and further guidance.

Number of Comments: 12

Comment:	NRC Response:
a) Fission product release fraction guidance and radiological consequence related guidance in should be moved to existing Regulatory Guides 1.183 and 1.195 to be consistent. [NEI-3, D-5, SCEG-4, GE-13]	a) The NRC agrees in part with this comment. The fission product release fraction guidance should reside in RG 1.195 and 1.183. Unfortunately, it will take addition calendar time to revise these RGs. To avoid delay in fully implementing the revised guidance for CRD and CRE, the staff has decided to place this information in an appendix. Upon completion of the RG 1.195 and 1.183 revisions, this appendix will be deleted.
b) The fission product release fractions content and the radiological consequences content should be deleted from DG-1327. The industry references for these subjects include RG 1.183, RG 1.195, and NUREG-0800, and any revisions should be to those documents. [NEI-A5]	
c) Fission product release fractions should remain in RG 1.183 and RG 1.195. Section 4 should be removed from DG-1327. Information related to the performance of radiological consequence analyses should remain in RG 1.183 or equivalent guidance. Not every licensee has adopted RG 1.183, so RG 1.195 guidance remains applicable. Section 4 introduces many conflicts with NRC approved implementation of AST (RG 1.183 and DG 1.199). Clarification is also needed on the applicability of ANS 5.4. The proposed radiological consequence limits may conflict with the licensing bases for some licensees.[APS-6]	
d) If the NRC is pursuing a revision to RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," in the relatively near future, Exelon believes it would not be prudent to provide guidance related to the radiological inventory in the gap region and dose consequences associated with fuel failure. The guidance in Sections 4 and 5 of DG-1327 should be relocated to the next revision of RG 1.183 and RG 1 .195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors." [Exelon-4]	
e) DG-1327 Section 4.4 makes reference to Regulatory Guide 1.183 for the steady-state fission product gap inventories. The new REA regulatory guide (DG-1327) should include these steady-state fission product gap inventories. [Perez-1]	b) The NRC agrees with this comment. Steady-state fission product gap inventories will be added to the new appendix.
f) As currently stated, it is unclear whether this multiplier should also be applied to other alkalis (i.e., Rb isotopes). [A-21]	c) The NRC agrees with this comment that further clarification may be necessary. There is no further information available to support treating the various alkali metal isotopes differently. As such, the 1.414 multiplier which reflects the ANS 5.4 standard recommendation for Cs (relative to nobles) will be applied to all nuclides within this group.

g) As currently stated, it is unclear whether this multiplier should also be applied to other halogens (i.e., Br isotopes). [A-22]	d) The NRC agrees with this comment that further clarification may be necessary. The investigation into the effect of differences in diffusion coefficients and radioactive decay on fission product transient release (Ref. 13) concluded that adjustments to the empirically based correlations are needed for different radionuclides. For the short-lives isotopes, the investigation compared the predicted release of I-131 (R/B) to the stable long-lived isotopes and estimated a factor of 6 to 15 difference depending on power level (temperature) and burnup. Given the lack of specific data on short-lived releases under transient conditions, a conservative multiplier of 1/3 was selected. I-131 with a half-live of 8.04 days is more stable and less likely to decay during transport to the grain boundary than the other halogen. For example, Bromine has many short-lived isotopes, with Br-77 being the most stable with a half-life of 57.04 hours. As such, the 1/3 multiplier should be applied to all of the short-lived halogens.
h) The steady-state fission product gap inventories differ between RG 1.183, DG-1199 Revision 1, PNNL-18212 Revision 1 and RG 1.195. It is noted that RG1 .183, DG-1199 and PNNL-18212 all have different release fractions for RIA events. This specific guidance may not be consistent with the guidance used in a plants actual licensing or cause uncertainty with respect to the actual release inventories to be used. Table 3 of RG 1.183 contains a footnote indicating higher release fractions for RIA. [A-23]	e) The NRC agrees with this comment. Steady-state fission product gap inventories will be added to the new appendix. The RG 1.183 Table 3 footnote no longer applies and is superseded by the transient release component.
i) The basis for the internal pin pressure of many of the RIA test rods is configured so that the desired differential pressure between the pin and system pressure is obtained rather than restoring the original pin pressure. Hence, the internal gas pressures and elemental species in the remanufactured test pins are atypically much lower than the original rod. Heating this rod by any means would result in fission gas release, not just an RIA. Attributing all the fission gas release to the enthalpy deposited is a conservative approach. This level of conservatism of the approach using the fission gas release of the test data is not mentioned nor identified in the literature. Comment provided to express the conservatisms of this approach. [A-32]	f) The NRC agrees with this comment in that it is conservative to attribute all of the FGR with the transient. No changes necessary.
j) RG 1.183, rev. 0 does not use the terminology "steady-state" fission product gap inventory for the control rod drop or rod ejection accident. Once finalized into a Regulatory Guide (the guidance in DG-1327), as proposed, the new regulatory guidance will coexist with RG 1.195 and RG 1.183. Neither RG 1.195 nor 1.183 discuss the fission gas released during the transient. The conflicting guidance will cause confusion. Ideally, the guidance in DG-1327 regarding the fission product release should be incorporated into RG 1.195 and 1.183 rather than in DG-1327. The steady-state fission product release guidance in DG-1327 conflicts with those provided in DG-1199. DG-1327 points to in RG 1.183 and RG 1.183 states that 10% of the core inventory of noble gases and iodines is in the gap. In DG-1199 the steady-state fission gas in the gap for noble gases and iodines is from 4% to 35% of the core. Clarify DG-1327 with the following text to resolve the above issues so that improper guidance is not used. Also, a note should be added to the recommended table below to state that the calculated values of transient and steady-state combined releases are limited to a value of 1.0. [DRA/ARCB-2] 4.4 Until RG 1.195 and 1.183 can be updated with revised guidance on the steady-state and transient gap fission product fraction available for release following a control rod ejection or control rod drop accidents, the following guidance should be used. The transient fission product release fractions discussed above in DG-1327 must be added to the steady state gap activity for each radionuclide (present before the event) to obtain the total radiological source term for control rod ejection or control rod drop accident dose calculations. Revised steady-state gap activities for the control rod ejection and control rod drop are given below. See RG 1.195 and 1.183 for further guidance regarding which radionuclides are included in the analyses. [The values below are taken from the draft final version of RG 1.183 derived from DG-1199]	g) The NRC agrees with this comment. Appendix B added to capture both steady-state and transient releases, until RG 1.183 and RG 1.195 can be updated.

Table X. Steady-State Fraction of Fission Product Inventory Available for Release from Reactivity Initiated Accidents

Group	Steady State Release Fraction·
I-131	0.08
I-132	0.09
Kr-85	0.38
Other Noble Gases	0.084
Other Halogens	0.05
Alkali Metals	0.50

Resolution / Path Forward / Action Item:

Transient fission product releases removed from main body. Appendix B added with both steady-state and transient fission product inventories.

Revised RG Text:

See Appendix B.

TOPIC: **Allowable Limits on Radiological Consequences**

DG-1327 Draft Text:

5. Allowable Limits on Radiological Consequences
The offsite radiological consequences should be limited to "well within" the guidelines in 10 CFR Part 100, "Reactor Site Criteria," except for plants that adopt the alternate source term, which will be limited to "well within" the guidelines in 10 CFR Part 50.67. The term "well within" equates to 25 percent of allowable limits. For example, the allowable radiation dose for an individual located on the boundary of the exclusion area for any 2-hour period would be 6.25 rem total effective dose equivalent (TEDE) (equivalent to 25 percent of 25 rem TEDE prescribed in 10 CFR 50.67(b)(2)(i)). See RG 1.183 for further guidance.

Number of Comments: 2

Comment:	NRC Response:
a) Remove this paragraph and refer to RG 1.183 and RG 1.195 for dose considerations. [GE-14]	a) The NRC agrees with these comments. The paragraph will be revised to simply refer to the two RGs.
b) RG 1.195 also provides guidance and should be referenced here. [A-30]	
c) Title 10 of the Code of Federal Regulations, Part 100 and 10 CFR 50.67 provide "reference values" to evaluate the proposed design basis. While a very few SE's have carried forward the "well with" terminology from plants licensed under Part 100, the NRC guidance (Regulatory Guide (RG) 1.183 and 1.195) does not contain the "well within" terminology and the NRC staff no longer uses it. Reference to the "well within" terminology should be removed. The text should also reference the regulations applicable for new reactor designs. The following recommended text was adapted from DG-1199 and RG 1.195. It incorporates acceptance criteria for 10 CFR 50.67 and 10 CFR 100 plants as well as for new reactor applicants. [DRA/ARCB]	
5. Acceptance Criteria for Radiological Consequences The accident dose radiological consequences criteria for the exclusion area boundary (EAB) and the outer boundary of the low population zone (LPZ) are given in 10 CFR 50.34, 10 CFR 50.67, 10 CFR Part 52, and 10 CFR Part 100.11, as applicable to the type of application. The accident dose radiological consequences criteria for power reactor control room habitability are given in General Design Criterion 19 and 10 CFR 50.67. The offsite criteria are stated for evaluating reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation, e.g., a large-break loss of coolant accident. For the CRD and CRE accidents, that have a higher probability of occurrence, the postulated EAB and LPZ doses should not exceed 6.3 rem whole-body or 75 rem thyroid dose (licensees using 10 CFR 100.11) or 6.3 rem total effective dose equivalent (licensees using 10 CFR 50.34, 10 CFR 50.67, and 10 CFR Part 52). These stated criteria for the CRD and CRE accidents are the same criteria provided in Regulatory Guide 1.195 (Ref. 4) and Regulatory Guide 1.183 (Ref. 5).	

Resolution / Path Forward / Action Item:
Revise text.

Revised RG Text:

4. Allowable Limits on Radiological Consequences
The accident dose radiological consequences criteria for CRD and CRE accidents are provided in Regulatory Guide 1.183 (Ref. 4) and Regulatory Guide 1.195 (Ref. 5).

TOPIC: **Allowable Limits on Reactor Coolant System Pressure**

DG-1327 Draft Text:

6. Allowable Limits on Reactor Coolant System Pressure
The maximum reactor coolant system pressure should be limited to the value that will cause stresses to not exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code (Ref. 14).

Number of Comments: 3

Comment:	NRC Response:
a) The content on maximum reactor coolant pressure should be deleted to prevent conflicts with the current licensing basis as specified in FSARs. [NEI-A6] b) Allowable limits on reactor coolant system pressure are already specified in a plant's Final Safety Analysis Report and may differ from the limit defined in DG-1327. Therefore, specific acceptance criteria should not be provided for the reactor coolant maximum pressure. [APS-7] c) The reactor coolant peak pressure acceptance criterion is already defined in a plant's Final Safety Analysis Report and may differ from the limit defined in DG-1327. The Regulatory Guide should not override existing licensed limits. [SCEG-5]	a) The NRC agrees, in part, with this comment. Inclusion of the ASME Emergency Condition allowable stress limit is consistent with GDC-28 and RG 1.77 (which this RG replaces). This guidance will be retained for future reactor licensing. The staff agrees that with the adoption of this guidance, licensees are not expected to alter their existing design basis stress and strain limits for their reactor coolant pressure boundary. Text will be added. Licensees may be required to adopt, as necessary, changes to their analytical methods and assumptions (e.g., PCMI cladding failure, transient fission gas release, at-power analyses) as reflected in this guidance. See Section D.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

5. Allowable Limits on Reactor Coolant System Pressure
For new license applications, the maximum reactor coolant system pressure should be limited to the value that will cause stresses to not exceed Emergency Condition (Service Level C), as defined in Section III of the ASME Boiler and Pressure Vessel code (Ref. 14). For existing plants, allowable limits for the reactor pressure boundary as specified in the plant's UFSAR should be maintained.

TOPIC: **Allowable Limits on Damaged Core Coolability**

DG-1327 Draft Text:

7. Allowable Limits on Damaged Core Coolability

7.1 The limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-coolant interaction (FCI) is an acceptable metric to demonstrate limited damage to core geometry and that the core remains amenable to cooling.

7.2 The following restrictions should be met:

7.2.1 Peak radial average fuel enthalpy must remain below 230 cal/g.

7.2.2 A limited amount of fuel melting is acceptable provided it is restricted to the fuel centerline region and is less than 10 percent of pellet volume. The peak fuel temperature in the outer 90 percent of the pellet's volume must remain below incipient fuel melting conditions.

For fresh and low-burnup fuel rods, the peak radial average fuel enthalpy restriction will likely be more limiting than the limited fuel centerline melt restriction. However, because of the effects of edge peaked pellet radial power distribution and lower solidus temperature, medium- to high-burnup fuel rods are more likely to experience fuel melting in the pellet periphery under prompt power excursion conditions. For these medium- to high-burnup rods, fuel melting outside the centerline region must be precluded, and this restriction will likely be more limiting than the peak radial average fuel enthalpy restriction.

Number of Comments: 2

Comment:	NRC Response:
a) It is not clear how this guidance applies to annular pellets. The definition above is clear for a solid cylindrical pellet. For annular fuel, there is no centerline region. [A-24]	a) The NRC agrees with this comment. Proposed text adopted.
b) There are two subsections 7.1 and 7.2 which explain the same topic. Using two sections suggests there are two different concerns and reduces clarity. [A-31]	b) The NRC agrees with this comment. Text revised.

Resolution / Path Forward / Action Item:

Text revised.

Revised RG Text:

6. Allowable Limits on Damaged Core Coolability

Limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-coolant interaction (FCI) is an acceptable metric to demonstrate limited damage to core geometry and that the core remains amenable to cooling. The following restrictions should be met:

a) Peak radial average fuel enthalpy must remain below 230 cal/g.
b) A limited amount of fuel melting is acceptable provided it is restricted to less than 10 percent of fuel volume. The peak fuel temperature in the outer 90 percent of the fuel volume must remain below incipient fuel melting conditions.

For fresh and low-burnup fuel rods, the peak radial average fuel enthalpy restriction will likely be more limiting than the limited fuel melt restriction. However, because of the effects of edge peaked pellet radial power distribution and lower solidus temperature, medium- to high-burnup fuel rods are more likely to experience fuel melting in the pellet periphery under prompt power excursion conditions. For these medium- to high-burnup rods, fuel melting outside the centerline region must be precluded, and this restriction will likely be more limiting than the peak radial average fuel enthalpy restriction.

TOPIC: **Implementation**	
DG-1327 Draft Text:	
Number of Comments: 8	
Comment:	**NRC Response:**
a) The proposed implementation process may invoke the DG-1327 guidance when a licensee submits a voluntary license amendment request (LAR) that "involves a regulatory issue directly relevant to this new or revised guide". This language needs clarification so that the large expense to transition to the new guidance is not required for insignificant changes. There is also exposure to the industry to inconsistent application of this language by the NRC. The industry proposes continued dialog with the NRC staff to develop criteria that will achieve reasonable future regulatory expectations for compliance.[NEI-5]	a) The NRC agrees that continued dialogue would be beneficial to define a clear implementation plan.
b) Implementation of the final RG guidance is only applicable to burnup extensions LARs.[NEI-A24]	b) The NRC disagrees with this comment. Burnup extension is but one example of a significant change which would trigger adoption of this guidance.
c) The language of DG-1327 in Sections A, Introduction, and D, Implementation, should be revised to indicate the continued acceptability of legacy reactor kinetic methodologies (e.g., zero-point or one-dimensional spatial kinetics). [NEI-A26]	c) The NRC agrees with this comment. Legacy models and methods may continue to be used. However, once triggered, changes may be necessary to the application methodology to capture the updated guidance.
d) The technical content in a voluntary LAR invoking DG-1327 guidance needs to be specified. [NEI-A27]	d)

e) The quote, "If an existing licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised regulatory guide," is the trigger for NRC to impose the guidance. APS suggests replacing "regulatory issue directly relevant" with a list of specific types of LARs (e.g., power uprate, fuel design transition, etc.) that will impose the guidance.	e) The NRC agrees with this comment. Specific examples would be helpful.
	
	First cut:
	  a. Power uprate
	  b. Fuel design change (with existing fuel vendor or to a different fuel vendor)
	  c. Burnup extension
	  d. Increase in U235 enrichment
	  e. Change in operating domain (e.g., MELLA+, load follow)
	  f. Change in reload cycle length (e.g., 18 to 24 month reload cycle)
	  g. Change in PDIL
	  h. Change in CRE/CRD dose calculations (e.g., AST)
	  i. Change in CRD/CRE analytical methods (e.g., migrate to 3D kinetics)
	  j.

f) If an analysis is updated based on cycle-specific parameters, which is routine for BWRs and not so routine for PWRs, it is typically performed under the 10 CFR 50.59 process. Exelon requests further clarification as to whether the change would be considered a change to the current licensing basis. [Exelon-9]	f)
g) Section D, Implementation, of DG-1327 allows the NRC staff to broadly interpret when a licensee may be required to comply with the guidance of DG-1327 or an equivalent alternative process without the need for a back fit analysis. As such, DG-1327 should be reviewed by the CRGR for consistency with regulatory policy. [D-1]	g) The NRC agrees with this comment. The finalized version of DG-1327 will be reviewed by the CRGR, per NRC procedures.
h) The guidance provided by the staff in Section D, Implementation, contains the following text: "Current licensees may continue to use guidance the NRC found acceptable for complying with the identified regulations as long as their current licensing basis remains unchanged." It is understood that plant changes such as stretch or extended power uprates, fuel burnup extensions, or use of transient, three-dimensional, core simulation codes within the safety analysis would naturally lead to the imposition of the requirements in DG-1327. However, the text in DG-1327 is overly restrictive with regards to the use of the	h)

conservative zero- (point) or one-dimensional spatial kinetics for the analysis of reactivity initiated accidents. These conservative methods should continue to be acceptable when applying the guidelines of RG 1. 77. This is supported by the NRG staff reports since 2004. Specifically, the NRG staff performed an assessment of postulated reactivity-initiated accidents for operating reactors in the US in Research Information Letter 0401, dated March 31, 2004, that concluded there was no concern related to protecting the health and safety of the public for the operating reactors. The NRC has issued two memorandums (dated January 17, 2007 and March 16, 2015) on the proposed technical and regulatory basis for reactivity-initiated accident acceptance criteria since the 2004 assessment. The two memorandums reference the 2004 safety assessment. As such, Dominion recommends that DG-1327 should not be applied to licensees that continue to employ FSAR Chapter 14/15 Safety Analysis methodologies that make use of the conservative zero- (point) or one-dimensional spatial kinetics for the analysis of reactivity-initiated accidents. [D-6]	
Resolution / Path Forward / Action Item:	
Revised RG Text:	

Attachment 1: **RXA** PCMI Cladding Failure Threshold Curves

Framatome (formerly AREVA) provided a comment to extend the RXA PCMI cladding failure curves (Comment A-4). DG-1327 had truncated the RXA PCMI curves at 300 wppm excess hydrogen due to the extent of the database. The commenter provided more recent RIA test data including NSRR test VA-6 (UO2 fuel rod, MDA RXA cladding, 80 GWd/MTU, and 708 wppm hydrogen content). VA-6 failed at a reported fuel enthalpy of 34 cal/g. The commenter proposed revised curves, adding a linear relationship between the last two failed tests at 300 wppm and 708 wppm, and extending out to 800 wppm excess hydrogen.

Instead of a broken linear relationship, the staff elected to draw a curve fit through the data, including the new VA-6 failure data point. This approach is similar to the SRA failure threshold curves. The problem is that there is no reported failure data below 150 wppm excess hydrogen to capture the expected change in slope. To address this lack of data, the staff decided to treat the highest enthalpy, highest hydrogen content fuel rod segment to survive as a failure point. FK-3 (non-failed, 72 wppm, 150 cal/g) is shown below as a red symbol.

The plots below illustrate the curve fit, along with the empirical database (scaled for hot conditions), original DG-1327 failure threshold and commenter proposed failure threshold. A power function provided the best fit to the RXA data, as opposed to the logarithmic function used for the SRA data. This difference is not unexpected given the higher sensitivity RXA cladding exhibits to zirconium hydrides (due to the greater number of radial oriented hydrides).