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March 31, 2020

MEMORANDUM TO: Joseph E. Donoghue, Director
Division of Safety Systems
Office of Nuclear Reactor Regulation

FROM: Paul M. Clifford, Senior Technical Advisor **/RA/**
Division of Safety Systems
Office of Nuclear Reactor Regulation

SUBJECT: REGULATORY GUIDE 1.236 FUEL ROD BURNUP RANGE OF APPLICABILITY

In order to achieve improved fuel economics and plant operating costs, the commercial nuclear industry is planning on requesting approval for an increase in allowable fuel rod burnup to 68 GWd/MTU (rod average). The purpose of this memorandum is to document the staff's technical justification of an extended fuel rod burnup range of applicability up to 68 GWd/MTU (rod average) for the guidance in draft Regulatory Guide (RG) 1.236, *Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents*. This draft RG describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission considers acceptable when analyzing a postulated control rod ejection accident for pressurized water reactors and a postulated control rod drop accident for boiling water reactors. It defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet clad mechanical interaction. It also describes analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits. To facilitate implementation, draft RG 1.236 also provides acceptable analytical models for cladding hydrogen uptake and transient fission gas release.

Enclosure:
Technical Bases

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DATED: March 31, 2020

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Technical Justification for the Fuel Rod Burnup Range of Applicability for Regulatory Guide 1.236

PURPOSE:

In order to achieve improved fuel economics and plant operating costs, the commercial nuclear industry is planning on requesting approval for an increase in allowable fuel rod burnup to 68 GWd/MTU (rod average). The purpose of this memorandum is to document the staff's technical justification of an extended fuel rod burnup range of applicability up to 68 GWd/MTU (rod average) for the guidance in draft Regulatory Guide (RG) 1.236, *Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop Accidents*. This draft RG describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when analyzing a postulated control rod ejection (CRE) accident for pressurized water reactors (PWRs) and a postulated control rod drop accident for boiling water reactors (BWRs). It defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet clad mechanical interaction (PCMI). It also describes analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits. To facilitate implementation, draft RG 1.236 also provides acceptable analytical models for cladding hydrogen uptake and transient fission gas release (FGR).
Section 1 of draft RG 1.236 defines the limits on applicability for the fuel rod cladding failure thresholds, fission product release fractions, and allowable limits on damaged core coolability provided in this guidance. This paper provides the technical justification for the following range of applicability:

1.1.3 The applicability of this guidance is limited to a maximum fuel rod average burnup of 68 GWd/MTU.

BACKGROUND:

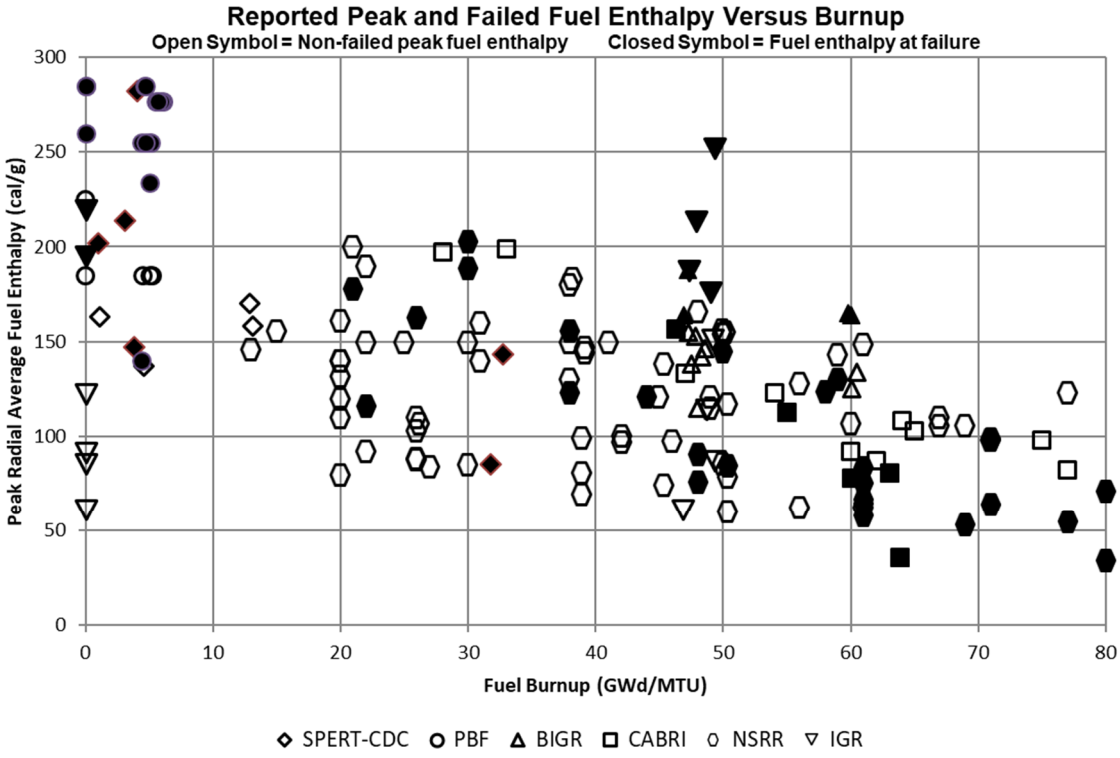
The NRC staff initially provided guidance for PWR CRE in RG 1.77 in 1974 (Reference 1). The state of knowledge of fuel rod performance under prompt power excursion conditions has increased significantly since publication of that guidance. This knowledge has prompted the need for new guidance to build on the enhanced database drawn from operating experience and controlled experiments. The empirical database has expanded from the earlier Special Power Excursion Test Reactor and Transient Reactor Test Facility research programs (which formed the basis of the initial RG 1.77 analytical limits) to include test results from the Power Burst Facility as well as significant, more recent contributions from international research programs at the CABRI research reactor (France), Nuclear Safety Research Reactor (NSRR) (Japan), Impulse Graphite Reactor (IGR) (Russian Federation), and Fast Pulse Graphite Reactor (BIGR) (Russian Federation). In 2007, the staff provided interim acceptance criteria and guidance in Appendix B of Section 4.2 of NUREG-0800. The basis for the revision was provided in NRC Memorandum, "Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance" (Reference 2). In 2015, the staff evaluated newly published empirical data and analyses and identified further changes to guidance in the NRC memorandum, "Technical and Regulatory Basis for the Reactivity Initiated Accident Acceptance Criteria and Guidance, Revision 1" (Reference 3).

Draft RG 1.236 captures the current state-of-knowledge from all of the RIA experiments to date. RG 1.236, formerly known as DG-1327, underwent two rounds of public comments. Disposition of the public comments and any changes to the guidance prompted by public comments is documented in References 4 and 5.

DISCUSSION:

The staff's technical and regulatory bases document (Reference 3) details the extent of the available empirical database from past and ongoing prompt-pulse research programs. Figure 1 illustrates the extent of the database, as a function of peak radial average fuel enthalpy rise verses fuel burnup. In general, fuel rod burnup is reported as the local burnup of the test specimen (i.e., active length of fuel within test rodlet) which ranges from 4 – 36 inches in length. Hence, this burnup is more representative of a peak nodal or peak pellet exposure than a rod average burnup. The relationship between rod average and peak local burnup is influenced by many parameters, especially plant type (BWR vs PWR) and axial loading (i.e., natural uranium and poison cutback regions, axial ^{235}U enrichment zoning). For the purpose of this paper, a 10 percent difference between the reported specimen burnup and rod average burnup is assumed. Hence, to support the target burnup applicability limit of 68 GWd/MTU (rod average), the staff will investigate the sensitivity of the guidance based upon trends in the empirical data up to 75 GWd/MTU (local burnup). Using the same adjustment, the current burnup limit of 62 GWd/MTU (rod average) correlates to a local burnup of approximately 68 GWd/MTU. Examination of Figure 1 reveals ten tests conducted on fuel rod segments beyond the current burnup limit.

Figure 1: RIA Empirical Database



Cladding Failure Thresholds

RG 1.236 defines analytical thresholds for fuel rod cladding failure based on two failure modes: high temperature cladding failure and PCMI failure. The cladding failure thresholds are provided as a function of cladding differential pressure and cladding excess hydrogen. **Both parameters are related to, but not strictly defined by, fuel rod burnup**. Each failure mode has its own unique empirical database. The extent of the supporting database up to the target burnup of 68 GWd/MTU (rod average) and the sensitivity of the failure mode to burnup is discussed below.

High Temperature Cladding Failure:

The high temperature cladding failure threshold curve is a composite that covers both high temperature brittle failure (i.e., embrittlement due to oxygen ingress) and ductile failure (i.e., balloon and rupture). Figure 2 shows the proposed failure threshold along with the supporting empirical database. The analytical failure threshold is expressed as peak radial average fuel enthalpy (cal/g) as a function of cladding differential pressure (ΔP). While developing this curve, the staff assessed the sensitivity of high temperature cladding failure to multiple parameters including fuel burnup. Figure 3 shows the empirical database as a function of fuel burnup. Examination of Figure 3 reveals that the failure threshold, peak radial average fuel enthalpy (cal/g), **does not appear sensitive to fuel burnup**. Whereas, as shown on Figure 2, the ductile failure mode is **sensitive to cladding differential pressure** and continues to decrease until a terminal failure threshold at 100 cal/g. With respect to the extent of the empirical database, examination of Figure 3 reveals four test results beyond 68 GWd/MTU (local) with one test at or beyond 75 GWd/MTU. **None of the extended burnup tests experienced cladding failure and these tests were subjected to peak fuel enthalpy as high as 123 ca/g which is above the terminal failure threshold of 100 cal/g.**

Based on the observation that the failure modes do not appear to be burnup dependent and acceptable performance of the limited extended burnup test results (relative to the proposed failure threshold), the staff finds the applicability of this failure threshold up to 68 GWd/MTU rod average burnup acceptable.

This finding is predicated on applicants using approved core neutronics and fuel rod thermal-mechanical models capable of conservatively predicting radial average fuel enthalpy, fission gas release and rod internal pressure up to 68 GWd/MTU (rod average).

Figure 2: High Temperature Cladding Failure Threshold

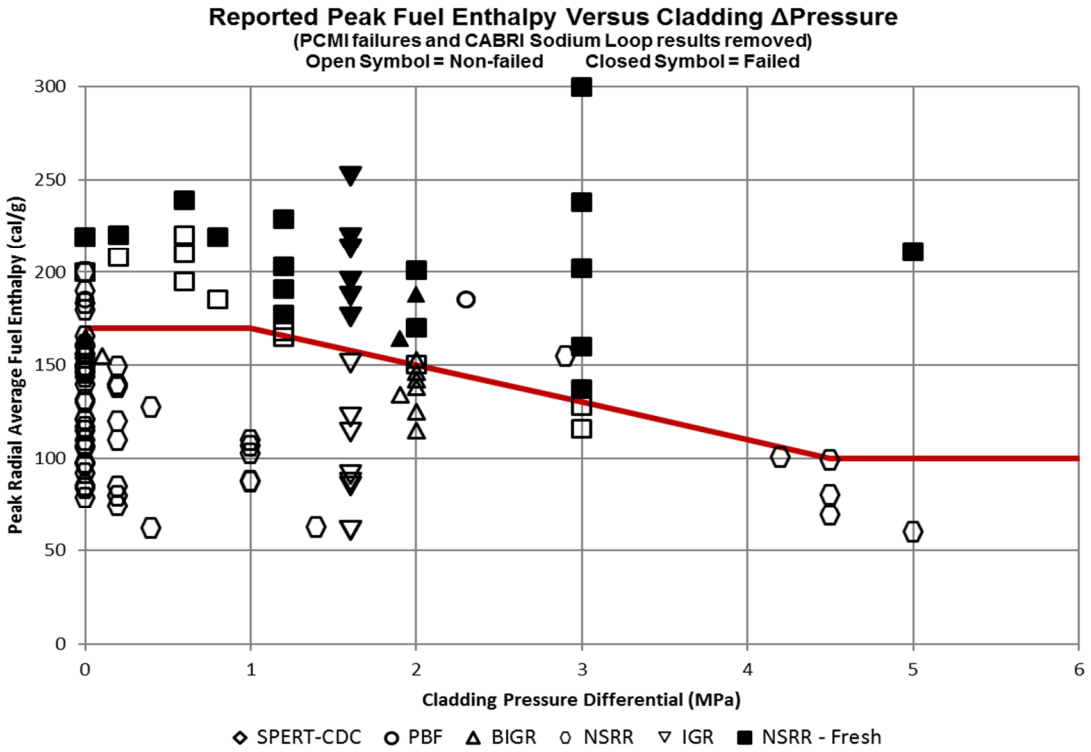
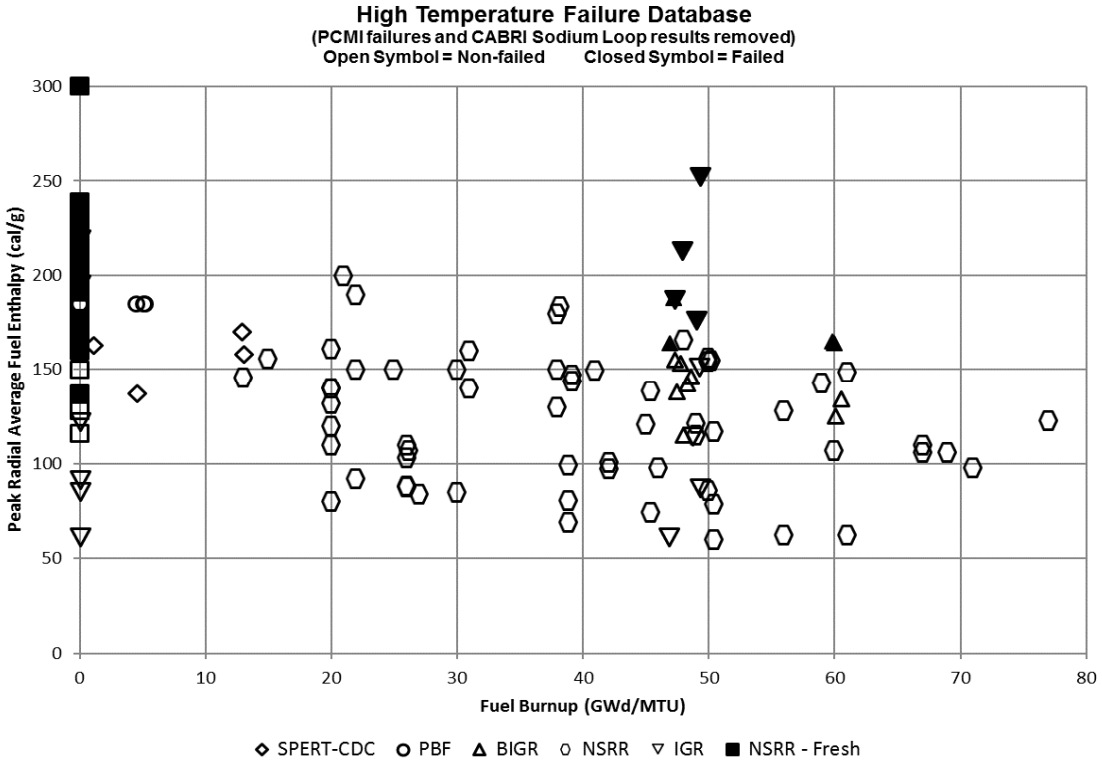


Figure 3: High Temperature Cladding Failure Empirical Database



PCMI Cladding Failure:

Based on the sensitivity of PCMI failure with zirconium hydride concentration, distribution, and orientation, separate cladding failure threshold curves were developed for stress relief annealed (SRA) and fully recrystallized annealed (RXA) cladding. Figure 4 shows the proposed SRA cladding PCMI failure threshold along with the supporting empirical database. The analytical failure threshold is expressed as peak radial average fuel enthalpy rise (∆cal/g) as a function of cladding excess hydrogen (i.e., zirconium hydrides).

While developing the PCMI failure curves, the staff assessed the sensitivity of PCMI cladding failure to multiple parameters including fuel burnup. **Burnup related changes in the microstructure of the fuel pellet are expected to influence the pellet's performance under RIA conditions**. For example, higher burnup will promote a lower melting temperature, enhanced gaseous swelling, and an edge peaked power profile. With respect to irradiated cladding properties and performance, irradiation-induced embrittlement is known to saturate relatively early in life and the empirical database suggests that hydrogen-induced embrittlement also tends to saturate. As shown in Figure 4, the proposed analytical failure threshold approaches an end-of-life asymptote of 65 Δcal/g.

Figures 5 and 6 show the SRA cladding PCMI failure empirical database as a function of enthalpy rise versus burnup and excess hydrogen versus burnup. A comparison of Figure 4 and Figure 5 reveals that **the failure threshold, peak radial average fuel enthalpy rise (Δcal/g), is more sensitive to, and more accurately related to, excess cladding hydrogen (wppm) than fuel burnup**.

With respect to the extent of the empirical database, examination of Figure 5 reveals seven test results beyond 68 GWd/MTU (local) with four tests at or beyond 75 GWd/MTU. **None of the extended burnup tests experienced cladding failure below 65 Δcal/g**. **However, this observation alone does not support extended burnup because it does not consider low corrosion zirconium alloys which absorb less hydrogen and remain well above 65 Δcal/g on this failure threshold curve**.

Examination of Figure 6 reveals that all extended burnup test specimens were at or above 300 wppm excess hydrogen. This is expected given the duration at operating temperatures needed to acquire high burnup. Nevertheless, it is difficult to distinguish between potential extended burnup effects and the known degradation in cladding ductility with hydrogen absorption. However, the empirical database contains 25 tests performed on specimens with less than 150 wppm excess hydrogen, with fuel burnup ranging from 13 – 64 GWd/MTU, and radial average fuel enthalpy rise ranging from 36 – 200 ∆cal/g. Except for CABRI test CIP-Q (46.3 GWd/MTU, 105 wppm), which is thought to have been influenced by atypical test conditions, none of these tests experienced cladding failure. Even though burnup related changes in the pellet microstructure accumulated between 13 and 64 GWd/MTU, these test results **suggest that PCMI failure may not be strongly dependent on fuel rod burnup**.

Based on a weak dependence with accumulated burnup and acceptable performance of the limited extended burnup test results (relative to the proposed failure threshold), the staff finds the applicability of the SRA cladding PCMI failure threshold up to 68 GWd/MTU rod average burnup acceptable.

Figure 4: SRA Cladding PCMI Failure Threshold

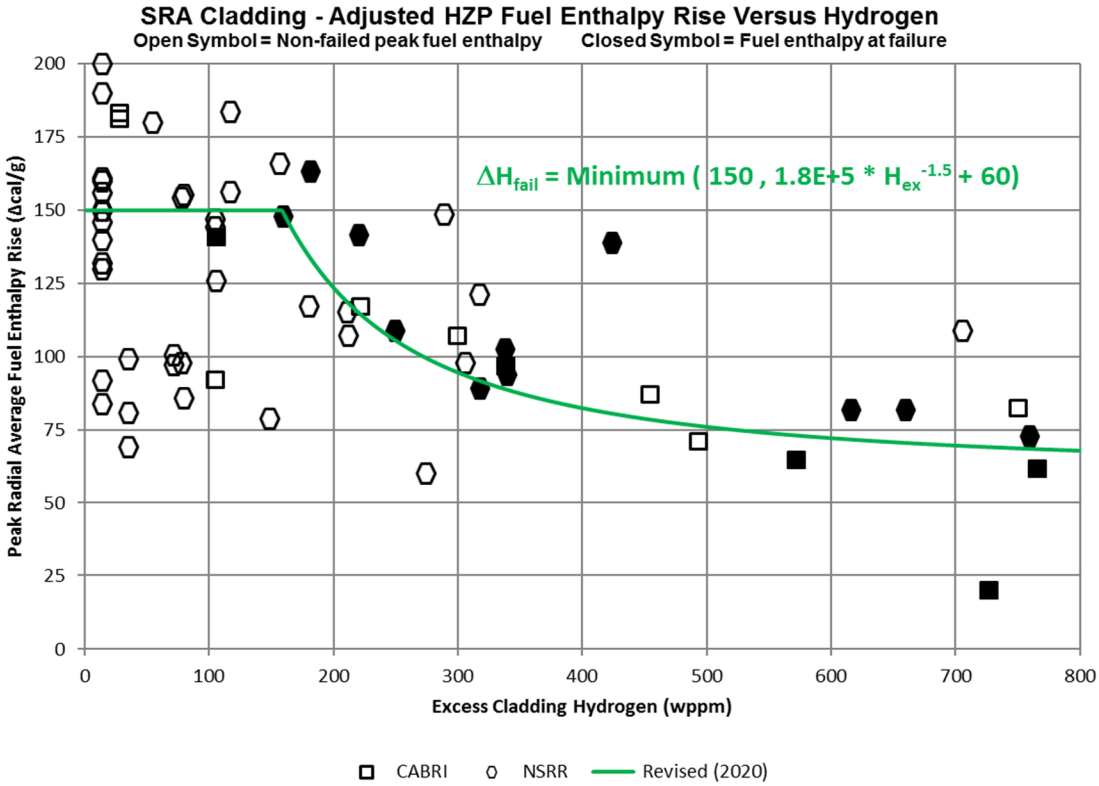


Figure 5: SRA Cladding PCMI Failure Empirical Database – Enthalpy Rise vs. Burnup

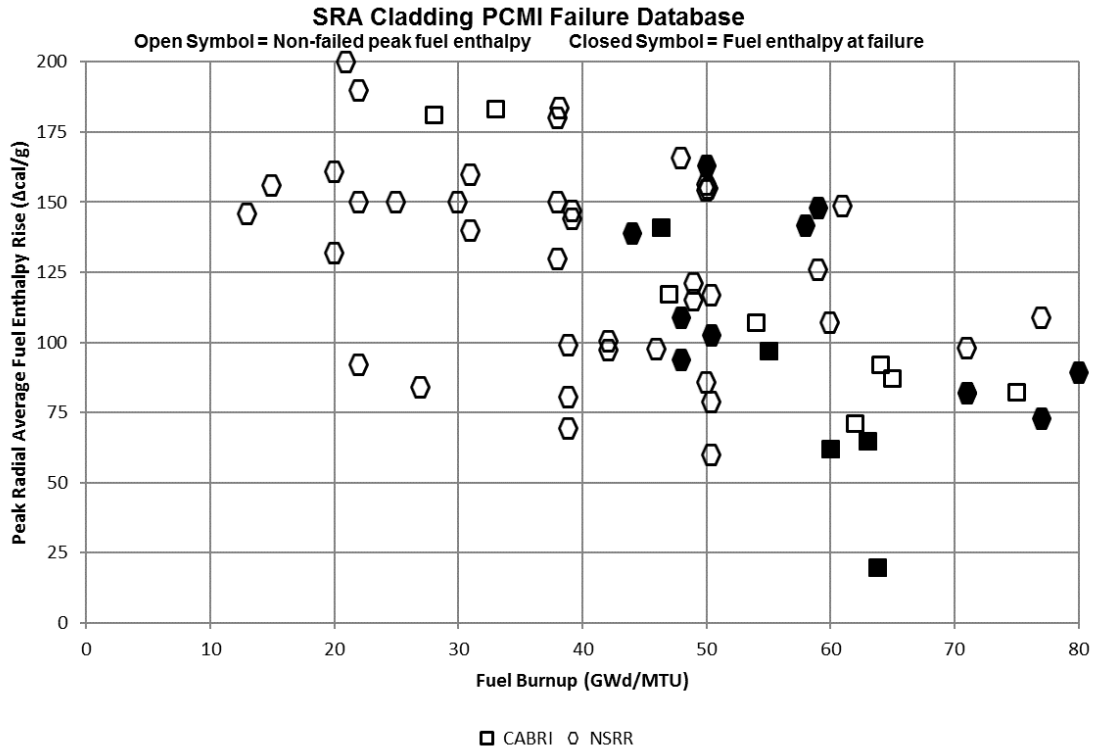
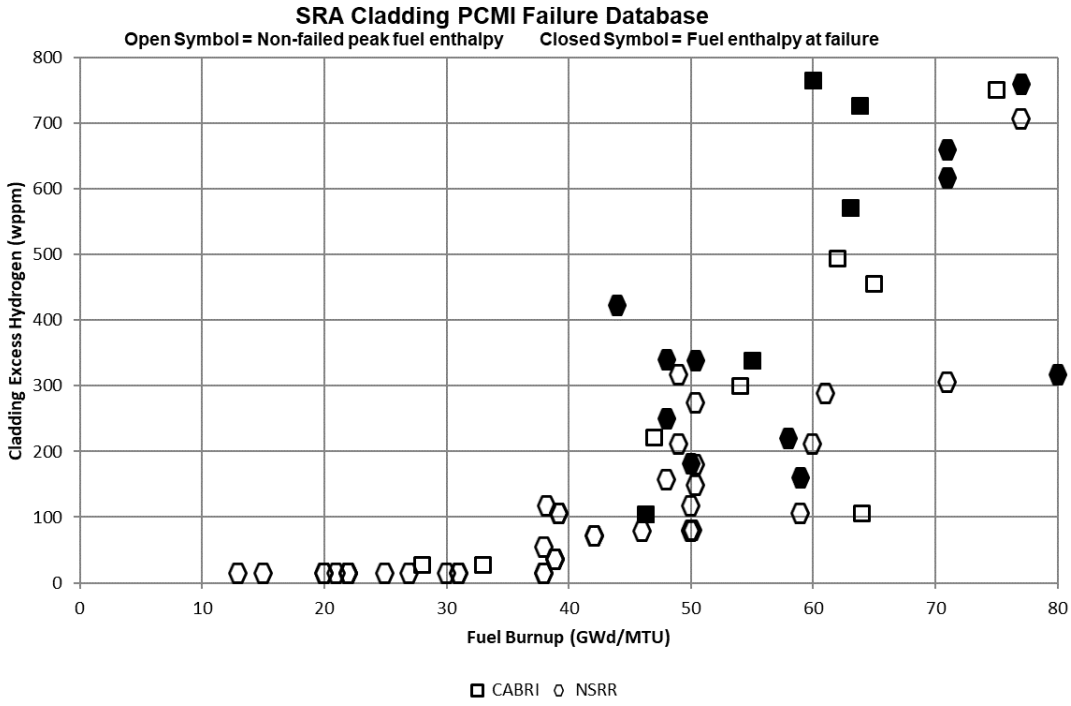


Figure 6: SRA Cladding PCMI Failure Empirical Database – Excess Hydrogen vs. Burnup



For RXA cladding, Figure 7 shows the proposed analytical PCMI failure threshold along with the supporting empirical database. Examination of this figure reveals that **RXA cladding is even more sensitive to cladding excess hydrogen content** with the analytical failure enthalpy rise beginning to decrease below 100 wppm and quickly approaching an end-of-life asymptote of 50 Δcal/g.

Figures 8 and 9 show the extent of the RXA cladding PCMI failure empirical database as a function of enthalpy rise versus burnup and excess hydrogen versus burnup. Examination of these plots show that while the higher burnup specimens fail at lower enthalpy, these same specimens have progressively more cladding excess hydrogen. Hence, these is **no apparent sensitivity to fuel rod burnup**. With respect to the extent of the empirical database, examination of these figures reveals four test results beyond 68 GWd/MTU (local) with two tests at or beyond 75 GWd/MTU. **None of the extended burnup tests experienced cladding failure below 50 Δcal/g. However, as with SRA cladding, this observation alone does not support extended burnup because it does not consider low corrosion zirconium alloys which absorb less hydrogen and remain well above 50 Δcal/g on the RXA failure threshold curve.**

The RXA cladding PCMI failure empirical database contains 24 tests performed on specimens with less than 100 wppm excess hydrogen, with fuel burnup ranging from 20 – 77 GWd/MTU, and radial average fuel enthalpy rise ranging from 57 – 150 ∆cal/g. **None of the tests experienced cladding failure.** Even though burnup related changes in the pellet microstructure accumulated between 20 and 77 GWd/MTU, these test results **suggest that PCMI failure may not be strongly dependent on fuel rod burnup**.

Based on a weak dependence with accumulated burnup and acceptable performance of the limited extended burnup test results (relative to the proposed failure threshold), the staff finds the applicability of the RXA cladding PCMI failure threshold up to 68 GWd/MTU rod average burnup acceptable.

This finding is predicated on applicants using approved core neutronics and fuel rod thermal-mechanical models capable of conservatively predicting radial average fuel enthalpy and cladding absorbed hydrogen content up to 68 GWd/MTU (rod average).

While not specifically described above, it is important to note that excess cladding corrosion will promote localized effects (e.g., spallation, hydrogen blisters) which have been shown to significantly reduce the cladding failure threshold. **Hence, the applicability of the high temperature and PCMI cladding failure thresholds to any fuel burnup, including extended burnup up to 68 GWd/MTU (rod average), is limited to fuel rod designs, cladding alloys, and plants which control and limit oxide thickness to prevent these localized effects.**

Figure 7: RXA Cladding PCMI Failure Threshold

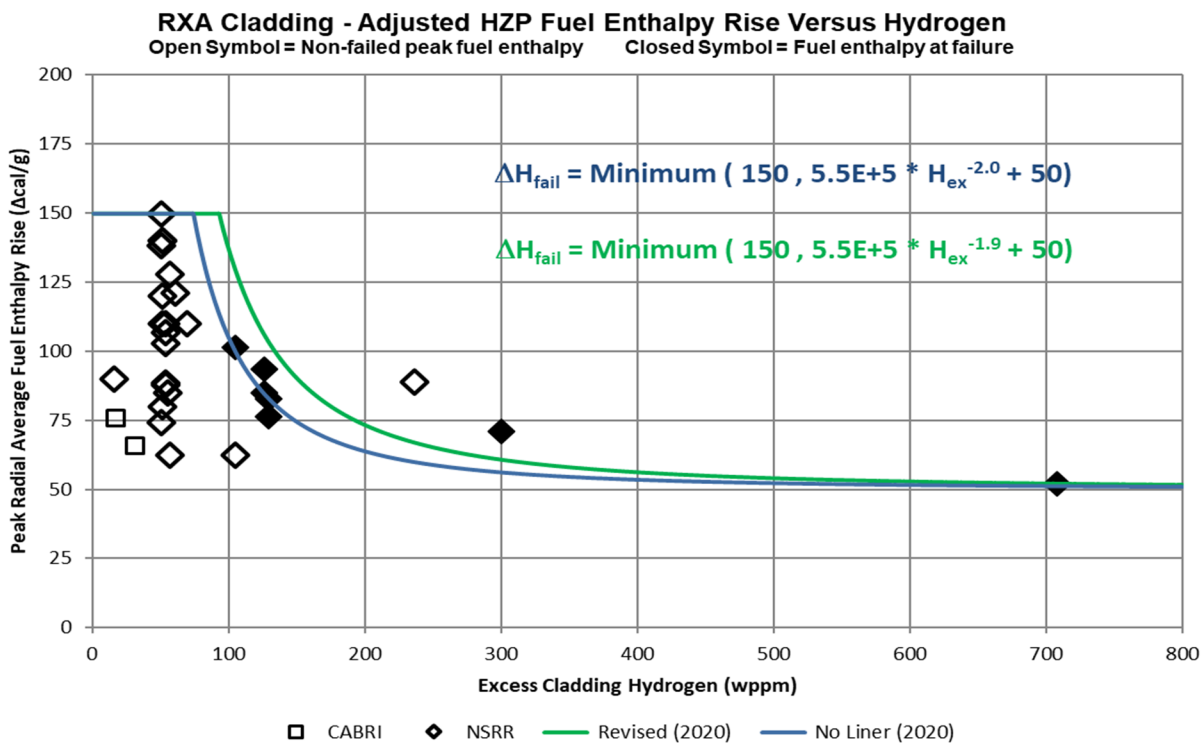


Figure 8: RXA Cladding PCMI Failure Empirical Database – Enthalpy Rise vs. Burnup

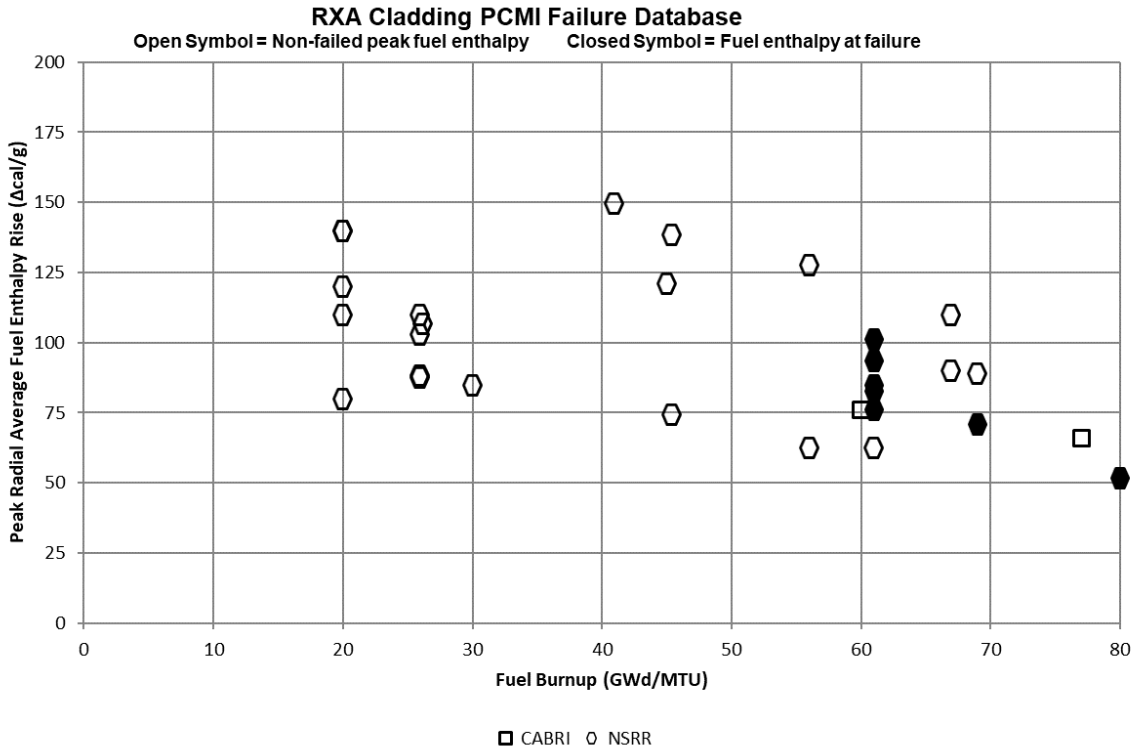
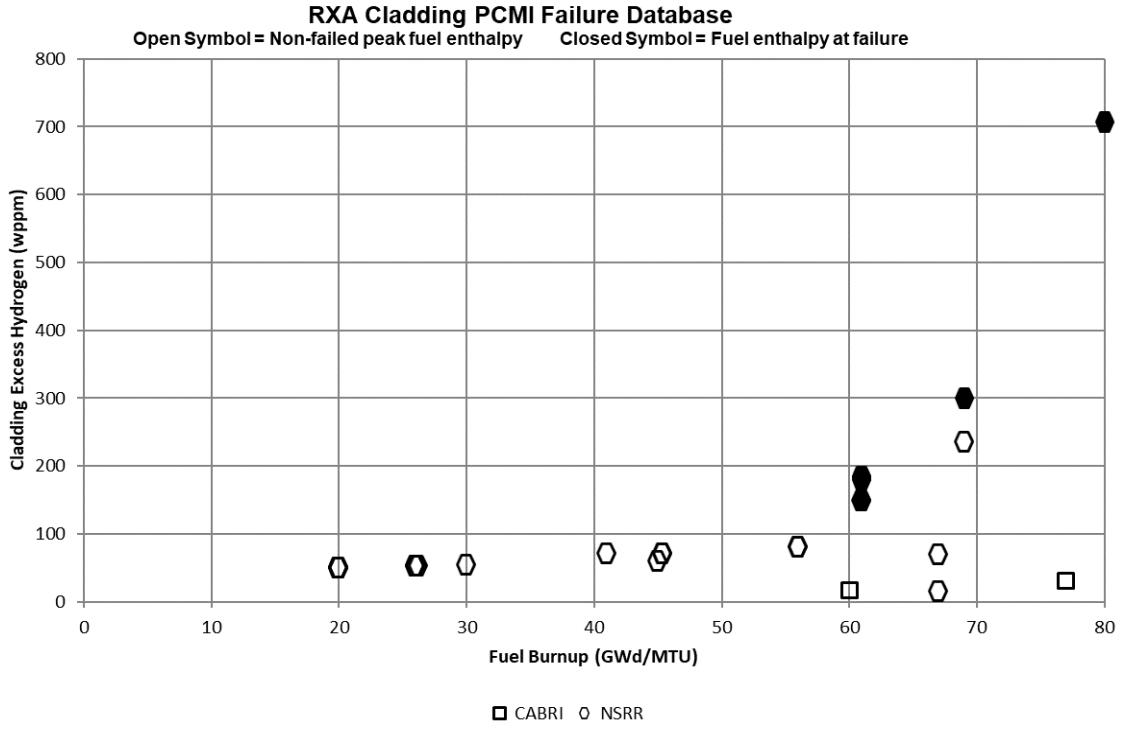


Figure 9: RXA Cladding PCMI Failure Empirical Database – Excess Hydrogen vs. Burnup



Damaged Core Coolable Geometry

RG 1.236 defines the following acceptable analytical limits to preserve a damaged core coolable geometry.

> Limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel coolant interaction is an acceptable metric to demonstrate that there is limited damage to core geometry and that the core remains amenable to cooling. The following restrictions should be met:
>
> a. Peak radial average fuel enthalpy should remain below 230 cal/g.
>
> b. A limited amount of fuel melting is acceptable provided that it is less than 10 percent of fuel volume. If fuel melting occurs, the peak fuel temperature in the outer 90 percent of the fuel volume should remain below incipient fuel melting conditions.

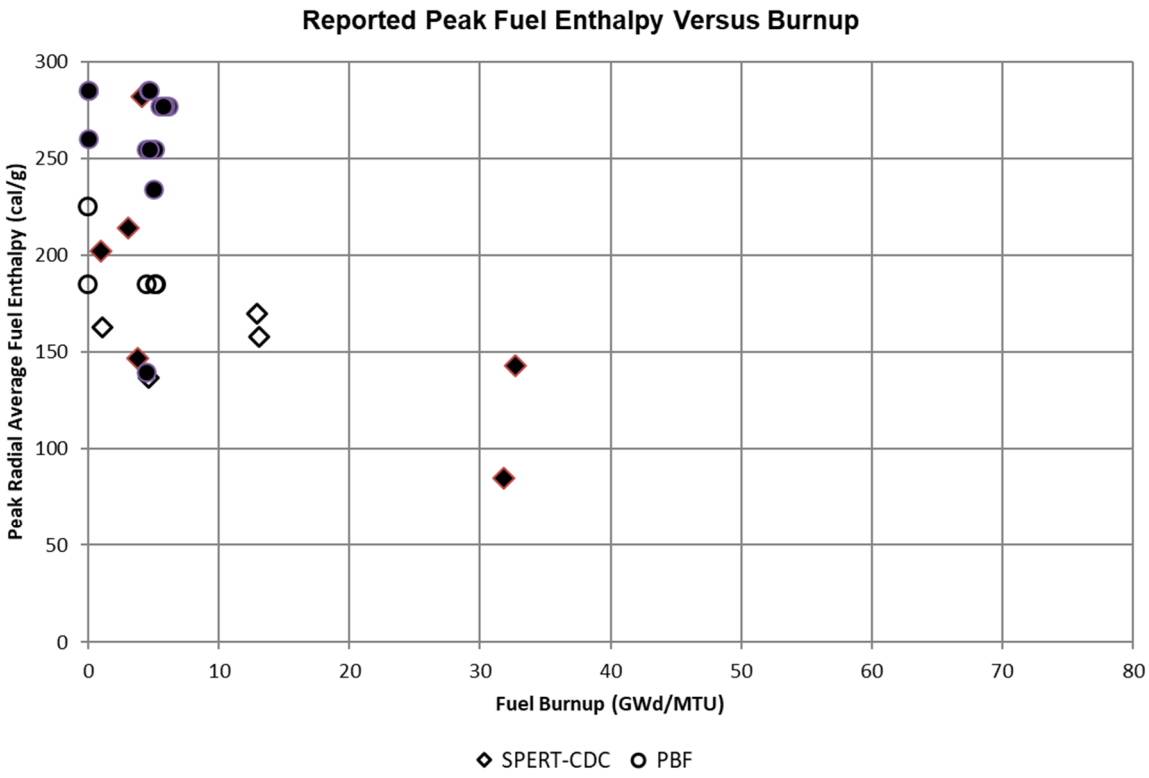
Unlike the cladding failure empirical database, the prompt-pulse empirical database used to assess damaged core coolable geometry has not been expanded and continues to rely on high enthalpy testing performed several decades ago. The testing shown in Figure 10 was used to establish the baseline coolable geometry limit of 230 cal/g peak radial average fuel enthalpy.

Burnup related changes in the microstructure of the fuel pellet are expected to influence the pellet's performance under RIA conditions. For example, higher burnup will promote a lower melting temperature, enhanced gaseous swelling, and an edge peaked power profile. For fresh and low-burnup fuel rods, the 230 cal/g peak radial average fuel enthalpy restriction will likely be more limiting than the limited fuel melt restriction. However, because of the effects of edge-peaked pellet radial power distribution and lower solidus temperature, medium- to high-burnup fuel rods are more likely to experience fuel melting in the pellet periphery under prompt power excursion conditions. For these medium- to high-burnup rods, fuel melting outside the centerline region should be precluded, and this restriction will likely be more limiting than the peak radial average fuel enthalpy restriction.

While the damaged core coolable geometry empirical database does not include any high-burnup fuel rod segments, **the detrimental effects of higher burnup must be accounted for to satisfy the limited fuel melt restriction**. Hence, the maximum allowable radial average fuel enthalpy decreases with any burnup extension. Based on the above, the staff finds the applicability of the damaged core coolable geometry analytical limits up to 68 GWd/MTU rod average burnup acceptable.

Fuel fragmentation, relocation, and dispersal (FFRD) is a phenomenon which challenges coolable geometry and has been shown to be sensitive to fuel burnup. The susceptibility of fuel pellets to fragment into fine particles increases with burnup. **RG 1.236 does not provide guidance related to an acceptable treatment of FFRD nor does this paper address the impacts of FFRD on coolable geometry up to 68 GWd/MTU.**

Figure 10: Damaged Core Coolable Geometry



Transient Fission Gas Release

RG 1.236 provides guidance on transient FGR. Unlike steady-state FGR (into the rod plenum) which is controlled by diffusion during normal operations, pellet fracturing, and grain boundary separation are the primary mechanisms for FGR during the transient. The release of additional fission gas during the transient will contribute to higher rod internal pressure.

Figure 11 provides acceptable transient FGR correlations along with the supporting empirical database. 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.

Figures 12 and 13 shows the extent of the transient FGR empirical database. With respect to the extent of the high-burnup database, examination of these figures reveals four test results beyond current burnup limits (68 GWd/MTU local) with two tests at or beyond 75 GWd/MTU. **All of the extended burnup data points fall below the transient FGR correlation**. Given that two of the extended burnup tests experienced a peak radial average fuel enthalpy at or above 130 cal/g, these tests provide reasonable assurance that the transient FGR correlation is valid up to 68 GWd/MTU.

Figure 11: Transient Fission Gas Release Correlation

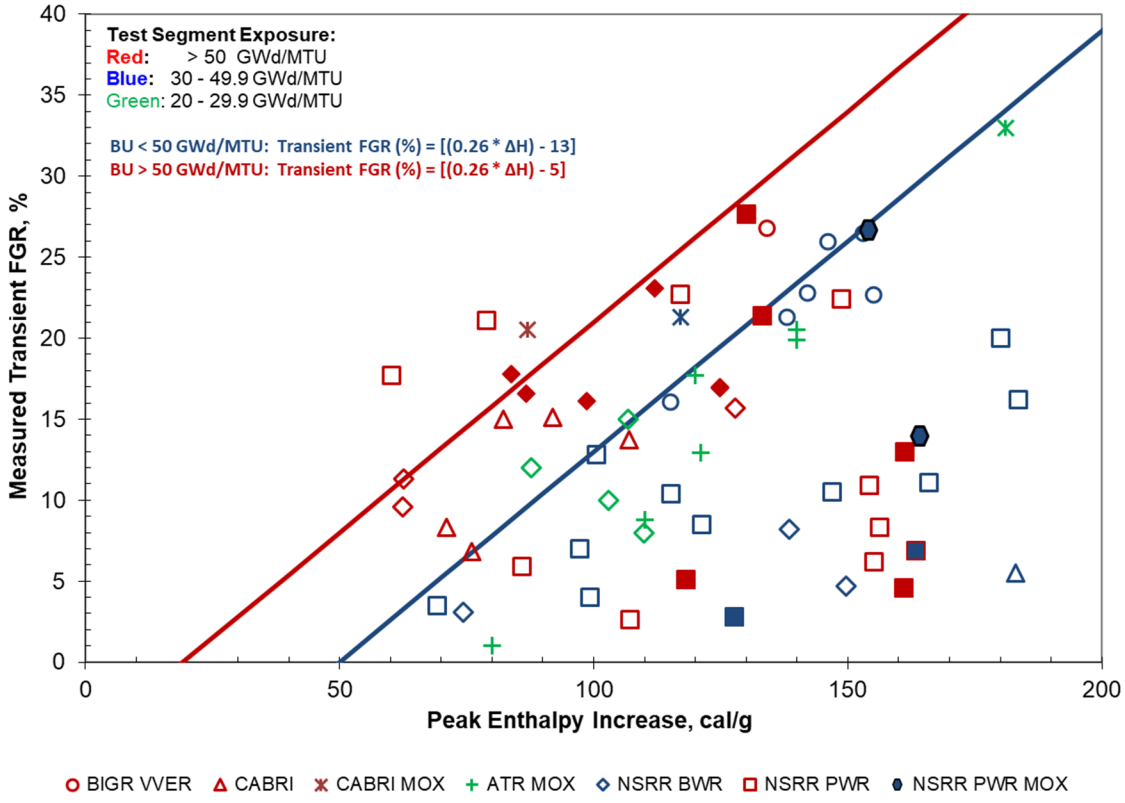


Figure 12: Transient Fission Gas Release Database – FGR vs. Burnup

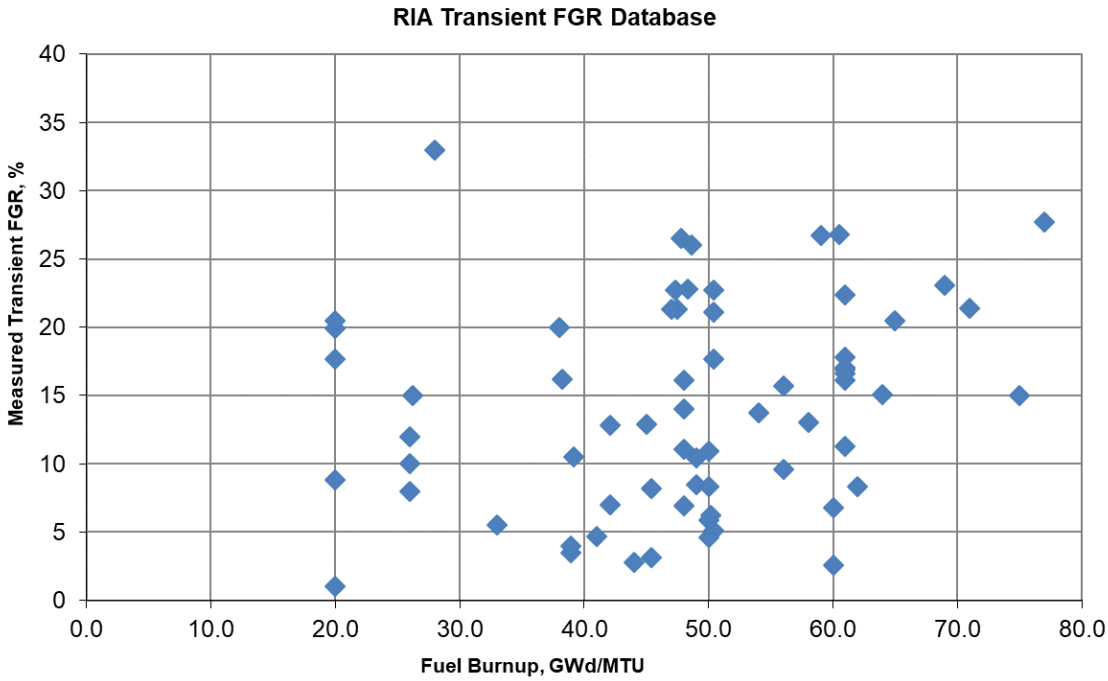
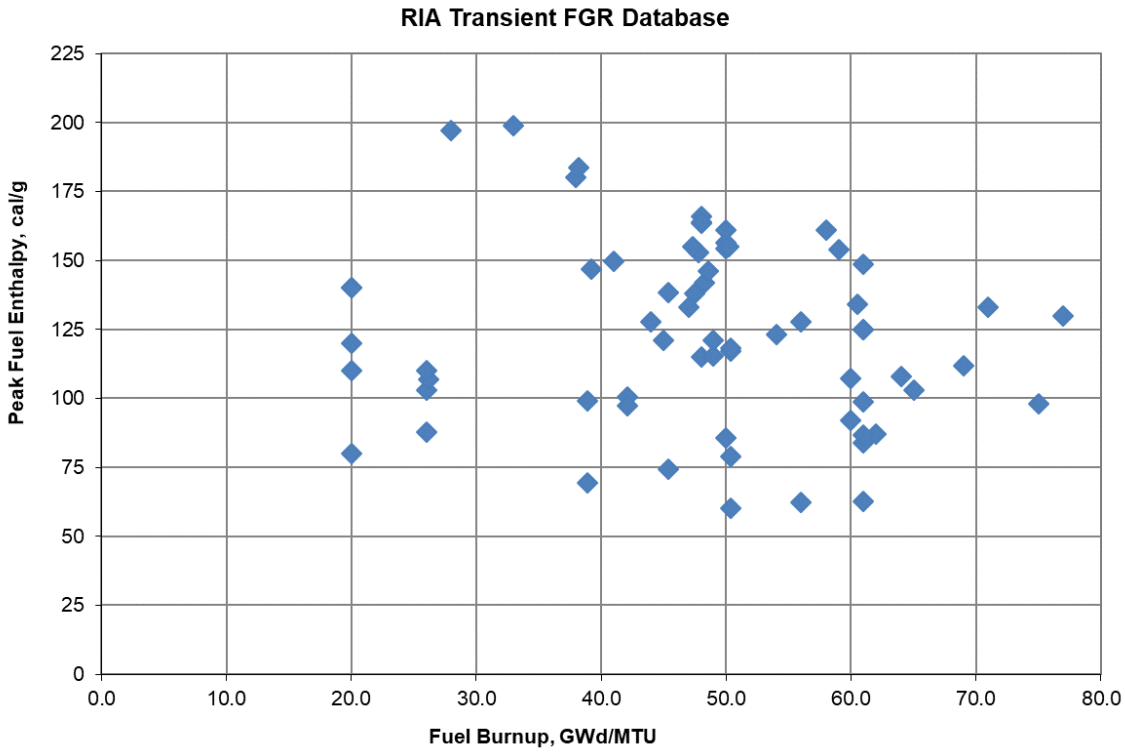


Figure 13: Transient Fission Gas Release Database – Enthalpy vs. Burnup



DATA GAPS FOR FUEL BURNUP BEYOND 68 GWd/MTU ROD AVERAGE

The nuclear industry is also considering a further increase in allowable fuel burnup up to 75 GWd/MTU rod average (82-85 GWd/MTU local burnup). As shown in Figure 1, the RIA empirical database is limited at extremely high burnup with only five tests above 75 GWd/MTU (local) and none above 80 GWd/MTU (local). To support an allowable fuel burnup limit of 75 GWd/MTU (rod average), the following data gaps need to be addressed:

1. Testing on high and extended burnup fuel rod segments with deposited energy beyond predicted cladding damage needed to investigate FFRD and loss of coolable geometry. These tests should also measure transient FGR. The influence of pulse widths should be investigated, ranging from bounding (i.e., narrow) prompt pulses to broader pulses more representative of at-power conditions.

2. Testing on irradiated, doped UO2 fuel pellets and IFBA fuel pellets needed to better understand impact of additive agents (e.g., larger grain size, retained fission gas, grain boundary hold-up, thermal conductivity) on cladding failure thresholds, FFRD, transient FGR, and coolable geometry.

3. Testing on high and extended burnup fuel rod segments, especially RXA cladding, with low corrosion needed to better understand burnup-effects and define cladding failure thresholds.

4. Testing on RXA cladding at all corrosion levels needed to better define PCMI failure threshold.

5. Neutronics analyses needed to better understand any new or expanded analytical boundaries (e.g., higher rod/blade worth, reactor kinetics, MTC, PDIL) associated with reactor operation with extended burnup and ^{235}U enrichment.

CONCLUSIONS:

To support an industry initiative to extended allowable fuel rod burnup to 68 GWd/MTU (rod average), the staff completed a detailed assessment of the applicability of guidance and analytical limits within draft RG 1.236. Based upon the extent of the empirical database and sensitivity of important parameters and phenomena to burnup, the staff finds RG 1.236 to be applicable up to a fuel rod average burnup of 68 GWd/MTU.

This assessment is predicated on the use of approved core neutronics and fuel rod thermal-mechanical models. The following limitations were identified:

1. Excess cladding corrosion will promote localized effects (e.g., spallation, hydrogen blisters) which have been shown to significantly reduce the cladding failure threshold. Hence, the applicability of the high temperature and PCMI cladding failure thresholds to any fuel burnup, including extended burnup up to 68 GWd/MTU rod average, is limited to fuel rod designs, cladding alloys, and plants which control and limit oxide thickness to prevent these localized effects.

2. Fuel fragmentation, relocation, and dispersal (FFRD) is a phenomenon which challenges coolable geometry and has been shown to be sensitive to fuel burnup. The susceptibility of fuel pellets to fragment into fine particles increases with burnup. RG 1.236 does not provide guidance related to an acceptable treatment of FFRD nor does this paper address the impacts of FFRD on coolable geometry.

Finally, data gaps associated with fuel burnup extensions beyond 68 GWd/MTU (rod average) were identified.

REFERENCES:

1. NRC, RG 1.77, “Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors,” Washington, DC.

2. NRC, Memorandum, Ralph Landry to Thomas Martin, “Technical and Regulatory Basis for the Reactivity-Initiated Accident Interim Acceptance Criteria and Guidance,” January 19, 2007 (ADAMS Accession No. ML070220400).

3. NRC, Memorandum, Paul M. Clifford to Timothy J. McGinty, “Technical and Regulatory Basis for the Reactivity-Initiated Accident Acceptance Criteria and Guidance, Revision 1,” March 16, 2015 (ADAMS Accession No. ML14188C423).

4. NRC, Memorandum, Paul M. Clifford to Mirela Gavrilas, “DG-1327 Comment Resolution Table,” September 6, 2018 (ADAMS Accession No. ML18249A346).

5. NRC, “DG-1327, “Response to Second Round of Public Comments Draft Guide (DG)-1327” (ADAMS Accession No. ML20055F489).