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ATTN: Program Management, Announcements and Editing Staff

**Subject:** Industry Comments on Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents" (Federal Register 84FR49125, dated September 18, 2019 and 7590-01-P, dated September 18, 2019, Docket ID NRC-2016-0233)

**Project Number: 689**

Dear Program Management, Announcements and Editing Staff:

The Nuclear Energy Institute (NEI),<sup>1</sup> on behalf of the nuclear industry, appreciates the opportunity to provide comments for NRC staff consideration on the subject draft regulatory guide, DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents", as requested by the subject Federal Register Notices (FRN 84FR49125 and 7590-01-P). Our high priority comments are summarized in this letter below and detailed comments are provided in the attachment.

The industry comments on DG-1327 presented in this document are made on behalf of NEI members and are applicable to PWR and BWR fuel design, core design, and other aspects related to the PWR control rod ejection accident, and the BWR control rod drop accident. Our comments reflect our member's concerns regarding the technical and regulatory guidance, the technical and regulatory bases for the guidance, and implementation of new analytical methodologies and design basis analyses of record that could require significant cost to implement without commensurate safety benefit.

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<sup>1</sup> The Nuclear Energy Institute (NEI) is the organization responsible for establishing unified industry policy on matters affecting the nuclear energy industry, including the regulatory aspects of generic operational and technical issues. NEI's members include all entities licensed to operate commercial nuclear power plants in the United States, nuclear plant designers, major architect/engineering firms, fuel cycle facilities, nuclear materials licensees, and other organizations and entities involved in the nuclear energy industry.

While detailed comments are provided in the attachment to this letter, the following summarizes the higher-priority industry comments on the draft regulatory guidance in DG-1327:

- The reactivity-initiated event (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 (CRE) or BWR control rod drop (CRD) 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. Our detailed comments beginning on page 13 of the attachment describe EPRI test programs and analyses performed to address the effects of temperature and pulse width leading to the proposal of more appropriate cladding failure thresholds.
- Fission product release fraction guidance and radiological consequence related guidance provided in Appendix B of DG-1327, should be moved to existing Regulatory Guides 1.183 and 1.195 for consistency.

We appreciate the staff's consideration of these comments and trust that they will be found useful and informative as you proceed to finalize this guidance. We would be pleased to answer any comments or questions you might have on the contents of this letter as well as for scheduling future public interactions. I may be contacted at [fap@nei.org](mailto:fap@nei.org) or 202-739-8132.

Sincerely,



Frances Pimentel

c: Mr. Paul Clifford, NRR/DSS, NRC  
Mr. Edward O'Donnell, NRR/DSS, NRC

Attachment



# Industry Response to Draft Regulatory Guide DG-1327 issued July 2019

**November 2019**

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## Acknowledgements

This report was developed with the hard work of multiple participants. We would like to thank all the individuals and group members who joined in the effort:

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Westinghouse

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## Executive Summary

The Nuclear Energy Institute, Inc. (NEI) on behalf of the industry is pleased to offer these consolidated comments on the draft regulatory guide DG-1327, “Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents, dated July 2019 (Ref. 1). This draft regulatory guide was distributed for public comment in the Federal Register Notice, per References (Refs.) 2 and 6.

The attached comments fall into two categories: 1) statements of the industry position regarding the NRC responses to the industry comments on the original draft of DG-1327 that were forwarded to the NRC by NEI letter dated April, 21 2017 (Ref. 3), and 2) new industry comments on DG-1327 Revision 1.

Among all the comments, those related to RG 1.183 (comments 21 and 23) are of the most concern to the industry at this time.

The industry requests the NRC staff consider the attached comments in the development of the final regulatory guide.

## List of Acronyms

ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
AST	Alternate Source Term
BOC	Beginning-of-Cycle
BTD	Brittle-to-Ductile
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRD	Control Rod Drop
CRDA	Control Rod Drop Accident
CRE	Control Rod Ejection
DG	Draft Guide
EPRI	Electric Power Research Institute
FGR	Fission Gas Release
FRN	Federal Register Notice
GE	General Electric
GEH	General Electric Hitachi
GNF	Global Nuclear Fuel
GWd	Gigawatt Days
JAEA	Japan Atomic Energy Agency
LOCA	Loss-of-Coolant Accident
LWR	Light Water Reactor
MTU	Metric Tons Uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
PCMI	Pellet-to-Cladding Mechanical Interaction
PNNL	Pacific Northwest National Laboratory
ppm	Parts per Million
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RG	Regulatory Guide
RIA	Reactivity Initiated Accident
RXA	Recrystallized Annealed
SRA	Stress Relief Annealed

## References

1. DG-1327, Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents, USNRC Draft Regulatory Guide, July 2019, ML18302A106.
2. FRN 84-146, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, Nuclear Regulatory Commission Federal Register Notice, pages 36961 thru 36963, publication dated July 30, 2019.
3. "Industry Comments on Draft Regulatory Guide-1327", letter, Stephen E. Geier (NEI) to C. Bladey (NRC), April 21, 2017.
4. DG-1327, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, USNRC Draft Regulatory Guide, November 2016, ML16124A200.
5. Response to Public Comments Draft Regulatory Guide (DG)-1327, (NRC Docket 2016-0233), ML18302A107.
6. FRN 84-181, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents, Nuclear Regulatory Commission Federal Register Notice, page 49125, publication dated September 18, 2019.

## Background

The U. S. Nuclear Regulatory Commission (NRC) has published draft DG-1327, ““Pressurized-Water Reactor Control Rod Ejection and Boiling-Water Reactor Control Rod Drop Accidents” (Ref. 1). Reference 1 is a revision to the original draft DG-1327 (Ref. 4). Industry comments on the original draft were submitted by NEI in Reference 3. Reference 1 includes revisions to the original draft based on public comments including industry comments, and also includes new content originated by NRC staff. The NRC documented their disposition of the public comments in Reference 5.

## Purpose

The purpose of this document is to provide industry comments on the revision to DG-1327. The industry requests that the NRC staff consider the attached comments in the development of the final regulatory guide.

## Overview

The attached comments fall into two broad types. Type 1 comments concern previous industry comments on the original draft DG-1327 (Ref. 4) forwarded to the NRC by NEI letter dated April, 21 2017 (Ref. 3). Type 2 comments are simply new relative to the publication of the latest DG-1327 Revision per Reference 1.



## Comment 1 (Type 2)

**TOPIC - Background****DG-1327 Draft Text**

Section B, page 5, first paragraph of Background:

*In 2015, the staff evaluated newly published empirical data and analyses and identified further changes to guidance in the NRC memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1". This memorandum, as amended by public comments documents the empirical database, as well as the technical and regulatory bases for this guide. To reflect the latest state of knowledge, this guide presents that information.*

**Industry Comment**

Characterization of public comments in the Background section implies the public comments were made on the NRC memorandum supporting the technical and regulatory basis which is not appropriate. The public comments were provided on the initial DG-1327(Reference 4). Please remove text indicating it was amended by public comments as shown below.

*".....This memorandum documents the empirical database, as well as the technical and regulatory bases for this guide."*

## Comment 2 (Type 2)

**TOPIC: Limits on Applicability****DG-1327 Draft Text**

Section C.1, page 7, paragraph 1:

*The analytical limits and guidance described may not directly apply to anticipated operational occurrences (AOOs) and other postulated accidents involving positive reactivity insertion (e.g., PWR excess load, PWR inadvertent bank withdrawal, PWR steamline rupture, BWR turbine trip without bypass, BWR rod withdrawal error). Furthermore, depending on design features, reactor kinetics, and accident progression, this guide may not apply directly to advanced light-water reactors (LWRs) and modular LWRs. The staff will consider application of this guide beyond PWR CRE and BWR CRD, as well as the range of applicability described below, on a case-by-case basis.*

**Industry Comments**

1) In the NRC comment response ([Reference 5, Page 6](#)) Staff's "Revised RG text" does not appear to have been implemented in the revised DG.

The NRC had provided Revised RG Text in the response to comments from the first public comment period that was not incorporated in to the DG posted for the second public comment period. The Revised RG Text provided in the NRC response read as:

*"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)."*

Please incorporate the revised NRC RG Text as indicated above into Section C.1, page 7, paragraph 1.

2) Also, in Appendix B, replace all instances of the term "Non-LOCA" with "RIA", as events other than RIA are not germane to the Regulatory Guide.

## Comment 3 (Type 1)

<b>TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions</b>
<b>DG-1327 Draft Text</b>
<p>Section C.2.1.1, page 8: <a href="#">(Reference 4, DG-1327, November 2016)</a></p> <p><i>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.</i></p>
<b>Original Industry Comment NEI-A9 with Respect to Reference 4</b>
<p>RG 1.203 is not mentioned in DG-1327. To clarify non-applicability of RG 1.203 some clarification should be added.</p>
<b>NRC Reply to Industry Comment, Reference 5</b>
<p><u>NRC Response</u></p> <p>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.</p> <p><u>Resolution</u></p> <p>Text revised for other comments</p> <p><u>Revised RG Text</u></p> <p>Section 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</p>

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.

### Industry Position

The original public comment on DG-1327 (Reference 4) was with regards to general issue of methods development, not any specific, existing Vendor method.

The NRC needs to clearly establish the relationship between DG-1327 and RG 1.203. The NRC has stated DG-1327 describes methods and procedures considered acceptable when analyzing a postulated CRE or CRD which are the design basis accidents for reactivity initiated accidents

RG 1.203 describes a process considered acceptable for use in developing and assessing evaluation models used to analyze transient and accident behavior within the design basis of a nuclear power plant. It is unclear why the NRC doesn't clarify the relationship between DG-1327 and RG 1.203.

NRC has included sufficient guidance within DG-1327 Section C.2 on the analytical inputs, assumptions, and methods required for an approach to be acceptable when evaluating the postulated CRE and CRD accidents. It is requested the NRC indicate RG 1.203 does not need to be applied when the guidance of DG-1327 is employed for the evaluation of postulated CRE and CRD accidents, regardless of existing Vendor models/methods.

Add the following to the end of section C.2.1.1

*Note, if the guidance provided in this section is employed for the evaluation of postulated CRE and CRD accidents, the staff recognizes that RG 1.203 does not need to be applied.*

## Comment 4 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

Section C.2.2.1.2, page 8:

*Accident analyses at zero power should encompass both (1) BOC following core reload and (2) restart following recent power operation.*

Section C.2.2.2.2, page 10:

*Accident analyses at zero-power conditions should encompass both BOC following core reload and restart following recent power operation.*

**Industry Comment**

For consistency with NRC memorandum supporting the technical and regulatory basis for RIA acceptance criteria and guidance, it is requested the references to zero power in Items C.2.2.1.2 for PWRs and C.2.2.2.2 for BWRs be updated to include hot zero power for PWRs and cold zero power for BWRs.

*For example: Accident analyses at zero power should encompass both (1) BOC following core reload hot zero power for PWRs and cold zero power for BWRs and (2) restart following recent power operation.*

## Comment 5 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

Section C.2.2.1.5, page 9:

*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 the maximum uncontrolled rod worth defined in Regulatory Position C.2.2.4 (e.g., uncontrolled rod motion at a core location adjacent to higher burnup fuel assemblies). For this reason, 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 PWR ejected rod core locations and exposure points to identify the limiting scenarios.*

**Industry Comment**

Section C.2.2.4 should be Section C.2.2.1.4

## Comment 6 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

## Section C.2.2.1.10:

*The moderator reactivity coefficients resulting from 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.*

## Section C.2.2.1.11:

*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 in predicting fuel temperatures at different power levels, should be reflected by conservative application of Doppler feedback.*

## Section C.2.2.2.10:

*The moderator reactivity coefficients resulting from 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 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*

## Section C.2.2.2.11

*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 in predicting fuel temperatures at different power levels, should be reflected by conservative application of Doppler feedback.*

**Industry Comment**

Removing terms “coefficients” and “coefficient of” with a more generic term such as “reactivity feedback”, as there are multiple ways to simulate the reactivity mechanisms within an analysis.

## Comment 7 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

## Section C.2.3.3:

*Because of 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.*

## Section C.2.4:

*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. Appendix B gives more information and guidance.*

**Industry Comment**

The segmenting of the axial length uses the word "several". It is expected that the number of axial nodes would be much larger than several. Replace "several" with "selected".



## Comment 8 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

## Section C.2.3.4:

*In the application of the PCMI cladding failure thresholds, an NRC-approved alloy-specific cladding corrosion and hydrogen uptake model should be used to predict the initial, pre-transient cladding hydrogen content. These approved models should account for 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, either directly or implicitly through the supporting database.*

## Appendix C:

*The purpose of this appendix is to provide acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys to aid in the implementation of threshold curves for hydrogen-dependent, pellet-clad mechanical interaction cladding failure. These models also are acceptable for implementing other hydrogen-dependent fuel performance requirements (e.g., emergency core cooling system) analytical limits on peak cladding temperature and integral time-at-temperature (expressed as equivalent cladding reacted and calculated using the Cathcart-Pawel correlation) as a function of pre-transient cladding hydrogen content.*

**Industry Comment**

Section C.2.3.4 states that an “NRC-approved” hydrogen uptake model should be used. The hydrogen uptake model in Appendix C is designated as “acceptable.” The concern is that a vendor/licensee submittal of the Appendix C hydrogen uptake model would be subject to additional NRC review. In Appendix C replace “acceptable” with “NRC-approved”.

## Comment 9 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

## Section C.2.3.7:

*For plants in which gross failure (sufficient to allow a control rod to be ejected rapidly from the core) of a control rod drive mechanism housing is not considered credible, fuel failure predictions do not need to consider any reactor coolant system depressurization resulting from a mechanistic evaluation of a ruptured control rod drive mechanism housing. If credible, it should be shown that failure of one control rod housing will not lead to failure of other control rod housings.*

**Industry Comment**

The staff added Item C.2.3.7 in response to comment AREVA-17 from the first public comment period. The comment requested clarification on the treatment of the potential pressure reduction caused by the assumed failure of the control rod pressure housing for criterion other than RCS peak pressure.

The NRC agreed with the comment and indicated the NRC staff believes the original CRE design basis should be preserved, and plant's existing license basis should be maintained (i.e., consideration of high worth rod ejections).

Additionally, comment GE-11 on the same section as comment AREVA-17 to which the NRC agreed, identified this item as only being applicable to PWRs.

Item C.2.3.7 as currently written implies the need to perform additional analyses of the control rod housing which are beyond the scope of the DG. Specifically, the NRC cited NUREG-0800, Section 3.9.4 and the requirements of GDC 14 as the basis for the additional requirements in the response to comment AREVA-17.

It is requested the NRC replace Item C.2.3.7 with the suggested text below and relocate it to Section C.2.2.1, such that there is no confusion with BWR's.

*"Fuel failure predictions do not need to consider any reactor coolant system depressurization resulting from the assumed failure of the control rod pressure housing."*

## Comment 10 (Type 2)

**TOPIC: Physics and Thermal-Hydraulics Analytical Methods and Assumptions****DG-1327 Draft Text**

Section C.2.4:

*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. Appendix B gives more information and guidance.*

**Industry Comment**

In the context of the proposed Section C.2.4 wording, to what extent will realistic rod power histories be allowed in the context of AST? It makes no physical sense to say all bundles are at 54 MWd/MTU exposure, and all the rods in the bundle are at 62 GWd/MTU. If an approved CRE/CRDA method is applied on a cycle-specific basis, is it acceptable to use cycle specific rod source terms as cycle specific rod worths are already used?

Please clarify the expectations between DG-1327 and RG1.183.

## Comment 11 (Type 2)

**TOPIC: Fuel Rod Cladding Failure Thresholds****DG-1327 Draft Text**

Section C.3:

*Conservative and bounding 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.*

**Industry Comment**

The addition of the words “Conservative and bounding” to the allowance to propose alternate fuel failure criterion creates confusion and is not consistent with the move towards more performance based requirements. Nor is it consistent with how the staff developed the limits proposed in the DG as the PCMI cladding failure thresholds are deemed to be a best-fit of the experimental data (response to comments NEI-A7 and GE-3). In evaluating the conservative and bounding nature of alternate limits, it is unclear as to how one is to make this determination. Should the alternate limits be conservative and bounding compared to the limits proposed in the DG or the experimental data supporting the alternate limits? It is recommended that the NRC use the wording from the response to comments from the first public comment period (AREVA-18) without any additional changes. The revised text from the response to comments from the first public comment period is shown below:

“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.”

## Comment 12 (Type 1)

**TOPIC: Fuel Rod Cladding Failure Thresholds****DG-1327 Draft Text**

Section C.3.2: (Reference 4, DG-1327, November 2016)

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 ( $\Delta\text{cal/g}$ ) 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).

**Original Industry Comment NEI-A1, Reference 3**

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.

**NRC Reply to Industry Comment, Reference 5**NRC Response

The NRC agrees that some of the test conditions are not typical of in-reactor conditions. However, attempts have been made to understand the influence of non-typical experimental conditions and scale, (and) as appropriate, the experimental results. In scaling the results, some of the excess conservatism has been removed. Furthermore, employing a best-fit of the failure data reduces any excess conservatism, relative to a bounding fit of the data.

Resolution

Text revised for other comments

Revised RG Text

RXA PCMI cladding failure curves revised

**Industry Position**

Key test data that defines the proposed limits were generated under conditions far from prototypical of a commercial reactor rod ejection/rod drop design basis accident. Most of the test data were generated at room temperature and extremely short pulse width. The NRC has made minor adjustments for the temperature effect based on test data at high hydrogen concentration and did not consider pulse width effects. Ductility recovery from hydride induced degradation at operating temperatures is much more pronounced at hydrogen concentrations less than 500 ppm [1-3]. Adjustments NRC made were based on test data above 600 ppm hydrogen. Another effect not considered is the brittle-to-ductile transition temperature for fuel cladding with radial hydrides. Numerous publications, including NRC sponsored research, show a brittle-to-ductile recovery temperature of less than 150°C [4-9]. The brittle-to-ductile transition temperature has been demonstrated by in-pile tests [10]. The criteria are primarily based test data generated at pulse width 4-5 ms, while typical RIA pulse widths are 25-65 ms and 45-75 ms for PWRs and BWRs, respectively [11]. The short pulse widths results in high loading rates that are detrimental to cladding ductility [3,12]/energy absorption capacity of a fuel rod and increases the BTD transition temperature [3,13]. The atypical test conditions, from which the NRC proposed limits are based, produces results not representative of commercial LWR.

1. Daum, R.S., Majumdar, S., Bates, D.W., Motta, A.T., Koss, D.A., and Billone, M.C., "On the Embrittlement of Zircaloy-4 Under RIA-Relevant Conditions", Zirconium in the Nuclear Industry: 13th International Symposium, Annecy, France, June 10-14, ASTM 1423. – Figure 7
2. Whitmarsh, C.L., "Review of Zircaloy-2 and Zircaloy-4 Properties Relevant to N.S. Savannah Reactor Design", ORNL-3281/TID-4500 (17th ed.). – Figure 9
3. Yueh, K., Karlsson, J., Stjarnsater, J., Schrire, D., Ledergerber, G., Munoz-Reja, C., and Hallstadius, L., "Fuel cladding behavior under rapid loading conditions", Journal of Nuclear Materials, 469, pp177-186, 2016.
4. Billone, M.C., Burtseva, T.A., and Yan, Y., "Ductile-to-Brittle Transition Temperature for High-Burnup Zircaloy-4 and ZIRLOTM Cladding Alloys Exposed to Simulated Drying-Storage Conditions", ML12181A238, Sep. 28, 2012 – Figure 52
5. DTB – Motta, A.T., Capolungo, L., Chen, L.Q., Cinbiz, M.N., Daytmond, M.R., Koss, D.A., Lacroix, E., Pastore, G., Smion P.C.A., Tonks, M.R., Wirth B.D., Zikry, M., "Hydrogen in Zirconium Alloys: A review", JNM, 518 (2019) 440-460
6. Kim, J.S., Kim, T.H, Kook, D.H, Kim, Y.S, "Effects of hydride morphology on the embrittlement of Zircaloy-4 cladding", JNM, 456 (2015) 235-245.
7. Bai, J.B., "Effect of hydriding Temperature and Strain Rate on the Ductile-Brittle Transition in b Treat Zircaloy-4", Journal of Nuclear Science and Technology, Vol. 33, No. 2, p. 141-146, Feb. 1996.
8. K. Yueh, J. Karlsson, W. Lees, D. Mitchell, M. Quecedo, "New techniques for the testing of cladding material under RIA conditions", Proceedings of the 2012 Water Reactor Fuel Performance Meeting, Manchester, U.K., September 2-6, 2012.
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10. Sugiyama, T., "High Burnup Fuel Behavior Under High Temperature RIA Conditions", 2010 JAEA Fuel Safety Research Meeting, Tokai, Japan, May 19-20, 2010.
11. OECD Nuclear Energy Agency state-of-the art report, "Nuclear Fuel Behavior under Reactivity-initiated Accident (RIA) Conditions, NEA/CSNI/R(2010)1

12. Jernkvist, L.O., Massih, A.R., and Rudling, P., "A Strain-based Clad Failure Criterion for Reactivity Initiated Accident in Light Water Reactors", Swedish Nuclear Power Inspectorate (SKI), SKI Report 2004:32, August 2004. – Figure 5.7 and 5.8
13. Fearnough, G.D., and Cowan A., "The Effect of Hydrogen and Strain Rate on the "Ductile-Brittle" Behavior of Zircaloy, JNM Vol. 22 (1967), p137-147.

## Comment 13 (Type 1)

<b>TOPIC: Fuel Rod Cladding Failure Thresholds</b>
<b>DG-1327 Draft Text</b>
<p>Section C.3.2: (Reference 4, DG-1327, November 2016)</p> <p>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 (<math>\Delta\text{cal/g}</math>) 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).</p>
<b>Original Industry Comment NEI-A1, Reference 3</b>
<p>The DG-1327 PCMI guidance is not applicable for BWR Zr-2 RXA cladding if criticality is restricted to <math>\geq 100^\circ\text{C}</math> (<math>212^\circ\text{F}</math>) as the cladding will be ductile..</p>
<b>NRC Reply to Industry Comment, Reference 4</b>
<p><u>NRC Response</u> 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 <math>100^\circ\text{C}</math>.</p> <p><u>Resolution</u> RXA PCMI cladding failure curves revised for other comments</p> <p><u>Revised RG Text</u> RXA PCMI cladding failure curves revised</p>
<b>Industry Position</b>
<p>Numerous publications, including NRC sponsored research, show a brittle-to-ductile recovery temperature of less than <math>150^\circ\text{C}</math> [1-6] for cladding with radial hydride components. In a 2012 NRC sponsored research report [1], the brittle-to-ductile transition was determined to be influenced by the applied stressed used to re-orient hydride. In this report, a brittle-to-ductile transition temperature of <math>125^\circ\text{C}</math> was reported for an applied hydride re-orientation stress of 110 MPa for ZIRLO and Zircaloy-4 with high hydrogen concentration. A ductile-to-brittle transition temperature of less <math>100^\circ\text{C}</math> was later presented by the same author in 2013 for M5 at lower hydrogen concentration. The reported transition temperature is consistent with <math>100^\circ\text{C}</math> determined under RIA conditions in reference [7], for</p>



pulse width greater than 10 ms. In the past a NSRR RIA test was conducted at 85C but did not show noticeable improvement in energy absorption capacity. Test data from reference [7] would indicate at the 4-5 ms pulse width the brittle-to-ductile transition temperature is higher than 100°C. The brittle-to-ductile transition temperature is well demonstrated in the LS-series of tests conduct at the JAEA NSRR. Fuel from the same parent rod was tested at room temperature and 280°C. The test conducted at room temperature, LS-1, failed at an energy deposition of 53 cal/g while LS-1, conducted at 280°C, survived a maximum energy deposition of 89 cal/g [8].

The brittle-to-ductile transition temperature of ~100C is too low for significant hydride dissolution and ductility recovery is through other mechanism. The brittle-to-ductile transition behavior is a well-documented phenomenon. The RIA simulation tests merely provide a method to load the cladding. Test results under RIA loading conditions have been produced and verifies test data at other conditions.

1. Billone, M.C., Burtseva, T.A., and Yan, Y., "Ductile-to-Brittle Transition Temperature for High-Burnup Zircaloy-4 and ZIRLOTM Cladding Alloys Exposed to Simulated Drying-Storage Conditions", ML12181A238, Sep. 28, 2012 – Figure 52
2. DTB – Motta, A.T., Capolungo, L., Chen, L.Q., Cinbiz, M.N., Daytmond, M.R., Koss, D.A., Lacroix, E., Pastore, G., Smion P.C.A., Tonks, M.R., Wirth B.D., Zikry, M., "Hydrogen in Zirconium Alloys: A review", JNM, 518 (2019) 440-460
3. Kim, J.S., Kim, T.H, Kook, D.H, Kim, Y.S, "Effects of hydride morphology on the embrittlement of Zircaloy-4 cladding", JNM, 456 (2015) 235-245.
4. Bai, J.B., "Effect of hydriding Temperature and Strain Rate on the Ductile-Brittle Transition in b Treat Zircaloy-4", Journal of Nuclear Science and Technology, Vol. 33, No. 2, p. 141-146, Feb. 1996.
5. K. Yueh, J. Karlsson, W. Lees, D. Mitchell, M. Quecedo, "New techniques for the testing of cladding material under RIA conditions", Proceedings of the 2012 Water Reactor Fuel Performance Meeting, Manchester, U.K., September 2-6, 2012.
6. Bertolino, G., Ipina, J. P., and Meyer, G., "Influence of the crack-tip hydride concentration on the fracture toughness of Zircaloy-4", JNM, Vol. 348, Issues 1-2, p. 205-212, Jan 2006.
7. Yueh, K., Karlsson, J., Stjarnsater, J., Schrire, D., Ledergerber, G., Munoz-Reja, C., and Hallstadius, L., "Fuel cladding behavior under rapid loading conditions", Journal of Nuclear Materials, 469, pp177-186, 2016.
- 8.. Sugiyama, T., "High Burnup Fuel Behavior Under High Temperature RIA Conditions", 2010 JAEA Fuel Safety Research Meeting, Tokai, Japan, May 19-20, 2010.

Comment 14 (Type 2)

**TOPIC: Fuel Rod Cladding Failure Thresholds**

**DG-1327 Draft Text**

Section C.3.2:

*Figures 2 through 5 show the empirically based PCMI cladding failure thresholds. Because fuel cladding ductility is sensitive to hydrogen content, zirconium hydride orientation, and initial temperature, separate PCMI failure curves are provided for RXA and SRA cladding types at both low initial cladding temperature conditions (i.e., below 500 degrees F down to BWR cold startup) and high initial cladding temperature conditions (i.e., at or above 500 degrees F). The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise ( $\Delta\text{cal/g}$ ) versus excess cladding hydrogen content (wppm). Excess cladding hydrogen content refers to the portion of total hydrogen content in the form of zirconium hydrides (i.e., it does not include hydrogen in solution).*

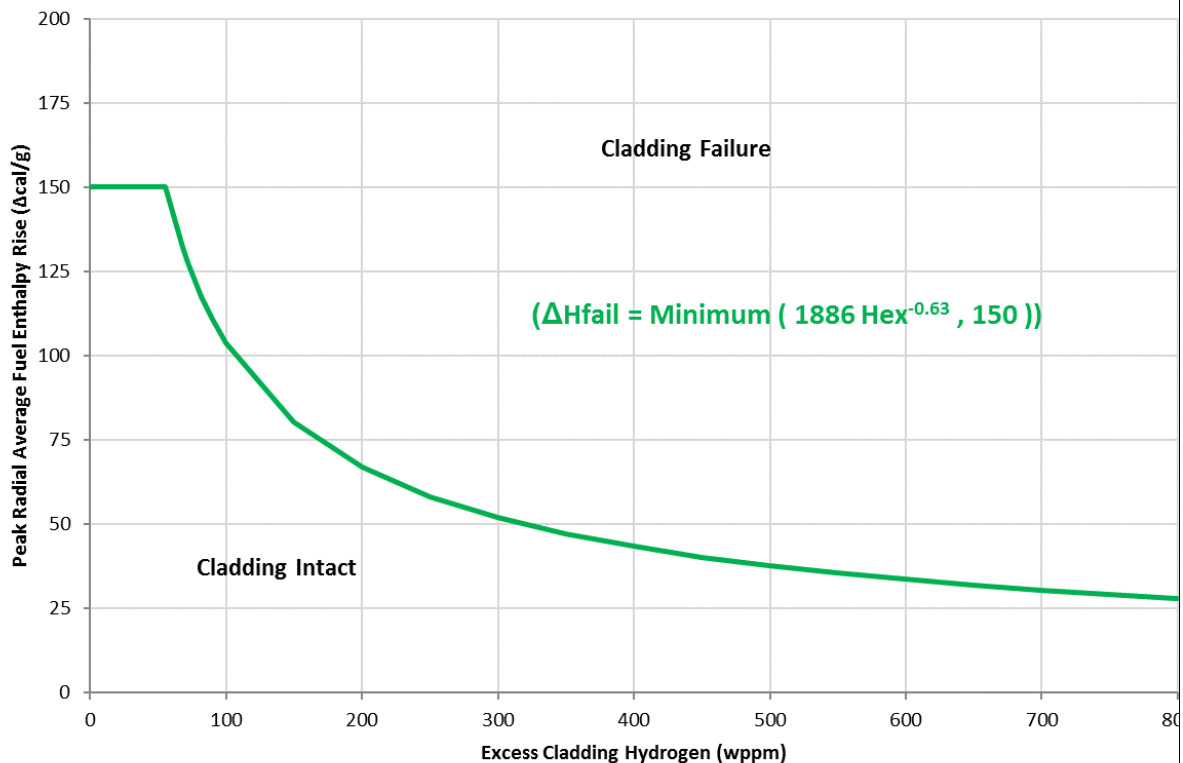


Figure 4. PCMI Cladding Failure Threshold—RXA Cladding below 500 Degrees F

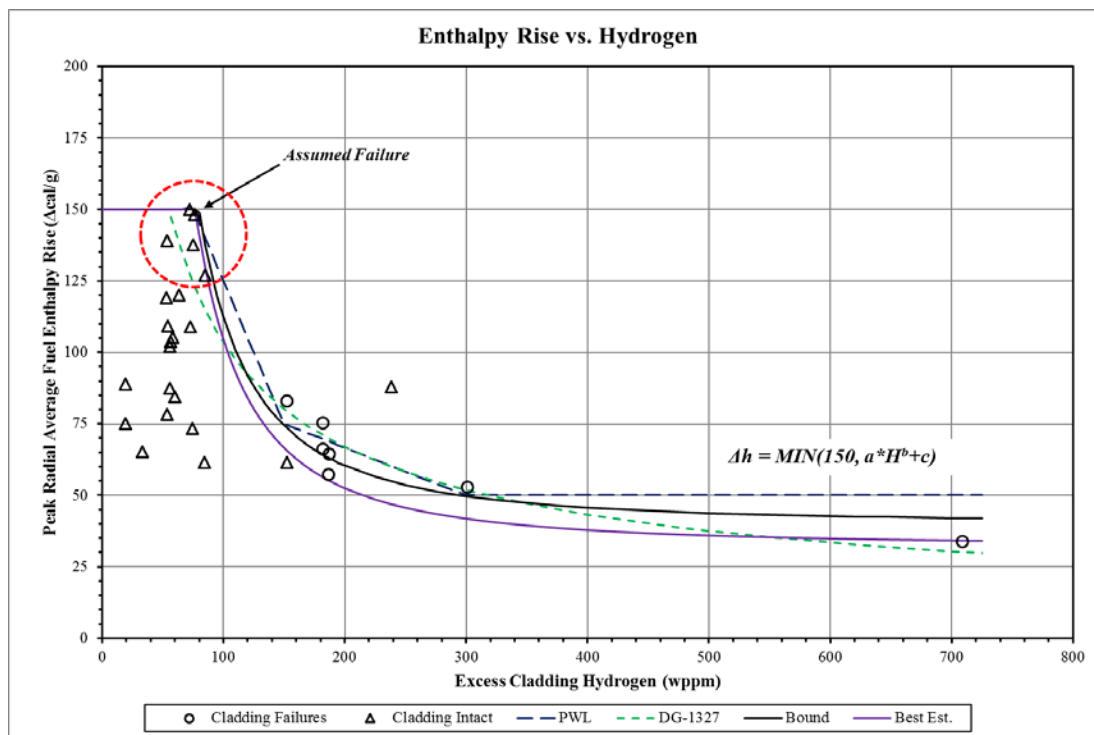
**Industry Comment**

Regarding Figure 4, the staff elected to replace the previous piecewise linear (PWL) relationship with a curve fit through the data. To facilitate the curve fitting process, it was necessary to treat the highest non-failure enthalpy/hydrogen content point (72 wppm, 150 cal/g) as a presumed failure point. This presumed failure point should serve as an anchor point for the curve fit. The current curve instead omits three other non-failure points.

The primary response from a CRDA is often from the fresh fuel (i.e. lower exposure) with highly exposed fuel reacting less energetically. Thus, the purposed failure threshold is less accurate in the area of interest particularly between 55-to-100 wppm. The figure below illustrates both a best-fit and a lower bound alternative using an exponential function.

$$\Delta h = \text{MIN}(150, a * H_b + c)$$

Where  $\Delta h$  is the enthalpy change and  $H_b$  is the excess hydrogen. In these examples, the best fit coefficients are:  $a = 3.31E+5$ ,  $b = -1.83$ , and  $c = 32$ . While the lower bound coefficients are:  $a = 3.31E+5$ ,  $b = -1.83$ , and  $c = 40$ .



Please redraw the curve to encompass more of the actual non-failure data points. While this comment directly pertains to Figure 4, the same concept applies to the other fitted curves, such as Figures 2, 3 & 5.

## Comment 15 (Type 2)

<b>TOPIC: Allowable Limits on Radiological Consequences</b>
<b>DG-1327 Draft Text</b>
Section C.4:  <i>RG 1.183 and RG 1.195 contain the accident dose radiological consequences criteria for CRD and CRE accidents.</i>
<b>Industry Comment</b>
Some licensees use 10 CFR 100 radiological consequences acceptance criteria. Revise Section 4 to include reference to 10 CFR 100 along with RG-1.183 or RG-1.195.

## Comment 16 (Type 2)

**TOPIC: Implementation****DG-1327 Draft Text**

Section D:

*The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. The NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG, unless the licensee makes a change to its licensing basis. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action that would require the use of this RG without further backfit consideration.*

**Industry Comment**

Does the content of this DG present a safety concern related to protecting the health and safety of the public for the operating reactors?

The NRC staff initially performed an assessment of postulated reactivity-initiated accidents for operating reactors in the US in Research Information Letter 0401, dated March 31, 2004. The 2004 assessment concluded that 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 continued to reference the 2004 safety assessment. Given the conclusion of the 2004 assessment and the continued reliance upon it, it is believed that NRC staff does not have a safety concern related to protecting the Health and Safety of the public for the operating reactors based on the issuance of the guidance contained in DG-1327.

## Comment 17 (Type 2)

<b>TOPIC: Implementation</b>
<b>DG-1327 Draft Text</b>
Use by NRC Staff
<b>Industry Comment</b>
<p>Include the staff requirements regarding forward fitting as defined in Management Directive 8.4 in the "Use by NRC Staff" section.</p> <p>The industry is concerned that the extensive RIA guidance in the DG will be used in the future by the NRC staff for license amendment requests that do not specifically involve RIA-related plant changes. The types of LARs that do involve RIA and DG-1327 evaluations have been identified by the staff in the NRC response to the first round of DG-1327 comments, (p. 45 Item e).</p>

## Comment 18 (Type 2)

**TOPIC: Appendix B Fission Product Release Fractions****DG-1327 Draft Text**

General Comment OR Appendix B Page B-1

*The U.S. Nuclear Regulatory Commission (NRC) memorandum titled “Revised Technical Basis for Fission Product Release Fractions,” dated June 4, 2019 (Ref. B-3), documents the derivation of the steady-state gap fractions, including the application of uncertainties. Pacific Northwest National Laboratory (PNNL) Report 18212, Revision 1, “Update of Gap Release Fractions for Non-LOCA Events Utilizing the Revised ANS 5.4 Standard,” issued June 2011 (Ref. B-4), documents an acceptable analytical method for calculating steady-state gap fractions. As an alternative to the above gap fractions, a licensee may use this analytical technique, described in the attachment, to calculate steady-state fission product gap inventories based on specific fuel rod designs or more realistic fuel rod power histories*

**Industry Comment**

Since RG 1.183 is not consistent with current codes and the consensus of fission gas gap fraction calculations, a technical basis document would be beneficial. Please revise PNNL-18212 to use the FAST code per ML19154A226.

## Comment 19 (Type 2)

**TOPIC: Appendix B Fission Product Release Fractions****DG-1327 Draft Text**

Appendix B (p. B-1)

*Table B-1. Steady-State Fission Product Inventory in Gap*

<u>Group</u>	<u>Fraction</u>
<i>I-131</i>	<i>0.08</i>
<i>I-132</i>	<i>0.06</i>
<i>Kr-85</i>	<i>0.36</i>
<i>Other Noble Gases</i>	<i>0.05</i>
<i>Other Halogens</i>	<i>0.05</i>
<i>Alkali Metals</i>	<i>0.49<sup>B1</sup></i>

**Industry Comment**

Table B-1 presents recommended steady state gap fractions documented in ML19154A226 for I-131 and other Halogens of 0.08 and 0.05, respectively. ML19154A226 reports the results of bounding FAST calculations for steady state non-LOCA gap fractions. Based on a review of the reported results in ML19154A226, and using conventional rounding techniques, appropriate gap fractions for I-131 and other Halogens would be 0.05 and 0.03, respectively. Update the gap fractions to reflect the results of ML19154A226 using conventional rounding techniques.



## Comment 20 (Type 2)

<b>TOPIC: Appendix B Fission Product Release Fractions</b>
<b>DG-1327 Draft Text</b>
Editorial changes
<b>Industry Comment</b>
<p>Page B-1 Paragraph 1 last sentence: It is confusing to refer to Appendix B within Appendix B. Please replace "Appendix B" with "this appendix".</p> <p>Page B-2 Paragraph 1 last sentence: The sentence uses the phrase "described in the attachment". Please replace "in the attachment" with "within this appendix".</p> <p>Page B-4 Paragraph 3 last sentence: Please make the following changes: While calibrated and validated against a large empirical database, FAST and its predecessors are not NRC-approved codes and may not be utilized to calculate that plant-specific, fuel-specific, or cycle-specific gap inventories that are in accordance with the acceptable analytical procedure below without further justification.</p> <p>Page B-7: Start the sample calculation on a new page.</p> <p>Page B-8: Is this page intentionally blank?</p> <p>Page B-11: Earlier in Appendix B a footnote was designated B1 on page 1. Yet, the footnotes on page B-11 are designated 1 and 2. Please adopt a consistent standard.</p>

## Comment 21 (Type 2)

**TOPIC: Appendix B Fission Product Release Fractions****DG-1327 Draft Text**

Appendix B p. B-1

*The fission product release fraction guidance contained in Appendix B for the CRE and CRD accidents should be used instead of the gap fractions provided in RG 1.183, Revision 0, for a CRE and CRD accident until RG 1.183 is updated*

**Industry Comment**

The industry is concerned the guidance in the final RG-1327 Appendix B may be subsequently changed by the NRC staff with the ongoing update to RG 1.183 and a subsequent deletion of Appendix B at a future point from DG-1327.

If that were to occur then an Appendix B-based methodology submitted by a vendor/licensee and approved by the NRC may not be consistent with the updated RG 1.183.

The industry requests the update to RG 1.183 and the deletion of Appendix B be an administrative change only, and that no technical changes are included.

The industry is also concerned that there is no indication a DG-1327 Appendix B Dose assessment is sufficient to demonstrate compliance to RG 1.183 which effectively requires use of source term values at the highest exposure limits while pin failure is being effectively tied to much lower exposures via the non-linear hydrogen uptake phenomenon.

The industry needs assurance that only ONE dose assessment is required to meet both RG 1.183, and future DG-1327 requirements.

## Comment 22 (Type 2)

<b>TOPIC: Appendix B Fission Product Release Fractions</b>
<b>DG-1327 Draft Text</b>
Figure B-1
<b>Industry Comment</b>
Please clarify the exposures discussed in the figure are pellet exposure, not rod exposure. Clearly identify exposure basis and application.

## Comment 23 (Type 2)

**TOPIC: General Comment****DG-1327 Draft Text**

Sections with “conservative” or “bounding” language

**Industry Comment**

The “conservative” or “bounding” terminology are relative terms. So, what are they relative too? Specifically Section C.2.3 is entitled “Predicting the total number of fuel rod failures”. Is the “conservative” or “bounding” terminology supposed to be with respect to the number of rods failed, or is it really supposed to be with respect to dose consequence?

When the failure criteria for a fuel rod was a constant with respect to exposure, a failed number of rods could be thought of as a surrogate for dose, and dose could be a surrogate for failed rods. The new non-linear failure criteria breaks that line of reasoning. It is possible to envision scenarios with higher dose consequence with fewer rod failures and not just from the rod eject / rod drop perspective, but from all non-LOCA events.

Please clarify the basis for DG-1327, and explicitly express what the appropriate metric is for assessing terminology such as “conservative” or “bounding”.

This issue is important with respect to how RG 1.183 comes into play. If I am doing an AST analysis defending fuel bundles at the exposure limits for source term purposes, then maybe I do want conservative/bounding choices with respect to failed rods because the source term is essentially fixed.

On the other hand, if analyses described in DG-1327 are automatically acceptable for satisfying RG 1.183, then I probably want conservative/bounding to be with respect to Dose, as not every contributing bundle/rod will be at the exposure limit of operation during the event.

## Comment 24 (Type 1)

**TOPIC: Fuel Rod Cladding Failure Thresholds****DG-1327 Draft Text**

Section C.2.1.1: ([Reference 4, DG-1327, November 2016](#))

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.

**Original Industry Comment NEI-A1, Reference 3**

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.

**NRC Reply to Industry Comment, Reference 5**NRC Response

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.

**Industry Position**

We should not confuse a statistical curve fit of data, with the nature of the test itself. A best estimate "curve fit" does not mean the proposed "limit" is best estimate, unless the experimentally derived data represent the nominal application condition. Data used for the purposes of input to the curve fit are "conservative" because the nature of the testing doesn't necessarily represent actual operating conditions. While the curve fit may be best estimate, the proposed limit is "conservative."