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April 21, 2017

Ms. Cindy Bladey
Office of Administration
Mail Stop: OWFN-12H08
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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* 81FR83288, dated November 21, 2016 and 7590-01-P, dated January 26, 2017, Docket ID NRC-2016-0233)

Project Number: 689

Dear Ms. Bladey:

The Nuclear Energy Institute (NEI)[1], 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 81FR83288 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 the this document are made on behalf of the operating fleet 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 industry 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.

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:

[1] [1] 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.

- 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) and 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 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.
- 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. Test programs performed by EPRI provide support to extend the hydrogen content to 593 ppm as discussed in our detailed comments.
- Fission product release fraction guidance and radiological consequence related guidance in should be moved to existing Regulatory Guides 1.183 and 1.195 for consistency.
- 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) that may require significant costs to the industry when considering the low probability of these hypothetical design basis events. The industry desires additional interaction with the staff through a workshop to define a sufficient and consistent set of cases commensurate with the safety case.
- 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 to ensure that the significant costs to transition to the new guidance are not required for insignificant changes. There is also concern by the industry due to the potential for inconsistent application of this language by the NRC. The industry desires additional interaction with the staff to develop criteria that will achieve reasonable regulatory expectations for future compliance.

Given the number and technical significance of our comments including the high priority comments summarized above, the industry would welcome additional staff interaction through public meetings or workshops to discuss, clarify and seek resolution on our submitted comments. We look forward to scheduling such an interaction with the NRC in the near future.

Please contact me (seg@nei.org, 202-739-8111) for additional information or questions on our comments as well as for scheduling future public interactions.

Sincerely,

Stephen E. Geier

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

Attachment

Industry Response to Draft Regulatory Guide DG-1327

Revision 0 (Final)

April 2017

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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U.S. Nuclear Fleet Utilities
Westinghouse

Executive Summary

The Nuclear Energy Institute, Inc. (NEI) on behalf of the industry is pleased to offer these consolidated comments on the proposed Draft Regulatory Guide DG-1327 published by the Nuclear Regulatory Commission (NRC) for public comment on November 21, 2016. NEI acknowledges the important work done by the subject matter experts on the Electric Power Research Institute (EPRI) Fuel Reliability Program Reg-TAC in assembling these comments.

DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents", proposes new regulatory guidance including limits that the NRC staff finds acceptable and that will replace the interim criteria that were published in Standard Review Plan Section 4.2, Appendix B, in March 2007. The proposed guidance is the culmination of research into the effects of higher fuel exposure on cladding and fuel performance during reactivity initiated accidents (RIA). These research insights were obtained primarily as a result of fuel rod RIA testing in the CABRI test facility in France and the NSRR facility in Japan that began in 1993 and remains active. The NRC technical basis for DG-1327 is an NRC memorandum dated March 16, 2015 and the associated references.

Since the release of the initial CABRI data in the 1990s the industry has been actively engaged with the NRC in discussions to understand and interpret the emerging RIA test results and their potential impact on RIA-related regulations, analysis methodologies, and licensing basis analyses. EPRI has served as the industry technical lead in performing research and analyses that have been documented in several EPRI reports. The EPRI Reg-TAC (and its predecessor committee), which includes international members as well as the domestic fuel fabrication vendors, has interacted with NRC staff for over two decades on this subject.

The higher-priority industry comments on the regulatory guidance in DG-1327 can be summarized as follows:

- 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.

  EPRI has performed test programs and performed analyses to address the effects of temperature and pulse width, and is proposing more appropriate cladding failure thresholds as described in the industry comments.

- 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.

- 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.

- 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.

- 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.

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R0	All	This is a new document

List of Acronyms

ADAMS	Agency-wide Document Access and Management System
ANS	American Nuclear Society
AOO	Anticipated Operational Occurrence
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
BOC	Beginning-of-Cycle
BS	Burst Strain
BTD	Brittle-to-Ductile
BWR	Boiling Water Reactor
CABRI	French Research Reactor
CFR	Code of Federal Regulations
CPR	Critical Power Ratio
CRD	Control Rod Drop
CRDA	Control Rod Drop Accident
CRE	Control Rod Ejection
CSED	Critical Strain Energy Density
CSNI	Committee on the Safety of Nuclear Installations
CWSRA	Cold-Worked Stress Relief Annealed
CZP	Cold Zero Power
DBE	Design Basis Event
DCD	Design Certification Document
DG	Draft Guide
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EOC	End-of-Cycle
EPRI	Electric Power Research Institute
FCI	Fuel Coolant Interaction
FGR	Fission Gas Release
FRN	Federal Register Notice
FSAR	Final Safety Analysis Report
GEH	General Electric Hitachi
GNF	Global Nuclear Fuel
GWd	Gigawatt Days
HFP	Hot Full Power
HZP	Hot Zero Power
LAR	License Amendment Request
LTTB	Loading Time to Burst
LWR	Light Water Reactor
MBT	Modified Burst Test
MOX	Mixed Oxide
MTU	Metric Tons Uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSRR	Nuclear Safety Research Reactor
OD	Outer Diameter
PCMI	Pellet-to-Cladding Mechanical Interaction
PNNL	Pacific Northwest National Laboratory

List of Acronyms

ppm	Parts per Million
PW	Pulse Width
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
REA	Rod Ejection Accident
RHLT	Rapid Heating and Loading Test
RG	Regulatory Guide
RIA	Reactivity Initiated Accident
RXA	Recrystallized Annealed
SRA	Stress Relief Annealed
SRP	Standard Review Plan
TE	Total Elongation
TID	Technical Information Document
TS	Technical Specifications
UE	Uniform Elongation
UFSAR	Update Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission

References

1. FRN 81-224, **Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents**, Nuclear Regulatory Commission Federal Register Notice, pages 83288 thru 83288, publication dated November 21, 2016.

2. DG-1327, **Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents**, USNRC Draft Regulatory Guide, November 2016

3. 3002005540, **Fuel Reliability Program: Proposed Reactivity Insertion Accident (RIA) Acceptance Criteria, Revision 1**, EPRI Technical Report, November 2015.

4. K. Yueh, J. Karlsson, J. Stjarnsater, D. Schrire, G. Ledergerber, C. Munoz-Reja and L. Hallstadius, "*Fuel Cladding Behavior Under Rapid Loading Conditions*", J. Nuclear Materials 469 pages 177-186, Elsevier Ltd., February 2016.

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. K. Yueh, "*Applicability of Modified Burst Test Data to Reactivity Initiated Accident*", Journal of Nuclear Materials 469, pages 338-345, Elsevier Ltd., May 2017.

1 Introduction

1.1 Background

The U. S. Nuclear Regulatory Commission (NRC) has published Reference 2 for public comment regarding reactivity initiated accidents.

1.2 Purpose

The purpose of this document is to provide industry comments on the proposed guidance.

1.3 Scope and Overview

1.3.1 Chapter 2

Provide a summary of the comments on the proposed guidance language. Cross reference to specific comments in Appendix A are included.

1.3.2 Chapter 3

Provide a summary of EPRI experimental work and recommendations to proposed guidance.

1.3.3 Chapter 4

Provide comments on bigger picture issues stemming from the proposed guidance.

1.3.4 Chapter 5

Provide a summary of the high-priority industry comments regarding guidance structure and implementation.

1.3.5 Appendix A

Provide specific comments regarding proposed guidance language.

GENERAL NOTE: Text highlighted in yellow identifies specific text addressed by comments in Appendix A.

2 NRC Proposed Guidance Language

The NRC has noticed a request for comment per Reference 1 regarding proposed guidance with respect to reactivity initiated accidents per Reference 2. This chapter identifies general comments relating to each part of the Reference 2 guidance, and includes a cross reference to detailed comments in Appendix A.

Comments are in the form of tables with the following format and content:

- DG-1327 Proposed Language Part
- General Comment: High level comment regarding proposed language.
- Cross Reference to Specific Comments: Pointers to comment details in Appendix A.

2.1 Limits and Applicability

The proposal, shown in Table 2.1, discusses the applicable DBE's, applicable fuel designs, and applicability for both high temperature and PCMI cladding failure thresholds.

Table 2.1: Guidance Applicability; Technical Information

DG-1327 Proposed Language
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. The applicability of the fuel rod cladding failure thresholds, fission product release fractions, and allowable limits on damaged core coolability provided in this guidance are limited as follows: 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. 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). 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.
General Comment
A relative power level value may not be the best surrogate for non-PCMI failure modes.
Cross Reference to Specific Comments
Table A.1, Table A.2, Table A.18, Table A.28

2.2 Methods and Models

The proposal, shown in Table 2.2, NRC identifies guidance related to methods and models for the DBE analysis.

Table 2.2: Methods and Models

DG-1327 Proposed Language
2.1 Methods and models 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. 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. 2.1.3 Calculations should be based upon design-specific information accounting for manufacturing tolerances. 2.1.4 Burnup-related effects on reactor kinetics (e.g., βeff, 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.
General Comment
None
Cross Reference to Specific Comments
Table A.7, Table A.8, Table A.9, Table A.10, Table A.15, Table A.19

2.3 Initial Conditions

In the proposal, shown in Table 2.3, NRC identifies guidance related to initial conditions for the accident analyses.

Table 2.3: Initial Conditions

DG-1327 Proposed Language
2.2 Initial Conditions
2.2.1 Accident analyses should be performed at beginning of cycle (BOC) and intermediate burnup intervals up to end of cycle (EOC).
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.
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.
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.
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.
2.2.6 The reactivity insertion rate should be determined from differential control rod worth curves and calculated transient rod position versus time curves.
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.
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.
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.
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.
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.

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.

2.2.12 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.

General Comment

Section 2.2.4 essentially acknowledges that the hydrogen uptake in the cladding leads to a non-linear problem to determine the limiting number of rod failures. Section 2.2.5 utilizes the term "Sufficient parametric studies..." leads to confusion regarding how one knows enough evaluation has been performed to identify "limiting" case. Please clarify.

There are two sections numbered 2.2.12.

Cross Reference to Specific Comments

Table A.4, Table A.22, Table A.25

2.4 Estimating Total Rod Failures

In the proposal, shown in Table 2.4, NRC identifies guidance related to predicting the total number of fuel rod failures for the DBE analysis.

Table 2.4: Predicting Rod Failures

DG-1327 Proposed Language
2.3 Predicting the total number of fuel rod failures 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. 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. 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. 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 bind the entire fuel rod. 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.
General Comment
None

Cross Reference to Specific Comments
Table A.11, Table A.20, Table A.22, Table A.23

2.5 Fission Product Release Fractions

In the proposal, shown in Table 2.5, NRC identifies guidance related to modeling the fission product release for the DBE analyses.

Table 2.5: Fission Product Release

DG-1327 Proposed Language
2.4 Fission product release fractions 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.
General Comment
None
Cross Reference to Specific Comments
Table A.5,

2.6 Reactor Coolant System Peak Pressure

The proposed language, shown in Table 2.6, NRC identifies guidance related to modeling the reactor coolant system peak pressure for the PWR CRE DBE analysis.

Table 2.6: RCS Peak Pressure

DG-1327 Proposed Language
2.5 Reactor coolant system peak pressure 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.
General Comment
None
Cross Reference to Specific Comments
None

2.7 Cladding Failure Criteria

The proposed language, shown in Table 2.7, NRC identifies guidance related to the fuel rod cladding failure thresholds.

Table 2.7: Rod Failure Threshold

DG-1327 Proposed Language
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. 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). 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). 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.
General Comment
None.
Cross Reference to Specific Comments
Table A.1, Table A.2, Table A.3, Table A.13, Table A.14, Table A.16, Table A.17, Table A.21, Table A.28

2.8 Release Fractions

In the proposed language, shown in Table 2.8, NRC identifies guidance related to modeling the fission product release fractions.

Table 2.8: Fission Product Release Fractions

DG-1327 Proposed Language
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 gap 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.
General Comment
None
Cross Reference to Specific Comments
Table A.5

2.9 Allowable Radiological Consequences

In the proposal, shown in Table 2.9, NRC identifies guidance related to allowable limits on radiological consequences.

Table 2.9: Allowable Radiological Consequences

DG-1327 Proposed Language
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.
General Comment
None
Cross Reference to Specific Comments
Table A.5

2.10 Allowable RCS Pressure

The proposed language, shown in Table 2.10, NRC identifies guidance related to allowable limits on reactor coolant system pressure.

Table 2.10: Allowable RCS Pressure

DG-1327 Proposed Language
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).
General Comment
None
Cross Reference to Specific Comments
Table A.6

2.11 Core Coolability

The proposed language, shown in Table 2.11, NRC identifies guidance related to allowable limits on damaged core coolability.

Table 2.11: Allowable Damaged Core Coolability

DG-1327 Proposed Language
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.
General Comment
None
Cross Reference to Specific Comments
None

3 EPRI Experimental Results

A variety of publicly available sources document EPRI work related to fuel rod failure per References 3, 4, 5, & 6.

3.1 Temperature and Pulse Width Effects

Section C.3.2: Temperature and Pulse Width Effect on PCMI Cladding Failure Threshold for SRA Cladding at High Temperature and RXA Cladding at Low Temperature

The PCMI cladding failure thresholds proposed by the NRC in Figure 3 (SRA cladding at high temperature; e.g. PWR Zr-4 application) and Figure 4 (RXA cladding at low temperature; e.g. BWR Zr-2 application) are based on NSRR test conditions that are atypical of LWR RIA events. EPRI research results expand the database of cladding performance during RIAs with a focus on the effects of pulse width and cladding temperature. Then the EPRI Falcon code uses the expanded database to develop proposed cladding failure thresholds based on the critical strain energy density concept that are applicable to LWR operating conditions during RIA events. These proposed cladding failure thresholds are more appropriate than the overly conservative DG-1327 failure thresholds.

3.1.1 Proposed Change to DG-1327

Replace Reference 2 Figure 4 (RXA cladding at low temperature) with a constant PCMI cladding failure threshold of 150 Δcal/g. Replace Reference 2 Figure 3 (SRA cladding at high temperature) with the 280°C and 10ms pulse width (blue dotted curve) on Figure 3.1:

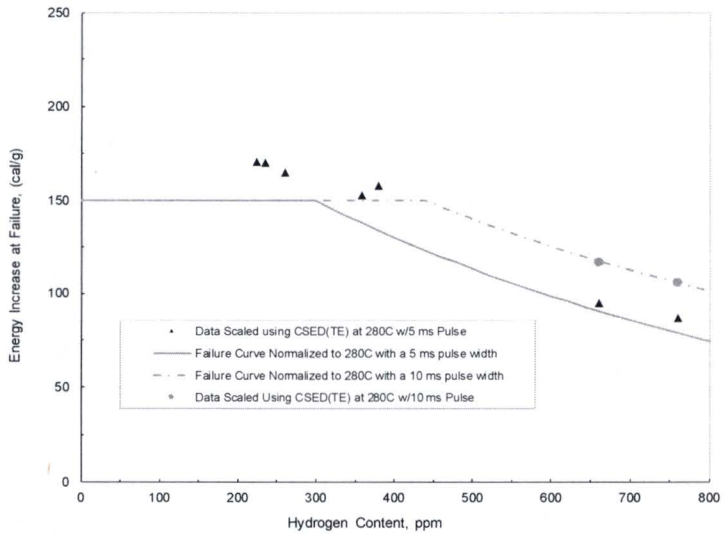


Figure 3.1 Proposed Replacement for Reference 2 Figure 3

3.1.2 Technical Justification

3.1.2.1 Background

The PCMI cladding failure threshold limits are stated as applicable to ≥500°F (Reference 2, Figures 2 and 3), or to <500°F (Reference 2, Figures 4 and 5). For PWRs only, Reference 2, Figures 2 and 3 are applicable as critical reactor initial conditions associated with the limiting control rod ejection accidents are restricted to >500°F. For BWRs the minimum temperature for criticality can be as low as ambient temperature, and the limiting control rod drop accident condition can also occur at the lowest ambient temperature.

The NSRR PCMI test data (Δcal/g to cause cladding failure) used by NRC to develop the PCMI limit curves is for cladding initial temperatures ranging from 20°C (68°F) to 280°C (536°F). With most of the data from 20°C tests, and with the allowable Δcal/g being lower at 20°C than at higher temperatures, a process was developed to scale the 20°C data up to 280°C. Based on two pairs of data a bias of +20 Δcal/g is used to adjust the Figure 4 low temperature RXA cladding failure threshold to the Reference 2, Figure 2 high temperature RXA cladding failure threshold. Similarly, a bias of -18 Δcal/g is used to adjust the Reference 2, Figure 3 high temperature SRA cladding failure threshold to the Reference 2, Figure 5 low temperature SRA cladding failure threshold. The NRC decided not to adjust the data for the effect of pulse width although the pulse width in the NSRR data is much shorter in duration than the BWR CRDA and PWR CRE events of concern.

3.1.2.2 EPRI Research Results

To study the temperature effect on cladding material properties EPRI has conducted two test programs with the intent of determining the brittle-to-ductile (BTD) transition temperature. The first test program "Rapid Heating and Loading Test" (RHLT) employed a hydraulically-driven tensile test machine with a very rapid displacement rate. Zircaloy-2 samples pre-hydrided to ~500 ppm hydrogen, and irradiated cladding specimens with ~160 ppm hydrogen were tested.

Test results indicate the BTD transition temperature is in the range of 75°C for the pre-hydrided samples (shown in Figure 3.2) and 85°C for the irradiated cladding specimen (shown in Figure 3.3). The specimens were fully ductile at 100°C. Therefore, for BWR RXA cladding, failure due to PCMI cannot occur above 100°C.

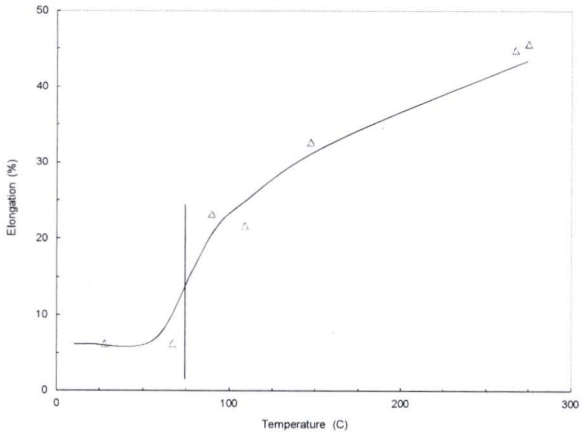


Figure 3.2 Ductile-Brittle Transition for Pre-Hydrided Samples: Reference 3, Figure 2-7

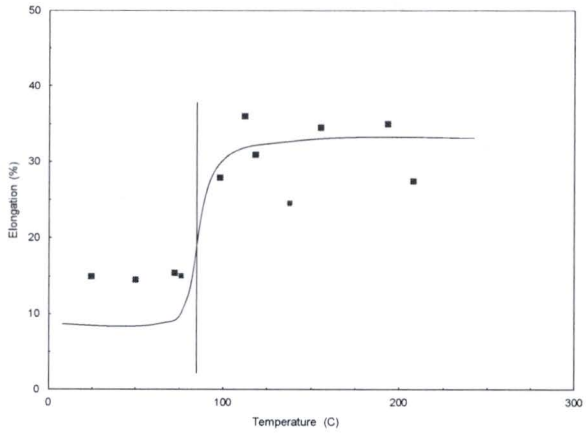


Figure 3.3 Ductile-Brittle Transition Irradiated Samples: Reference 3, Figure 2-8

The second EPRI test program "Modified Burst Test" (MBT) employed a pressurized driver tube inside irradiated cladding specimens. Rapid expansion of the driver tube simulates the fuel pellet expansion during an RIA. Results of EPRI MBT tests for PWR Zircaloy-4 cladding as a function of cladding hydrogen concentration are shown in Figure 3.4.

The 25°C data indicates that the SRA cladding is brittle with increasing loss of ductility with increasing hydrogen content. The burst strain for the 280°C temperature data decreases exponentially with increasing hydrogen concentration and approaches the room temperature ductility at ~750 ppm. Significant improvement in cladding ductility was indicated by the 280°C test data out to ~550 ppm hydrogen. Below 350 ppm the cladding is ductile (burst strain >2%) at 280°C.

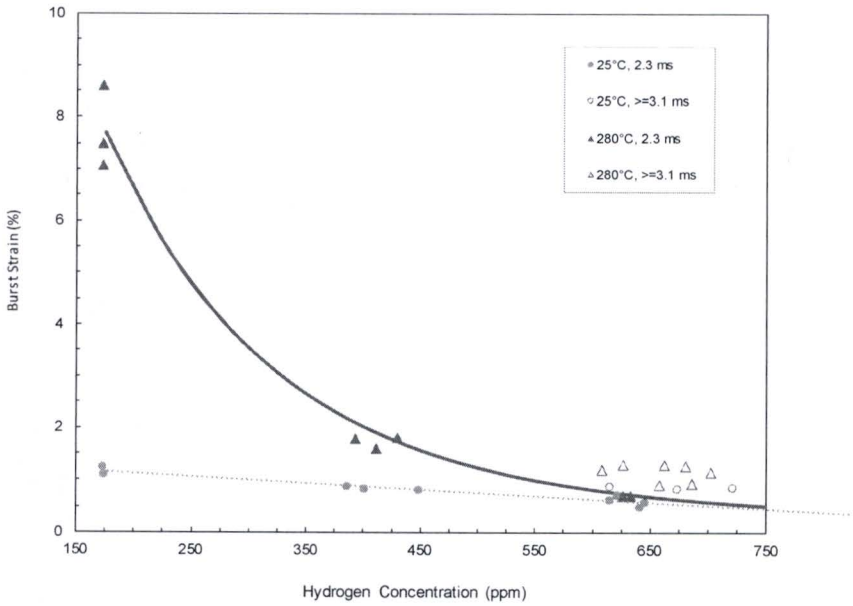


Figure 3.4 Modified Burst Test PWR Cladding Data: Reference 3, Figure 2-14

Results of EPRI MBT tests for PWR Zircaloy-4 cladding as a function of loading time to burst for the brittle high hydrogen content data at 280°C, are shown in Figure 3.5. The slope of the data indicates a 0.1% increase in burst strain per 1 msec increase in loading time to burst. PWR SRA cladding is estimated to be ductile (burst strain >2%) at 280°C based on extrapolation of these data at a loading time to burst of ~15 msec.

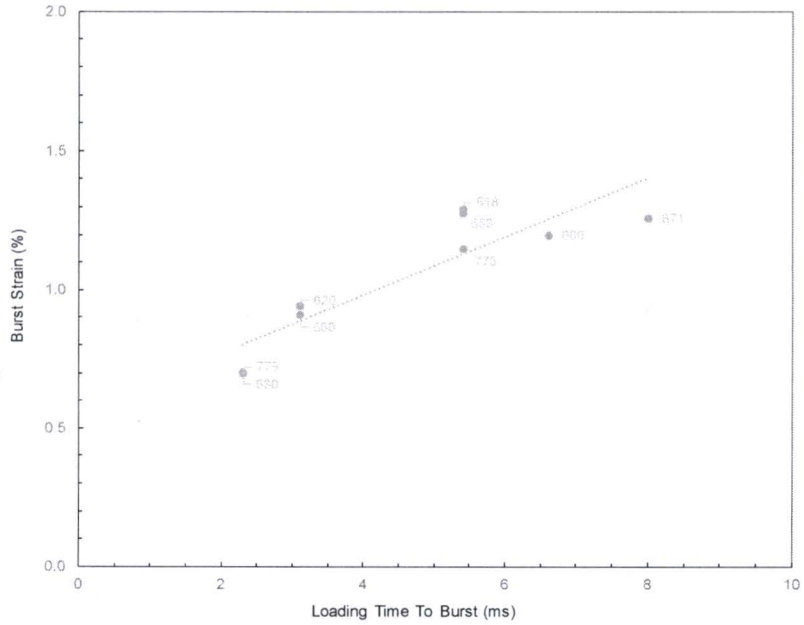


Figure 3.5 PWR Cladding Burst Strain Pulse Width Dependence: Reference 3, Figure 2-15

The PWR MBT data are fit to obtain the predicted cladding burst strain (BS) for NSRR RIA tests as a function of hydrogen concentration (H) in ppm, and for the NSRR test pulse width (PW msec) adjustment relative to the PWR MBT loading time to burst (LTTB = 2.5 msec). Correlations for burst strain are developed for 298°K and 553°K.

PWR SRA: BS(%) = 1.374 - 0.0012 * H + 0.1 * (PW – 2.5) [for 298°K]

PWR SRA: BS(%) = 20.92 + (9.34E-11 * H4) – (2.43E-7 * H3) + (2.42E-4 * H2) – (0.111 * H) + 0.1 * (PW – 2.5) [for 553°K]

Results of the EPRI MBT tests for BWR RXA cladding as a function of temperature are shown in Figure 3.6. These data confirm that BWR RXA cladding is ductile >100°C. There is no correlation on hydrogen concentration.

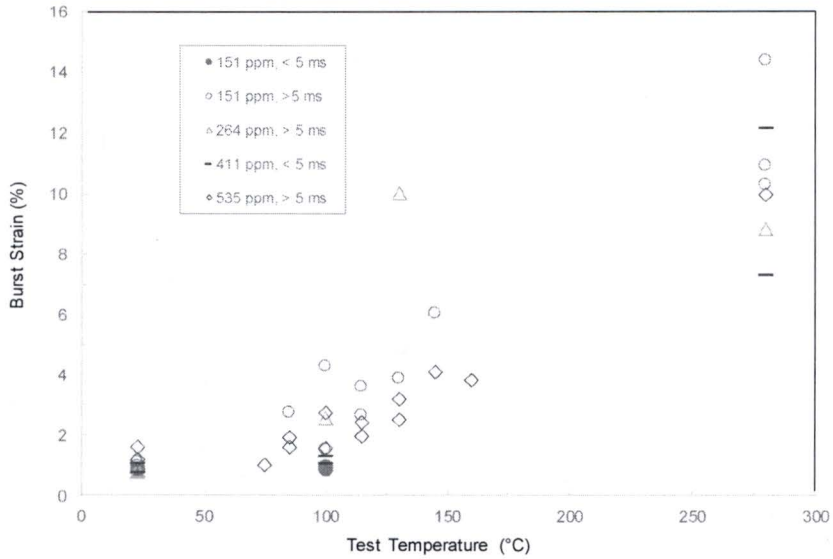


Figure 3.6 Burst Strain Temperature Dependence, BWR RXA Cladding

Results of the EPRI MBT tests for BWR RXA cladding as a function of loading time to burst, and by temperature, are shown in Figure 3.7. The data indicates that the cladding is ductile (burst strain >2%) for test temperatures >85°C for loading time to burst >15 msec. At 25°C extrapolation of the data indicates cladding ductility for a loading time to failure of ~25 msec. There is no correlation with hydrogen concentration (the small numbers near the data points).

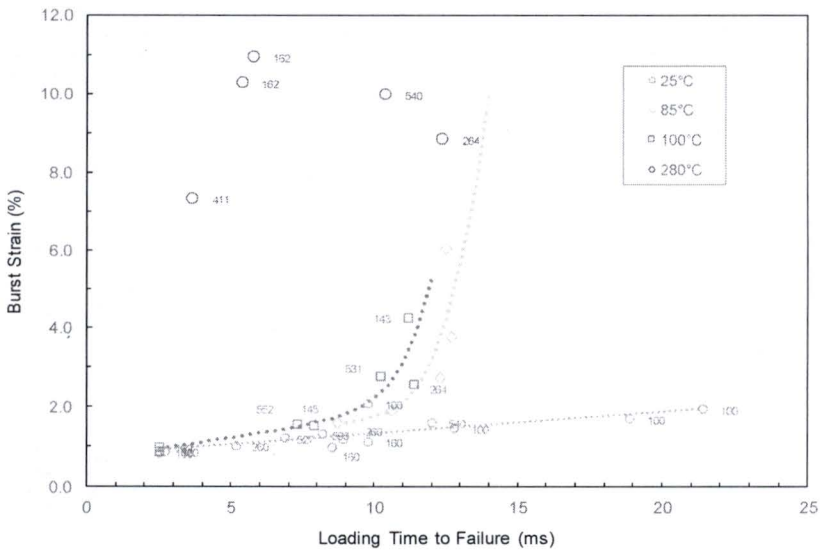


Figure 3.7 Burst Strain Time to Failure, BWR RXA Cladding

The Zr-2 MBT data are fit to obtain the predicted cladding burst strain as a function of pulse width/LTTB. There is no correction for hydrogen concentration.

BWR RXA: BS(%) = 0.6507 + 0.0743 * PW [for 298°K]

The Falcon code is then used to calculate the thermal expansion of the fuel in the NSRR tests at the energy deposition reported at the time of cladding failure. NSRR test data with PCMI failures are compared to MBT measured test results in Figure 3.8. The NSRR fuel strain at failure is calculated using Falcon based on the energy deposition, and that result is compared with the cladding burst strain equations (PWR SRA or BWR RXA) developed above from the MBT measured test results corrected for the NSRR test conditions. The results of the comparison indicate that the MBT tests can reasonably represent the NSRR RIA tests.

Table 3.1: Measured vs. Calculated Strain: Reference 6, Table 3

Test ID	Failure dH (Joules)	Cladding Type	Measured MBT Strain (%)	Calculated NSRR Strain (%)
VA-1	268	SRA	0.76	0.76
VA-2	230	SRA	0.64	0.61
VA-3	343	SRA	1.18	1.04
VA-4	NF*	SRA	1.09	1.40
MH-3	NF*	SRA	1.18	1.20
HBO-1	354	SRA	1.02	1.17
HBO-5	506	SRA	1.05	1.73
HBO-6	NF*	SRA	1.22	1.67
OI-11	502	SRA	1.30	1.71
FK-6	272	RXA	0.97	0.87
FK-7	242	RXA	0.97	0.77
FK-9	359	RXA	1.07	1.10
FK-10	318	RXA	1.04	0.91
FK-12	280	RXA	1.06	0.89
LS-1	222	RXA	0.98	0.77

*Calculated at the Maximum Enthalpy Increase

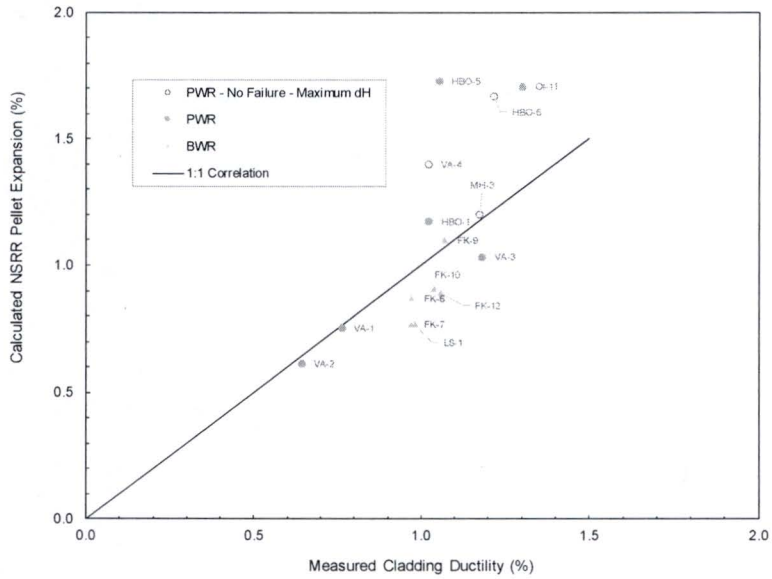


Figure 3.8 Pellet Expansion to Cladding Ductility Comparison: Reference 6, Figure 5

The EPRI Falcon fuel performance code uses NFIR-IV and TEPCO Zr-2 tube burst and EPRI MBT data in its critical strain energy density (CSED) failure model. The room temperature (20°C) data are adjusted to a consistent 7 msec time- to-burst (using the slope Figure 3.5) and a curve fit produces the CSED limit for those conditions; results are shown in Figure 3.9. Similarly, the 100°C MBT data are adjusted to a 7 msec time-to-burst and a curve fit produces the CSED limit for those conditions.

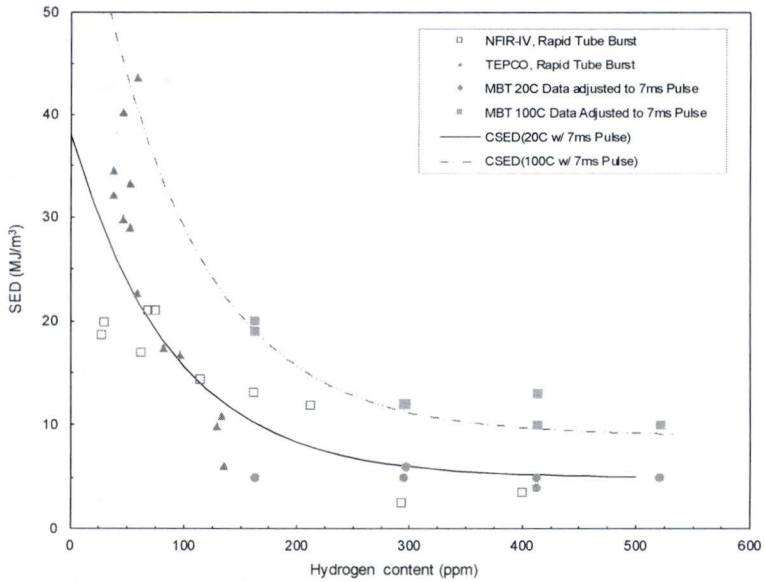


Figure 3.9 BWR RXA Cladding and Falcon CSED

The EPRI Falcon fuel performance code uses Zr-4 tube burst and EPRI MBT data in its critical strain energy density (CSED) failure model. The hot zero power temperature (280°C) data are adjusted to a consistent 5 msec time-to-burst and a curve fit produces the CSED limit for those conditions; results are shown in Figure 3.10. Then, those data are adjusted to a 10 msec time-to-burst and a curve fit produces the CSED limit for those conditions.

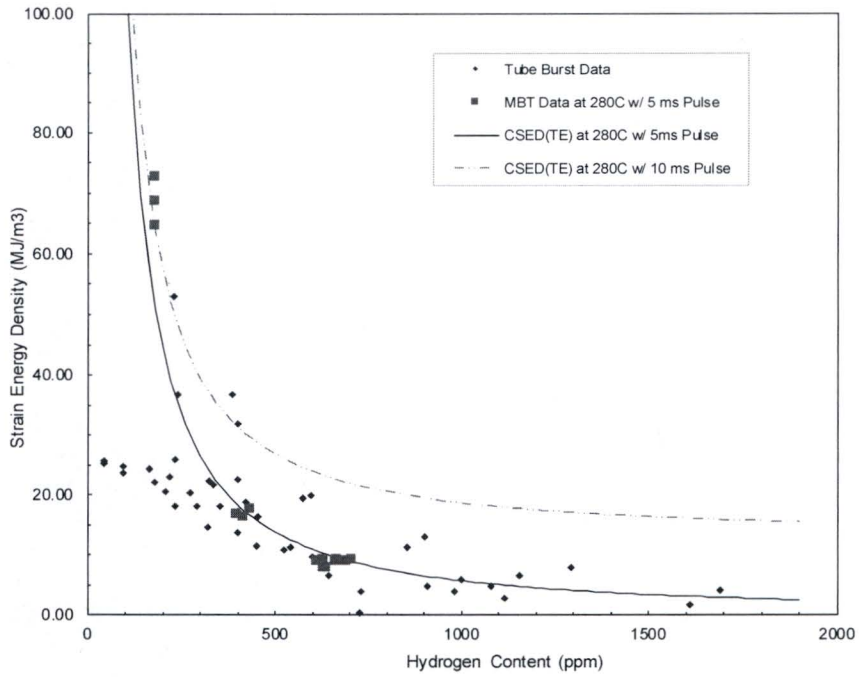


Figure 3.10 PWR SRA Cladding at 280°C and Falcon CSED

Falcon is used to model the NSRR and CABRI RIA test data. The strain energy density (SED) is calculated during the RIA pulse for each test and the result is compared to the CSED limit value corresponding to the local cladding material (Zr-2 or Zr-4), temperature, and hydrogen concentration. If the SED exceeds the CSED then cladding failure is predicted by Falcon.

The Falcon CSED (TE) approach successfully predicts the failure of CABRI and NSRR high-temperature high-pressure (HTHP) capsule data at 280°C; results are shown in Figure 3.11. These are PWR cladding tests. The two CABRI tests (RepNa8 and 10) and the one NSRR test (VA3)with cladding failure due to PCMI lie above the CSED limit assuming a 5 msec pulse width.

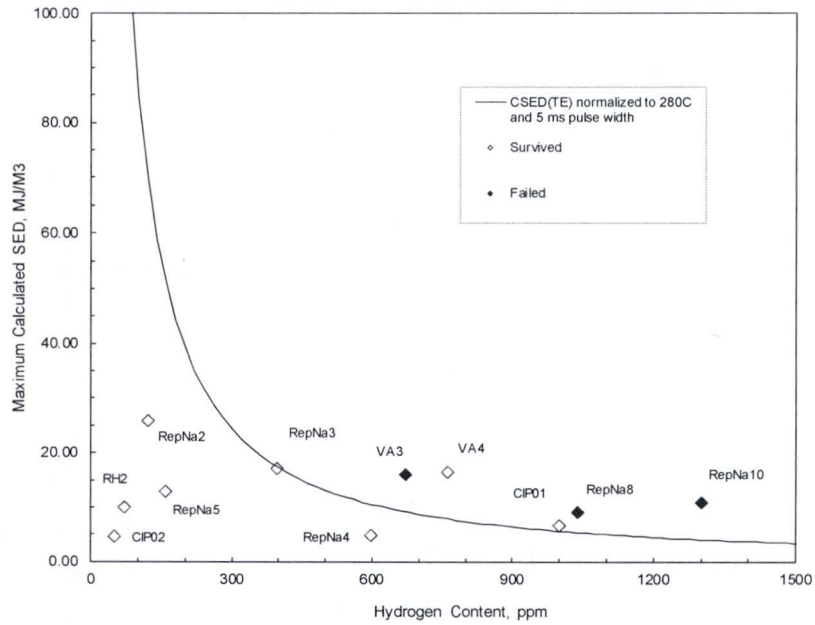


Figure 3.11 CSED High Temperature Comparison, PWR: Reference 3, Figure 4-7

The Falcon CSED (TE) approach successfully predicts the failure of NSRR BWR cladding tests with low temperature data at 20°C; results shown in Figure 3.12. Tests FK-10 (conducted at 80°C) and FK-12 (conducted at 85°C) are also shown in this figure as PCMI failures. This indicates that the BTD transition temperature was not reached. The NSRR tests with cladding failure due to PCMI lie above the CSED limit assuming a 5 msec pulse width.

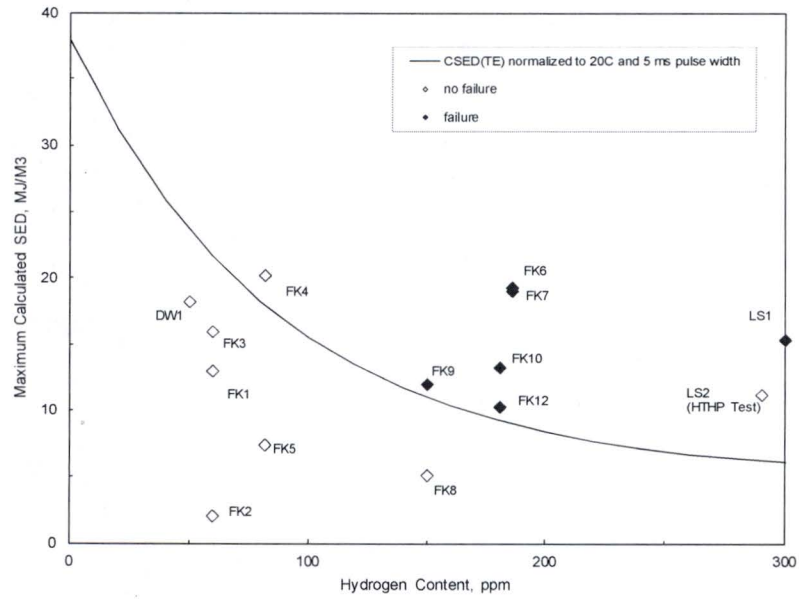


Figure 3.12 CSED Low Temperature Comparison, BWR: Reference 3, Figure 4-11

Falcon code prediction of a RIA pulse of ~5 msec typical of an NSRR test is shown in Figure 3.13. The cladding OD temperature increase is minimal out to the end of the energy deposition. NSRR tests experience only a minimal cladding OD temperature increase prior to the full PCMI loading, and the cladding ductility would not benefit from a temperature increase.

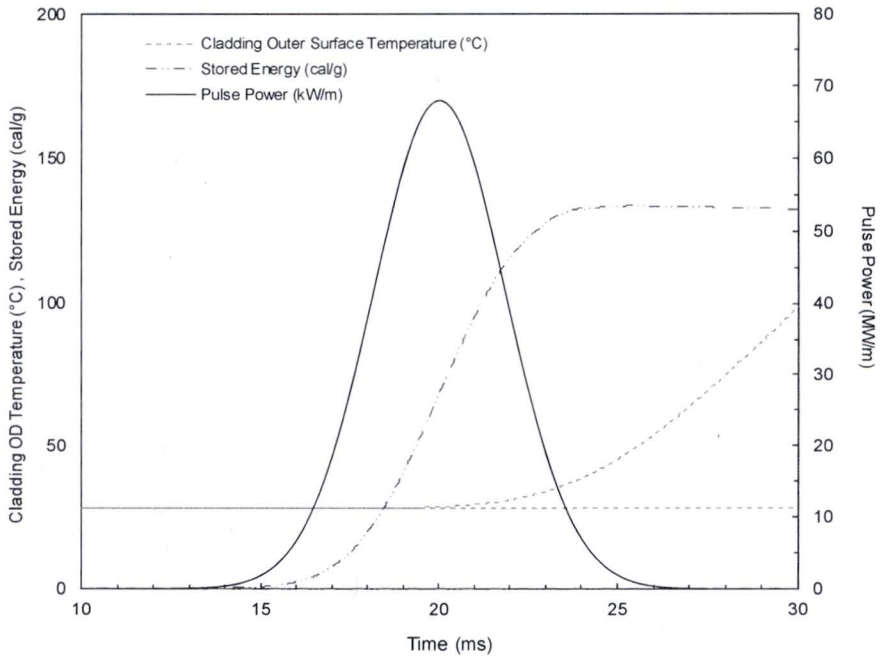


Figure 3.13 Falcon Calculation for Typical NSRR Test
~5 msec Pulse Width; 135 cal/g Energy Deposition

A Falcon code prediction of a BWR CZP RIA with a ~30 msec pulse and an energy deposition of ~115 cal/g is shown in Figure 3.14. 30 msec is used as a conservative lower bound BWR CRDA pulse width in this example. Based on CSNI/R(2010)1 the BWR CRDA pulse width ranges from 45-75 msec. The cladding OD temperature increases from 25°C to ~85°C at an energy deposition of 78 cal/g, and to ~100°C at an energy deposition of 89 cal/g, well prior to the end of the pulse, and then continues to increase. From the EPRI RHLT and MBT Zircaloy-2 BWR cladding test data it can be concluded that the BTD transition temperature is in the range of 75-85°C, and by 85°C the cladding is fully ductile with a pulse width ≥ 15 msec.

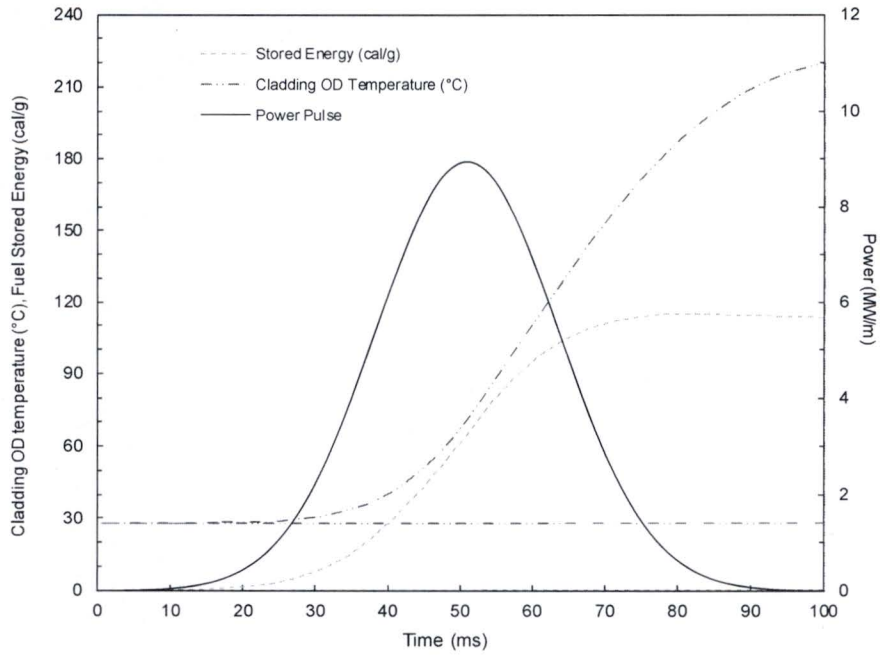


Figure 3.14 Falcon Calculation of Limiting BWR RIA Scenario
30 msec Pulse Width; 115 cal/g energy deposition

3.1.2.3 Proposed BWR RXA Cladding CZP PCMI Limit

Based on the BWR CRDA pulse width effect and the 85°C BTD transition temperature for Zr-2 demonstrated in the EPRI RHLT and MZBT, BWR Zr-2 cladding failure due to PCMI will not occur. In addition, based on EPRI MBT testing, the BWR CRDA pulse width results in a ductile burst strain >2% and cladding failure due to PCMI will not occur. There is no risk of Zircaloy-2 cladding failure due to PCMI following a BWR CRDA due to either the pulse width, or with the cladding outer surface or coolant temperature >85°C. Therefore, the proposed energy deposition limit for BWR Zr-2 RXA cladding for all hydrogen concentrations is the high-temperature cladding failure limit of 150 Δcal/g.

3.1.2.4 Proposed PWR SRA Cladding HZP PCMI Limit

Figure 3.4 shows the PWR SRA cladding at 280°C is ductile (BS >2%) for hydrogen concentrations below ~420 ppm. An extrapolation of the slope from Figure 3.5 indicates that PWR SRA cladding at 280°C is ductile for pulse widths >15 msec. Based on CSNI/R(2010)1 the PWR CRE pulse width ranges from 25-65 msec at HZP initial conditions. Therefore, the combination of adjustments for temperature and pulse width indicate that PCMI failures will not occur for PWR SRA cladding at HZP RIA conditions.

The Falcon CSED(TE) predicted cladding failure limit for PWR SRA cladding at 280°C and assuming a 5 msec pulse width (NSRR), or a 10 msec pulse width (bounding PWR HZP REA) is shown in Figure 3.10. Falcon is then used to determine the Δcal/g as a function of the PWR

NSRR failed cladding test hydrogen content values to reach the CSED(TE) limit for the 5 msec pulse width (NSRR), and for the 10 msec pulse width (PWR HZP REA).

The results are presented in Figure 3.15. The 10 msec data are then bounded by the selected limit line shown in Figure 3.15. The proposed limit line is a combination of the 150 Δcal/g high-temperature cladding failure limit and the proposed PCMI limit for PWR SRA cladding at 280 °C. It is noted that application of the same approach for a typical PWR HZP REA 25 msec pulse width would show additional margin. Based on the test data presented, the adjustment of the data to a bounding PWR HZP REA pulse width, and the Falcon CSED analysis results, the PCMI cladding failure threshold for PWR SRA cladding at 280°C should be revised accordingly.

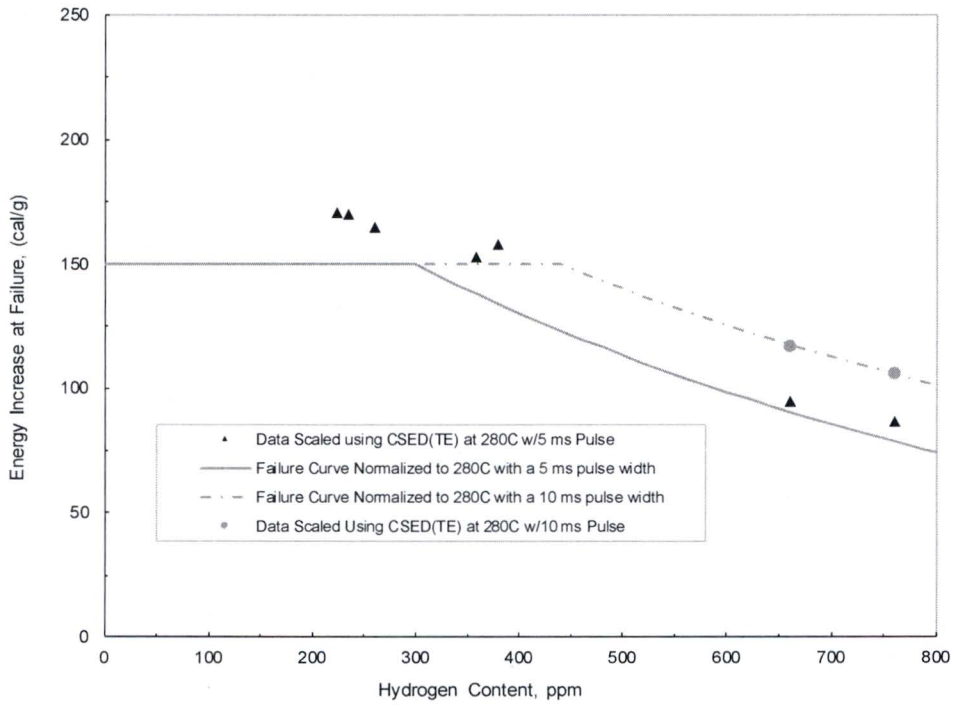


Figure 3.15 EPRI Proposed PCMI Failure Threshold for Zr-4 Cladding PWR HZP Conditions (280°C) Based on the Falcon CSED(TE) Methodology

3.2 RXA Cladding at High Temperature Limit

Section C.3.2 : Temperature and Pulse Width Effect on PCMI Cladding Failure Threshold for RXA Cladding at High Temperature

The PCMI cladding failure threshold proposed by the NRC in Figure 2 (RXA cladding at high temperature; e.g. PWR M5™ application) is based on a +20 Δcal/g adjustment to the low temperature RXA cladding failure threshold. Based on insights from EPRI research into pulse width and temperature effects an alternate cladding failure threshold for RXA cladding at high

temperature is proposed. The proposed cladding failure threshold is more appropriate than the overly conservative DG-1327 failure threshold.

3.2.1 Proposed Change to DG-1327

Replace Reference 2, Figure 2 (RXA cladding at high temperature) with a constant PCMI cladding failure threshold of 150 Δcal/g.

3.2.2 Technical Justification

RXA cladding at high temperature (e.g. PWR M5™ application) was not part of the EPRI MBT test program and the Falcon CSED methodology was not applied to this cladding material at high temperature. However, the BTD transition temperature measured for RXA material up to 593 ppm showed fully ductile behavior by 85°C for pulse widths ≥ 15 msec. It is noted that the hydrides are concentrated in the liner, and so the base material Zr-2 hydrogen concentration is assumed to be 50% of the average, or 296 ppm. For the purposes of the RXA cladding high temperature evaluation, with no liner, the hydride concentration is assumed the same as the BWR base material, or 296 ppm. For PWR REA at HZP the initial conditions are in the range of 280°C. At those cladding temperatures, and with the PWR pulse width >15 msec, it is expected based on the low temperature RXA test data, that RXA cladding at PWR high temperature conditions is ductile and there is no possibility of cladding failure due to PCMI. Therefore, the proposed energy deposition limit for PWR RXA cladding at HZP conditions for hydrogen concentrations up to 296 ppm is the high-temperature cladding failure limit of 150 Δcal/g.

3.3 SRA Cladding at Low Temperature Limit

The PCMI cladding failure threshold proposed by the NRC in Figure 5 (SRA cladding at low temperature; e.g. BWR Zr-4 application) is based on a -18 Δcal/g adjustment to the high temperature SRA cladding threshold. Based on insights from EPRI research into pulse width and temperature effects an alternate cladding failure threshold for SRA cladding at low temperature is proposed. The proposed cladding failure threshold is more appropriate than the overly conservative DG-1327 failure threshold.

3.3.1 Proposed Change to DG-1327

Replace Figure 5 (SRA cladding at high temperature) with the red line shown in Figure 3.16.

3.3.2 Technical Justification

PWR SRA cladding test data was obtained at room temperature and 280°C. Cladding of the same metallurgical condition (cold-worked SRA / CWSRA) is also used in BWR applications. Although the operating environments of the two reactor types are different, their impact on the cladding behavior is captured by their respective corrosion and hydrogen pickup models. The effects of pulse-width and temperature on cladding ductility observed with PWR SRA cladding would be applicable to SRA cladding in BWR application. Testing at intermediate temperatures was not conducted for the PWR SRA cladding, and therefore adjustment to the NSRR test results only considers the pulse-width effect. Test data indicates SRA cladding at room

temperature to have a stronger pulse width dependence (0.1% BS/msec) than RXA cladding (0.055% BS/msec), but because of the limited test data pulse width range, the lower slope room temperature RXA cladding pulse width dependence is used to adjust the NSRR test data assuming a conservative 20 msec pulse width. The proposed limit for SRA cladding at BWR CZP conditions is shown in Figure 3.16.

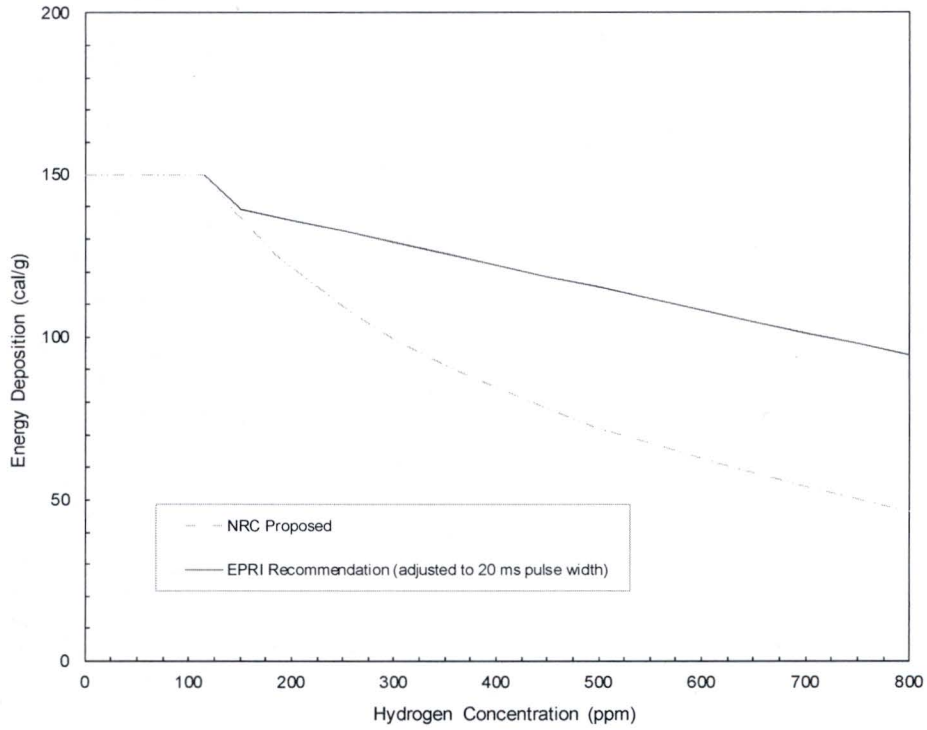


Figure 3.16 CWSRA cladding limit at CZP condition adjusted for loading rate effect.

3.4 Extended RXA Cladding Hydrogen Range

Section C.3.2 : Excess Hydrogen Concentration is Limited to 300 ppm for RXA Cladding.

Reference 2, Figures 2 and 4 limit the excess hydrogen concentration to 300 ppm for RXA cladding. This value does not support current BWR fuel burnup limits. Consequently this limit must be extended.

3.4.1 Proposed Change to DG-1327

Extend the excess hydrogen concentration to 593 ppm in Figures 2 and 4 based on EPRI test data. The cladding failure threshold can then be extrapolated out to 593 ppm.

3.4.2 Technical Justification

The PCMI limit for RXA material in Reference 2, Figures 2 and 4 is applicable up to 300 ppm of excess hydrogen. This value is lower than the present acceptance criteria for hydrogen content

used by the BWR vendors in the fuel rod design reports. 300 ppm can be exceeded if a conservative hydrogen uptake model is used. If the hydrogen uptake model of RG 1.224 is used the 300 ppm value is exceeded for rod local axial burnups above 65 GWd/MT, which is lower than the licensing value of the present BWR fuels (70 GWd/MTU).

The EPRI BWR MBT test database extends to 593 ppm, significantly higher than the 300 ppm established by the NSRR LS-series tests. Fuel cladding material from an LS-series sister rod was used in the MBT tests and the measured ductility tracks with the calculated NSRR failure strain.

The EPRI proposed PCMI cladding failure threshold for Zircaloy-2 includes the extension of the limit to 593 ppm hydrogen content.

3.5 CSED Elongation Approach

Section C.3.3: Total Elongation vs. Uniform Elongation

The NRC 3/16/2015 memo that is the technical basis for DG-1327 discusses in Section 3.2.2.2 the EPRI critical strain energy density (CSED) approach and is critical of the use of total elongation (TE) strain data. Technical justification for the use of the total elongation data is provided in this comment.

3.5.1 Proposed Change to DG-1327

The EPRI proposed cladding failure thresholds for RXA cladding (low temperature BWR application) and for SRA cladding (high temperature PWR application) are based on the FALCON CSED(TE) approach. This comment supports those proposed changes.

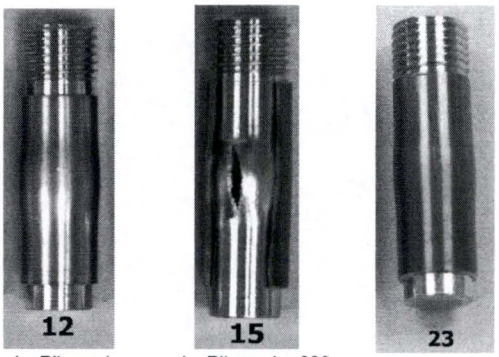
3.5.2 Technical Justification

The Falcon critical strain energy density (CSED) methodology used in the development of the proposed cladding failure thresholds for RXA cladding (low temperature BWR application) and for SRA cladding (high temperature PWR application) uses Zircaloy material properties test data with total elongation (TE) test results. The NRC has expressed concerns with the Falcon CSED(TE) approach. In response to the NRC's concern EPRI has previously shown comparisons using the Falcon CSED methodology assuming uniform elongation (UE). The basis for the NRC concern is the observed deformation instability in standard cladding burst test results beyond UE. The observed instability is a range of burst strains (scatter in the data) and is attributed to localized variations in cladding material thickness, properties, and imperfections.

The EPRI modified burst tests are more typical of an RIA response (compared to a standard burst test) in that the cladding loading is by strain rather than by pressure. The MBT strain loading is relatively uniform, whereas standard burst test pressure loading would result in amplified stresses at the location of cladding imperfections. In addition, since 2% strain due to fuel thermal expansion bounds the PWR REA and BWR CRDA events, strain behavior beyond 2% is not applicable. 2% strain is only slightly above the uniform strain. Stability in strain-driven

deformation is consistent with reported NSRR data, where in many of the tests the cladding deformed significantly above UE.

Examples of MBT test specimens in the post-test condition are shown in Figure 3.17. Test results show deformation instability does not occur upon reaching UE and samples survive to TE and maintain uniform wall thickness. Also, the scatter in the MBT test strain results is much less than in standard burst tests.



As-Pilgered
7.5% strain
Uniform strain=4%
25°C

As-Pilgered + 600
ppm hydrogen
3.5% target strain
25°C

As-Pilgered + 600
ppm hydrogen
5.6% strain
250°C

Figure 2-11
Examples of Samples Following a Modified Burst Test with an 18 ms Pulse

Figure 3.17 MBT Samples: Reference 3, Figure 2-11

MBT test results for test samples that required more than one loading cycle to burst are provided in Table 3.2. In many of the multiple loading cycle tests the strain following the next-to-last cycle was at or near the failure strain in the last loading cycle, but remained intact. These test results support the observation that post-uniform strain deformation instability is not a concern under strain driven type of deformation.

Table 3.2: Deformation History of Multi-Hit Test Samples, No Failure: Reference 4, Table 4

Rod	Sample	Hydrogen	Temp.	Diameter Change Hit Number		
		(ppm)	(K)	1	2	3
GN592	B3	162	553	10.8	11.0	-
	B6	162	553	10.1	10.3	-
	B7	140	418	2.3	4.9	6.0
	B8	141	388	1.5	3.6	-
	B9	142	358	2.1	2.7	-
	B11	144	296	0.9	1.0	1.1
	B15	148	553	7.6	12.5	14.4
	B16	149	296	1.0	1.0	-
AEB071	B7	264	403	6.7	10.0	-
F3A1	B1	543	296	0.9	0.9	1.6
	B2	540	553	7.4	10.0	-

4 Big Picture

This section of the report addresses issues and topics which are of a fundamental nature with respect to Reference 2, and its development.

4.1 What Kind of Analyses

4.1.1 Accident Analyses

Design basis events in licensee's FSAR documents have their own development history. As one reads Reference 2, the industry is concerned the draft guidance could be interpreted that beyond design basis considerations may be drifting into expectations for DBE evaluation. Related commentary can be found in Table A.25.

4.1.2 Conservative vs Realistic

DBE analyses for the CRE/CRDA have typically been conservative, bounding analyses. The choice is relatively easy to defend as current failure criteria are essentially numerical constants. In several ways, language in Reference 2 speaks to a more realistic assessment philosophy going forward, particularly regarding treatment of uncertainties. Does the Staff still expect conservative bounding analyses remain the only acceptable approach, and if so, please clarify by removing language implying a realistic analysis basis.

There is a statement in SRP Section 4.2.I Item (3) stating the number of fuel rod failures is not underestimated for postulated accidents. However, that does not justify the converse of mandating rod failures must be exaggerated, which may not be consistent with a standard of reasonable assurance.

4.1.3 Failure Criteria

Important aspects of the new failure criteria are addressed in the following subsections.

4.1.3.1 Non-Linearity

Reference 2 proposed failure criteria are no longer numerical constants, but non-linear functions. Fuel rod failure is also being defined as multi-modal: PCMI, DNB/CPR, and High Temperature cladding pressure differential modes. Consequently, this has ramifications for trying to determine a "bounding" number of rod failures using a conservative analysis approach. The nature of the failure criteria and modes argues for a more realistic approach, which parts of Reference 2 support. However, if Staff still expects conservative bounding analyses are the only acceptable approach, then there seems to be a significant mismatch. Please clarify the Staff's position on the subject.

4.1.3.2 Retained Margin

This is problematic as Reference 2 proposed failure limits are based on tests which are not necessarily representative of actual reactor conditions, (e.g., hyper short pulse width tests). Additionally, Hydrogen uptake models may be required to conservatively predict excess

Hydrogen content; thus pushing cladding evaluation to artificially lower failure criteria. In the area of transient fission gas, models don't make physical sense for fresh fuel (unirradiated fuel must assume substantial inventory release). The reason this matters is forcing conservative bounding analyses to compare against criteria with excessive retained margin could produce unrealistic, and possibly non-physical outcomes.

4.2 Dose

4.2.1 Existing Licensing Basis

AST/TID licensing is based on existing guidance per RG1.183 or RG1.195 and associated assumptions. To alter the way one arrives at a number of failed rods for a specific sub-analysis of the Dose assessment effectively invites backfit of the licensing basis.

4.2.2 New Assumptions

Existing AST analyses do not take transient fission gas release into account (RG1.183 Revision 0 does not have any such information). While it may be technically correct to introduce a new factor in the Dose calculation, the ramifications could cause unintended results. The industry would prefer changes to the existing licensing basis assumptions be kept in a single guidance, actually regarding Dose, rather than potentially scattered across many different guidance sources.

4.2.3 Unforeseen Consequences

New DBE assessment requirements combined with new Dose assessment assumptions mean there are no guarantees existing licensing bases Dose calculations can be defended. There is also a problem with having multiple analyses of record for the same DBE. In the first case, you would have an assessment conforming to the proposed new guidance. At the same time you must maintain a second Dose assessment to defend an AST basis. It may not be possible to reconcile the two calculations as the DBE assessment could be more detailed on a cycle specific basis, while the AST calculation may be a much less frequent assessment, requiring more conservative assumptions.

4.3 Implementation

The logistics of forward fit implementation will need further discussion with Staff. As it stands now, a licensee could be held to a standard which has no approved method; consequently there is no basis to determine the scope of submittal material even if asking the Staff for clarification.

Industry suggests holding a workshop to iron out issues associated with submittal work scope and timing.

5 Summary

The industry comments on DG-1327 presented in this document are on behalf of the operating fleet 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. The comments reflect industry 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. The highest priority industry comments are summarized in the following sections.

5.1 Comments on Technical and Regulatory Guidance

- Beneficial temperature and pulse width effects are not considered in the proposed PCMI cladding failure thresholds. EPRI research and CSED analysis methodology provides recommended changes.
- RXA cladding hydrogen concentration needs to be extended beyond 300 ppm. EPRI testing justifies extending to 593 ppm.
- Non-applicability of high-temperature cladding failure threshold to BWR CRDA for initial power level >5%. This addresses in part the industry concern with the expanded set of cases to be analyzed.
- Guidance on fission product release fraction, radiological considerations, and RCS pressure limits should be deleted.

5.2 Comments on Technical and Regulatory Bases

- PCMI limits are based on NSRR data that is not applicable to PWR REA and BWR CRDA. EPRI research and analyses provide reasonable alternative proposals.
- Continued use of legacy NRC-approved RIA analysis methodology where it can be justified.
- Additional justification applying the EPRI CSED (TE) approach has been provided and should be endorsed by NRC.

5.3 Comments on Implementation

- A large set of cases will need to be analyzed with a high cost of implementation.
- The content of an LAR resulting in NRC expecting compliance with DG-1327 content needs clarification.
- Licensees may need to apply new analytical methods prior to NRC approval.
- Consistent enforcement by NRC Staff is needed.

The industry recommends additional dialogue with the NRC staff in a public meeting to discuss the above implementation topics, and to provide NRC with any necessary clarifications regarding the comments included in this document.

Appendix A: *Specific Comments Draft Regulatory Guidance*

Table A.1: Specific Guidance Language Comment 1

Statement	Sections C.1.2 and C.3.1
Comment	The proposed high temperature cladding failure threshold requires use of Figure 1 for initial conditions <5% rated power. For ≥ 5% power initial condition the DNBR or CPR limit is required. What is the technical basis for the use of 5% power?
Rationale	*A 5% power level separating use of enthalpy (cal/gm) and DNBR/CPR acceptance criteria for high-temperature cladding failure. This may originate with NRC's understanding that 5% power separates prompt-critical from sub-prompt. For PWRs that is not necessarily correct.* *Also, just because one is prompt critical, doesn't mean there is a DNBR/CPR issue. Reactor period would need to be sufficiently low in order to challenge boiling heat transfer conditions.* *The 5% power value appears arbitrary, but could make sense if the methodology is lower order without an appropriate kinetics model.*
Proposal	None. Clarification on the technical basis is requested
Justification	*N/A*

Table A.2: Specific Guidance Language Comment 2

Statement	Sections C.1.2 and C.3.1
Comment	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.
Rationale	*Use of a 5% power level for DNBR/CPR acceptance criteria regarding high-temperature cladding failure could lead to irrational failures. Assumed physical failure commensurate with exceeding the licensed boundary conditions is artificial.* *It is also not clear that a steady state DNBR/CPR correlation is a good surrogate for short time scale transients such as a prompt enthalpy deposition event.* *Presumed failure due to exceeding thermal design limits appears arbitrary.*
Proposal	None. Clarification on the technical basis is requested
Justification	*N/A*

Table A.3: Specific Guidance Language Comment 3

Statement	Section C.3.1
Comment	Specifically include language allowing submittal of alternate cladding failure thresholds.
Rationale	
Proposal	Add the following to Section C.3.1: "Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data."
Justification	*The NRC memorandum dated 3/16/2015 that is the technical basis for DG-1327 states the following on p. 19:* *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.* *This language should be included in the final Regulatory Guide such that alternative failure criteria other than DNB/CPR can be used in the future as experimental data or improved methods become available.*

Table A.4: Specific Guidance Language Comment 4

Statement	Section C.2.2.3
Comment	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.
Rationale	*Above a certain power level, negative reactivity feedback from void reactivity is sufficiently large to make the event non-limiting.*
Proposal	In Section C.2.2.3 add the following statement. "For BWR CRDA the initial conditions are limited to from CZP to a low power level where the transient response is non-limiting due to small dropped rod reactivity worth and high void reactivity feedback."
Justification	*BWR physics. The void feedback counter balances the positive reactivity insertion.*

Table A.5: Specific Guidance Language Comment 5

Statement	Sections C.2.4, C.4, & C.5
Comment	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.
Rationale	
Proposal	Delete Sections C.2.4, C.4, and C.5.
Justification	*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 1.195). The proposed radiological consequence limits may conflict with the licensing bases for some licensees.*

Table A.6: Specific Guidance Language Comment 6

Statement	Section C.6
Comment	The content on maximum reactor coolant pressure should be deleted to prevent conflicts with the current licensing basis as specified in FSARs.
Rationale	*Above a certain power level, negative reactivity feedback from void reactivity is sufficiently large to make the event non-limiting.*
Proposal	Delete Section C.6
Justification	*The reactor coolant system 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.*

Table A.7: Specific Guidance Language Comment 7

Statement	Section C.2.1.1
Comment	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.
Rationale	
Proposal	Delete the following language in Section C.2.1.1: *"Accident analyses should be performed using NRC approved analytical models and application methodologies ~~that account for calculational uncertainties~~."*
Justification	

Table A.8: Specific Guidance Language Comment 8

Statement	Section C.2.1.3
Comment	Calculations should not need to include manufacturing tolerances.
Rationale	
Proposal	Delete the following language in Section 2.1.3: *"Calculation should be based on design-specific information ~~accounting for manufacturing tolerances~~."*
Justification	*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.*

Table A.9: Specific Guidance Language Comment 9

Statement	Section C.2.1.1
Comment	RG 1.203 is not mentioned in DG-1327. To clarify non-applicability of RG 1.203 some clarification should be added.
Rationale	
Proposal	Add a Footnote 1 via a superscript "1" to the end of the sentence in Section 2, and the following at the bottom of the page: 1 Because the analytical inputs, assumptions, and methods described herein are specific and sufficient, RG 1.203 need not be applied when this regulatory guidance is employed.
Justification	*RG 1.203, "Transient and Accident Analysis Methods," states, "This regulatory guide describes a process that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable for use in developing and assessing evaluation models that may be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. Evaluation models that the NRC has previously approved will remain acceptable and need not be revised to conform with the guidance given in this regulatory guide.[1]"* *However, Sections C.2.1 through C.2.5 discuss the acceptable bases for the "analytical inputs, assumptions, and methods (that) are considered acceptable for evaluation the postulated CRE and CRD accidents" as stated in Section 2. Therefore, because the NRC Staff has concluded that the specificity in DG-1327 is sufficient for developing an acceptable analytical basis for the CRE and CRD, a footnote is justified to indicate that RG 1.203 need not be applied when the guidance in this regulatory guide is employed.*

Table A.10: Specific Guidance Language Comment 10

Statement	Section C.2.1.3
Comment	Subjects discussed in DG-1327 include PWR-specific and BWR-specific topics. Separating or designating PWR-specific and BWR-specific content would avoid confusion.
Rationale	
Proposal	Re-write of DG-1327 to separate or designate all PWR-specific and BWR-specific content. The following is a list of examples of DG-1327 sections that require specific designation of PWR-specific and BWR-specific guidance: Section C.1.2: The NRC-approved GEH/GNF BWR CRDA analysis methodology does not require analysis of initial condition power levels >5% Section C.2.2.2: CZP initial conditions are applicable to BWRs only Section C.2.2.3: For GNF/GEH BWR CRDA the maximum initial power level is 5% Section C.2.2.5: The interpretation of language such as "additional fully or partially inserted misaligned or inoperable rods if allowed", and "different control rod configurations, both expected and unexpected" is different for PWRs and BWRs Section C.2.5: This is PWR-specific Section C.3.1: For GNF/GEH BRW CRDA Mode 1 is not applicable
Justification	*There are significant differences between PWR CREA and BWR CRDA that result in the DG-1327 content being confusing or subject to misinterpretation. The content should be separated and designated as PWR or BWR or PWR/BWR. In some sections of DG-1327 the NRC staff has provided specific designation (e.g., C.2.2.7). This type of designation should be included consistently throughout the final Regulatory Guide.*

Table A.11: Specific Guidance Language Comment 11

Statement	Section C.2.3.5.1
Comment	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.
Rationale	
Proposal	Extend the excess hydrogen concentration to 593 ppm in Figures 2 and 4 based on EPRI test data. The cladding failure threshold can then be extrapolated out to 593 ppm
Justification	*If an SRA cladding alloy exhibits more than 10% radial hydride reorientation, the Section C.2.3.5.1 guidance requires use of the Figures 2 and 4 RXA PCMI cladding failure thresholds. At moderate-to-high burnups, SRA cladding alloys can exceed 300 ppm excess hydrogen concentration, which is beyond the range of the RXA PCMI limit curves. This guidance results in an undefined failure limit, so Figures 2 and 4 cannot be used.* *The EPRI modified burst test database extends to 593 ppm for RXA cladding, significantly higher than the 300 ppm RXA threshold established by the NSRR LS series tests. Fuel cladding material from an LS-series sister rod was used in the MBT tests and the measured ductility tracks with the calculated NSRR failure strain.* *The EPRI proposed PCMI cladding failure thresholds for RXA cladding at high temperature (refer to Comment 2) includes the extrapolation of the failure threshold to 593 ppm excess hydrogen content (refer to Comment 4). This addresses to a large extent the range of excess hydrogen concentrations for SRA cladding.*

Table A.12: Specific Guidance Language Comment 12

Statement	Section E, Reference 13
Comment	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.
Rationale	
Proposal	Any changes in the PNNL Report 22549 content due to the corrections to the NSRR data need to be included in the final Regulatory Guide.
Justification	*PNNL Report 22549, "Pellet-Cladding Mechanical Interaction Failure Threshold for Reactivity Initiated Accidents for Pressurized Water Reactors and Boiling Water Reactors," June 2013, is Reference 5 in the NRC 3/16/2015 memo (DG-1327 Reference 8) that is the technical basis for DG-1327. Any of the PNNL Report 22549 content that is used as the technical basis for GD-1327 and that is affected by the corrections to the NSRR data, needs to be corrected prior to use in the final Regulatory Guide.*

Table A.13: Specific Guidance Language Comment 13

Statement	Section C.3.3
Comment	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.
Rationale	
Proposal	In Section C.3.3 clarify that the molten fuel cladding failure threshold is not applicable for <10% molten fuel to cladding that is ductile.
Justification	*Although there is a 20% volume change with complete fuel melting, if the cladding is ductile then it won't fail by PCMI provided the fuel melting is limited to 10% at the centerline. Assuming the 10% volumetric core coolability melt limit, the fuel volume only increases by 2%, which can be accommodated by ductile cladding in addition to the 2% from fuel thermal expansion. The cladding may be ductile at the initial conditions of the RIA event, or it may transition to ductile due to heating during the RIA.*

Table A.14: Specific Guidance Language Comment 14

Statement	Section C.3.2
Comment	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.
Rationale	
Proposal	Add the following sentence to Section C.3.2: "For the BWR CRDA event the deposited energy used for determining the margin to the Figures 2-5 PCMI cladding failure threshold is limited to the energy deposited during the prompt part of the pulse.
Justification	The PCMI cladding failure threshold is expressed in peak radial fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen concentration. The PCMI phenomenon occurs only during the prompt power pulse, and therefore only the energy deposited during the prompt part of the pulse should be used to compare with the PCMI threshold. This interpretation was accepted by the NRC in the 2007 interim criteria. Note that for BWR CZP conditions with significant subcooling the peak total enthalpy occurs well after the pulse peak when higher cladding temperature conditions occur, and is significantly higher that the prompt enthalpy due to the existence of a power tail during the RIA. Refer to EPRI report #1015206 for additional details

Table A.15: Specific Guidance Language Comment 15

Statement	Section C.2.1.1
Comment	There should not be a requirement to compare to more sophisticated spatial kinetics codes if the submitted methodology includes a 3D spatial kinetics code
Rationale	
Proposal	Section C.2.1.1 includes the following statement: *"...should be evaluated both by comparison with experiment and with more sophisticated spatial kinetics codes."* Add the following to the end of the above statement: "(if the proposed methodology does not use a 3D spatial kinetics code)".
Justification	

Table A.16: Specific Guidance Language Comment 16

Statement	Section C.3
Comment	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.
Rationale	
Proposal	Include the NRC-approved hydrogen uptake models in the final Regulatory Guide.
Justification	

Table A.17: Specific Guidance Language Comment 17

Statement	Section C.3.2
Comment	Will extrapolation of the cladding failure thresholds in Figures 2-5 to high excess hydrogen values be allowed?
Rationale	
Proposal	Add a statement to Section C.3.2 informing as to the acceptability of extrapolating to higher excess hydrogen values in Figures 2-5.
Justification	

Table A.18: Specific Guidance Language Comment 18

Statement	Section C.1.3
Comment	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.
Rationale	
Proposal	Revise Section C.1.3 to be consistent with Section C.2.3.5.1.
Justification	

Table A.19: Specific Guidance Language Comment 19

Statement	Section C.2.1.3
Comment	What is intended by the statement "Calculations should be based upon design-specific information accounting for manufacturing tolerances?"
Rationale	
Proposal	Add clarification to the scope of design-specific information that is intended in this guidance. For example, if the scope is limited to fuel design information.
Justification	

Table A.20: Specific Guidance Language Comment 20

Statement	Section C.2.3.4.1
Comment	Clarify that the hydrogen uptake models in RG 1.224 are acceptable to NRC.
Rationale	
Proposal	Add the following language to Section C.2.3.4.1: *"..(Ref. 9)* ***are accepted by NRC and*** *may be used to estimate..."*
Justification	

Table A.21: Specific Guidance Language Comment 21

Statement	Section C.3.3
Comment	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.
Rationale	
Proposal	Add RG 1.224 to the Related Guidance in Section A
Justification	

Table A.22: Specific Guidance Language Comment 22

Statement	Sections C.2.2.12, C.2.3.5.1
Comment	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". 3) Reference 10 should be "Terminal..." not "Thermal..."
Rationale	
Proposal	
Justification	

Table A.23: Specific Guidance Language Comment 23

Statement	Section C.2.3.5.2
Comment	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
Rationale	
Proposal	Provide additional guidance in Section 2.3.5.2 of an acceptable method to address hydride reorientation during power maneuvers or a reactor shutdown prior to an RIA event.
Justification	*Section C.2.3.5.2 establishes a requirement for each applicant to evaluate the potential for hydride reorientation within the cladding material. What NRC endorsed guidance exists for this type of assessment if any? If no NRC endorsed guidance exists, then the guidance should be included in this guidance document.*

Table A.24: Specific Guidance Language Comment 24

Statement	Section D
Comment	Implementation of the final RG guidance is only applicable to burnup extensions LARs.
Rationale	
Proposal	Add a statement in Section D that the regulatory guidance is not invoked unless a licensee requests a burnup extension via an LAR.
Justification	*At the recent Fuel Safety Research Meeting, an NRC staff presentation included the following statement:* *"These issues have highlighted the importance of ongoing research and experiments characterizing fuel behavior under limiting conditions PRIOR to additional burnup extensions."* *This raises a question of how much weight this statement holds with respect to the NRC's review of a voluntary license amendment, as stated on p. 22.* *"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, and (2) the specific subject matter of this regulatory guide is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this regulatory guide or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements."*

Table A.25: Specific Guidance Language Comment 25

Statement	Section C.2.2
Comment	Guidance for initial conditions requires clarification to avoid inconsistencies with ANSI/ANS N51.1 and N52.1.
Rationale	
Proposal	Revise Section C.2.2 as necessary to avoid any inconsistencies with ANSI/ANS N51.1 and N52.1.
Justification	*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 plan maneuvers or xenon oscillations*

Table A.26: Specific Guidance Language Comment 26

Statement	Sections A & D
Comment	The implementation of the requirements in DG-1327 should not be requested for licensees that employ FSAR Chapter 14/15 Safety Analysis methodologies that make use of zero- (point) or one-dimensional spatial kinetics for the analysis of the control rod ejection (CRE) for PWRs and control rod drop (CRD) for BWRs. These methodologies employ static, no feedback nuclear calculations for ejected rod worths and hot channel factors and feedback calculations to determine Doppler weighting factors (i.e., "adiabatic" assumption) and conservatively predict the margins to the fuel enthalpy limits in RG 1.77. As concluded in Reference 1 by the NRC staff, "...current operating reactors are not likely to experience cladding failure during the worst postulated RIAs." For those licensees that have maintained FSAR Chapter 14/15 safety analysis methodologies that make use of zero- (point) or one-dimensional spatial kinetics, this conclusion has not been substantially affected, and therefore the public health and safety is maintained though the continued use of the current methodologies. Conversely, should a licensee choose to implement three-dimensional space-time reactor kinetics methodologies for the analysis of the CRE and CRD, then the analytical results would be expected to provide improved margins, and via 10CFR50.59 the licensee may be required to submit the methodology for approval. Further, that licensee should implement the requirements of DG-1327.
Rationale	
Proposal	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).
Justification	The NRC issued an assessment of postulated reactivity-initiated accidents (RIAs) for operating reactors in the U.S. in Research Information Letter 0401, dated March 31, 2004 (Reference 1; Adams ML040920189). Reference 1 made the following statement in the conclusions: *"Neutronic analyses were then performed for a range of LWR conditions, and it was found that control rod worths needed to reach the enthalpy limits were very high (above $1.5). There is no comprehensive database of rod worths in U.S. power reactors. Based on available data, however, it is very unlikely for a rod worth to exceed $1.5. Therefore, it was concluded that current operating reactors are not likely to experience cladding failure during the worst postulated RIAs. Without cladding failure, coolable geometry is ensured and steam explosions cannot occur."* The NRC issued Reference 2, "Technical and Regulatory Basis for the Reactivity Initiated Accident Interim Acceptance Criteria and Guidance, in 2007 (Adams ML070220400). This document identified deficiencies in certain regulatory guidance documents (RG 1.77, 1.183, and 1.195) and defined interim acceptance criteria and revised guidance (related to fission-product gap inventory) for the RIAs. This document also provided the basis for the revision to the NUREG-0800, Standard Review Plan, Section 4.2, Fuel System Design. The development of the interim acceptance criteria evolved from the safety assessment of currently operating plants documented in Reference 1. Reference 2 focused on technical and regulatory basis of the interim acceptance criteria and guidance for the reactivity-initiated accident (RIA). Reference 2 cites the 2004 safety assessment (Reference 1), but provided no new information on the neutronic analyses, thus propagating the 2004 safety assessment forward as the continued bases for

	maintaining the public health and safety. The NRC issued Reference 3, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria And Guidance, Revision 1," in 2015 (Adams ML14188C423). Reference 3 was stated to be an update of Reference 2, but it contains significantly different information, failure modes, and acceptance criteria. It is properly viewed as a revision to Reference 2. As with Reference 2, Reference 3 focused on the technical and regulatory basis of the interim acceptance criteria and guidance. Reference 3 provided no new information on neutronic analyses, thus propagating the 2004 safety assessment forward as the continued bases for maintaining the public health and safety. The NRC then issued Draft Regulatory Guide DG-1327, "Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents," for public review and comment [Adams ML16124A200]. A public meeting was held on January 25, 2017 to discuss the content of DG-1327. During the meeting, the NRC staff indicated that there was no safety issue associated with DG-1327 for the current operating plants. NRC staff stated that the new failure acceptance criteria could not be met with the legacy methods. However, the NRC staff also acknowledged that no additional neutronic analyses had been performed since 2004 and that the conclusions from the 2004 assessment remain valid with regards to legacy methods. References 1. "An Assessment of Postulated Reactivity-Initiated Accidents (RIAs) for Operating Reactors in the U.S.," ADAMS Accession Number ML040920189. 2. NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance," ADAMS Accession Number ML0702204100, dated January 17, 2007. 3. NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1" ADAMS Accession Number ML14188C423, dated March 16, 2015.

Table A.27: Specific Guidance Language Comment 27

Statement	Section D
Comment	The technical content in a voluntary LAR invoking DG-1327 guidance needs to be specified.
Rationale	
Proposal	
Justification	*Section D includes language that the staff may request that the licensee follow the DG-1327 guidance following a voluntary LAR that is relevant. The LAR content triggers that would invoke the DG-1327 guidance are proposed as follows:* • *A new PWR control rod ejection accident or BWR control rod drop accident analysis of record* • *First use of a NRC-approved PWR control rod ejection accident or BWR control rod drop analysis methodology that is not previously referenced in the UFSAR or DCD* • *Fuel change* • *Any LAR that affects SRP 4.2 or 15.4.8 (PWR REA) or 15.4.9 (BWR CRDA)* • *This regulatory guidance is not required to be employed unless the licensee requests a burnup extension.* • *When licensee LAR uses new RIA methodology* • *Do not provide NRC with a list of triggers*

Table A.28: Specific Guidance Language Comment 28

Statement	Sections C.1.3 and C.3.2
Comment	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.
Rationale	
Proposal	In Sections C.1.3 and C.3.2 include the following statement: "For BWRs with Zr-2 RXA cladding, the PCMI limits are not applicable if criticality is restricted to ≥ 100°C (212°F).
Justification	*EPRI research has determined the brittle-to-ductile transition temperature forZr-2 RXA cladding material is ~85°C. Therefore, above that temperature the cladding cannot fail due to PCMI as the cladding can accommodate the fuel thermal expansion (~2% strain) resulting from a CRDA. For BWRs restricting control blade movement to ≥ 100°C (≥ 212°F) cladding failure due to PCMI following a CRDA is not applicable.*