Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Thermal Hydraulic Phenomena Subcommittee
Docket Number:	(n/a)
Location:	teleconference
Date:	Tuesday, May 5, 2020

Work Order No.: NRC-0902 Pages 1-112

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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THERMAL HYDRAULIC PHENOMENA SUBCOMMITTEE

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TUESDAY

MAY 5, 2020

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The Subcommittee met via Teleconference, at 1:00 p.m. EST, Walter Kirchner, Chairman, presiding.

COMMITTEE MEMBERS:

WALTER KIRCHNER, Chairman

RONALD G. BALLINGER, Member

DENNIS BLEY, Member

JOSE MARCH-LEUBA, Member

DAVID A. PETTI, Member

JOY REMPE, Member

ACRS CONSULTANT:

MICHAEL CORRADINI

STEVE SCHULZ

DESIGNATED FEDERAL OFFICIAL:

ZENA ABDULLAHI

ALSO PRESENT:

PAUL CLIFFORD, NRR

TOM EICHENBERG, Tennessee Valley Authority

T-A-B-L-E O-F C-O-N-T-E-N-T-S

P R O C E E D I N G S

1:04 p.m.

CHAIR KIRCHNER: Okay. Let me ceremoniously bang the table with a gavel. The meeting will now come to order. This is a meeting of the ACRS Metallurgy and Reactor Fuels Subcommittee and Thermal Hydraulics Subcommittee of the Advisory Committee on Reactor Safeguards. I am Walter Kirchner, Chairman of today's subcommittee meeting. ACRS members in attendance are -- and at this point, I'll just ask for an affirmation if you're there. Ron Ballinger?

MEMBER BALLINGER: Yep, I'm here.

CHAIR KIRCHNER: Dennis Bley?

(No response.)

CHAIR KIRCHNER: Jose March-Leuba?

MEMBER MARCH-LEUBA: Yes, I'm here.

CHAIR KIRCHNER: David Petti?

MEMBER PETTI: Here.

CHAIR KIRCHNER: Joy Rempe?

MEMBER REMPE: Here.

CHAIR KIRCHNER: Matt Sunseri?

PARTICIPANT: Walt, Matt said that he was not going to be able to participate.

CHAIR KIRCHNER: Okay. Thank you. And I

also note that Pete Riccardella would not be able -- he had a conflict as well. Vesna Dimitrijevic?

MEMBER BALLINGER: Vesna said that she was not going to be able to attend.

CHAIR KIRCHNER: Okay. Thank you. Now have I left anyone out? Joy Rempe?

MEMBER REMPE: You already asked, and yes, I'm here.

CHAIR KIRCHNER: Okay. Sorry, Joy. All right.

MEMBER REMPE: Not a problem.

CHAIR KIRCHNER: All right. And consultants, I believe we have Mike Corradini?

MR. CORRADINI: Yes.

CHAIR KIRCHNER: And Steve Schulz?

MR. SCHULZ: I'm here.

CHAIR KIRCHNER: Thank you. Okay. Zena Abdullahi is the Designated Federal Official for this meeting. During today's meeting, the subcommittee will hear presentations and will hold discussions with the NRC staff regarding Regulatory Guide 1.236, Pressurized Water Reactor Control Rod Ejection and Boiling Water Reactor Control Rod Drop. There's a long history related to this Regulatory Guide. The staff was -- the staff briefed the ACRS Committee on

the draft Regulatory Guide 1327, I believe back as far as 2007 and again in October 2016.

In today's meeting, the staff will present the finalized draft of DG-1327 which is now Regulatory Guide 1.236. This meeting is open to the public. The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts of all open and full and subcommittee meetings, including slides presented there.

The meeting notice and agenda for this meeting are posted there. We have received no written statements or requests to make an oral statement from the public. At this point, the Committee will gather information, analyze relevant issues, facts, and formulate proposed positions and actions as appropriate for deliberation by the full committee, and that would be in June.

A transcript of the meeting is being kept and will be made available at the NRC website. This meeting is being held virtually as part of the COVID-19 preventative measures. We ask everyone except the subcommittee members and presenters to mute their microphones until the public session starts as shown in the meeting agenda.

The meeting participants should first

identify themselves and speak with clarity and volume so that they may be readily heard. We will now proceed with the meeting, and I will start by calling on Paul Clifford of NRR. Good afternoon, Paul. Are you there?

MR. CLIFFORD: Good afternoon. I am here and ready.

CHAIR KIRCHNER: Okay. Did anyone from NRR management also wish to make a statement before we begin?

MR. CLIFFORD: I do not believe that is the case.

CHAIR KIRCHNER: Okay. Thank you, Paul. Why don't you go ahead then, Paul. We look forward to your presentation. I might note that this is very timely.

MR. CLIFFORD: Well, thank you very much. Okay. So hello, everybody. Good afternoon. Welcome to virtual ACRS. My name is Paul Clifford, and I am the Senior Level Advisor for Reactor Fuel in the Office of Nuclear Reactor Regulation. I've been with the NRC in the Division of Safety System since 2003. Prior to that, I worked in the commercial nuclear industry for 16 years, starting in Combustion Engineering in Connecticut, then moving to Arizona to

work at the Palo Verde Nuclear Generating Station. And finally, I moved to Maryland and worked at Calvert Cliffs.

The focus of my 30-plus years of experience has been nuclear fuel design and performance, poor reload safety analyses, and plant operations. Today, I will be presenting Reg Guide 1.236 which provides guidance for evaluating a nuclear reactor's initial response to a postulated control rod ejection or control rod drop accident. So let's begin.

Today, we will start off with an overview of the postulated reactivity insertion accidents and applicable regulatory requirements, then a timeline of the staff's guidance and how it evolved with an expanding empirical database. I will describe stakeholder comments received by two public comment periods along with major changes to the guidance prompted by those comments.

Next, I will walk through the guidance with emphasis on what has changed since 2016 which is the last time I briefed the ACRS, then I will describe cladding hydrogen uptake models which were developed to aid in the implementation of this guidance. And finally, I will describe staff efforts to support a

relatively new industry initiative on fuel rod burnup extension.

CHAIR KIRCHNER: Paul, this is Walt. Just to show you that these virtual meetings work, I can interrupt you. May I ask very simply why you went from the previous reactivity insertion accidents title for the Reg Guide to the rather more prescriptive PWR rod ejection, BWR rod drop title?

MR. CLIFFORD: Well, I think there's always been some confusion --

CHAIR KIRCHNER: Go ahead.

MR. CLIFFORD: Okay, sorry. I think there's always been some confusion with the title and classification of these types of accidents. People referred RIAs as reactivity insertion accidents. Some referred to them as reactivity initiated accidents. I was never a big fan because there are many accidents which involve reactivity insertion, and the accident is actually not initiated from reactivity perspective. It's initiated from the failure or the movement of a particular blade or a control element.

And as I mentioned, the guidance is stylized to these particular classes of events and not all accidents involved in reactivity. So that's kind of why I moved back because we're really dealing with

how to properly analyze these accidents and to assess the impacts and to show compliance with the regulations.

CHAIR KIRCHNER: Thank you. I would submit that the material that's in the Reg Guide has broad applicability beyond just the ejection and drop accidents.

MR. CLIFFORD: Yeah, I understand.

MEMBER MARCH-LEUBA: Yeah. Paul, this is Jose. I wanted to agree again with my colleague, Walt, that almost all of this Reg Guide applies to reactivity excursion events, whether it is because of a rod or something else. And the question I have is by changing the title, are we changing the requirements a plant would have to satisfy if they have a reactivity excursion by any other means?

I realize this is a Reg Guide. It's not a rule. But are we saying if you have a reactivity excursion, say, for example, you are injecting cold unborated water into the core and producing a critical event, would you generalize that differently, that rod ejection?

MR. CLIFFORD: Yes, you would. So there are different classifications of events, I'm sure you're aware. There are reactivity events that are

classified as AOOs. There are others that are classified as postulated accidents. And each one of the accidents can have a different set of acceptance criteria. I understand that the reactor kinetics are the same. However, the acceptance criteria and the guidance and the GDCs for which you're judging compliance are different for the different accidents.

MEMBER MARCH-LEUBA: So it would be based on the frequency for the excursion event that would make it more like a severe accident in which you can have more relaxed acceptance criteria?

MR. CLIFFORD: That's correct. Generally, the lower the probability of the initiating event, the higher allowable consequences.

MEMBER MARCH-LEUBA: But the consequences are more severe. The consequences are pretty severe, even on the control rod. But okay, we'll go through this during your presentation.

MR. CLIFFORD: Right, okay. So sounds like I'm in trouble. We haven't even gotten off the title page yet. Okay. Moving on to Slide 3. The reason for concern with this type of accident, the safety significance of a reactivity insertion accident is evident from the fatal accident that occurred at the U.S. Army's prototype modular reactor, SL-1.

On January 3rd, 1961, SL-1 experienced a prompt critical power excursion felt almost instantaneously by a steam explosion as a result of an improper central control rod withdrawal. All three reactor operators were killed as a result of the physical trauma received in the violent explosion. And even if the operators had shielded themselves from the explosion, lethal amounts of radiation was released into the building. This old INL safety poster shows what remained of the SL-1 reactor. It is a good reminder.

MEMBER MARCH-LEUBA: Yes, Paul. I know you will talk over all of Slide 3. I am very glad you presented this picture because this is a visual of how important this Reg Guide is. We are trying to protect. We're making sure that the plants have the proper power-dependent and pressure limits or what their limits you -- so this doesn't happen.

Now in a different time this week, we've been talking about a different reactivity event. Is there anything special in this event that was because it was a rod? You have put up prompt criticality source of reactivity in the core. It would have behaved differently, say, for example, a cold unborated water slug.

MR. CLIFFORD: Well, certainly, yes. So the width of the pulse has a significant effect on the performance and response of the fuel through that pulse. I mean, this type of prompt reactivity occurs as a result of the quick withdrawal of a single rod element results in a very localized prompt critical excursion. And it's the height and the width of the response in a local reactor power that influences how the rod responds, whether the rod fails, and whether you melt fuel or fail cladding or in a resulting --

MEMBER MARCH-LEUBA: Is it whether you have enough reactivity? If you have 0.9 dollars of reactivity, this will happen. So you have 1.2, 1.3, this will happen. If you have 1.05, it may not happen.

MR. CLIFFORD: That is correct. The width of the prompt pulse is directly related to the amplitude. So --

MEMBER MARCH-LEUBA: So the insertion reactivity or positive reactivity, the insertion rate is important on the development. But it's not because it is a rod. It is because rods can move fast where other types of reactivity essentially may or may not move that fast.

MR. CLIFFORD: I agree. I agree.

Fundamentally, you're designing your core. And if you think about it, you have other accidents where you rely upon the actions of a lot of safety-related systems. Your reactor trips or ESFAS or ECCS or something to mitigate the consequences.

This event, this accident -- this family of accidents I should say is really developed to limit the design of the fuel, as you mentioned, how much work you can have in a single blade, how much enrichment you can put in the core. You're designing the core so that the initial response, the prompt critical response would be limited and that you'll also have inherent feedback from Doppler that'll turn the power around before you even get to the point where you're relying on engineered safety functions to act.

MEMBER MARCH-LEUBA: Yeah, which you probably don't have time to react if you're in prompt critical.

MR. CLIFFORD: Absolutely.

MEMBER MARCH-LEUBA: Okay. I just wanted to emphasize that this is a really good visual of what we're trying to achieve today. We're going to prevent this.

MEMBER BLEY: This is Dennis Bley. Since

you brought this one up, SL-1 as I recall was very highly enriched, like, over 90 percent. So we had no Doppler in this plant. It was all thermal response that shut it down. Is that true?

MR. CLIFFORD: I'm not 100 percent sure. I mean --

MEMBER MARCH-LEUBA: I am looking at the size of the core, and I agree with Dennis. This cannot be other than highly enriched core if you were to make it critical.

MR. CLIFFORD: Okay. So as this poster shows, it's important that we protect against this class of accidents. And I believe that this nuclear accident was likely in the minds of the Atomic Energy Commission when they drafted the General Design Criteria in 10 CFR Part 50 just a few years later. Let's get to the specific regulatory requirements.

Appendix A, GDC 28 limits the amount and the rate of reactivity insertion to protect the reactor coolant pressure boundary and ensure a coolable geometry. Essentially, it looks like it's written to protect against this. In addition to the reactivity requirements, there are dose requirements to the general public located in 10 CFR Part 50 and Part 100. Reg Guide 1.236 provides and acceptable

means to meet these regulations.

Reactivity insertion accidents are safety significant because of their potential ability to challenge fuel rod integrity, fuel bundle geometry, and the integrity of the reactor coolant boundary. The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion that promptly increases local power and is considered the limiting reactivity insertion accident. Of the various postulated single failures to the control rod drive system which would lead to an uncontrolled movement of a single control rod, the PWR control rod ejection and BWR and control drop are considered the most limiting scenarios for the current operating fleet.

MEMBER MARCH-LEUBA: Hey, Paul. Sorry. This is Jose again. Sorry to interrupt, but this is how real meetings go. We are always interrupting you. A year or two ago over several review cycles, we have wondered if you have to assume an additional single failure.

So if you initiate the event, it's a control rod ejection. Do you need to assume that another rod doesn't scram, if it fails to scram? And we worked this with the staff. And at the end, I'm

not sure how we resolved it. But they convinced us that no, you don't have to. What's your view on that?

MR. CLIFFORD: I think when you get into the assume single failure, you're in the GDC 27. And this Reg Guide does not address GDC 27. But from my experience working in the industry, we would always assume that instead of an N-1 scram worth, it would be an N-2 scram worth for the rod ejection event.

However, in the current plant configurations of the operating fleet, N-1, N-2 doesn't make a difference because this is not a limiting shutdown margin accident. And as I mentioned, it's initially turned around by Doppler. So the assumption of an N-1 or N-2 scram configuration does not affect this accident.

MEMBER MARCH-LEUBA: It doesn't affect the first ten seconds of the transient, but it makes an impact 36 hours behind. So basically you're telling me that this Regulatory Guide is silent on the single failure criteria and other GDCs would control that?

MR. CLIFFORD: Correct. We focus -- the stated purpose of this Reg Guide is that it provides guidance on how to evaluate the initial response of the reactor. In other words, the first five seconds and how you show compliance with those applicable

regulations in 28 and in the dose requirements which all are occurring in the first five seconds.

MEMBER MARCH-LEUBA: Yeah, our job as ACRS members is to try to look a little bit in the future or what is changing instead of just blindly applying the rule. And one thing that's changing is reactors are becoming smaller and they have less control rods. So when it's in a very large PWR, N-1 or N-2 makes no difference.

In these small reactors, N-2 is a lot minus 2. When N is only 10, that minus 2 makes a big difference, but I get your meaning. This Reg Guide does not address the issue, and this has to be address by other Reg Guides or GDCs.

MR. CLIFFORD: Yes, 100 percent.

MEMBER MARCH-LEUBA: Yeah, okay. Thank you.

MR. CLIFFORD: Okay. Now getting into the accident sequence for these postulated accidents. The control rod ejection is postulated to occur because of the mechanical failure which causes an instantaneous circumferential rupture of the control element drive mechanism housing or its associated nozzle. This results in the reactor coolant pressure ejecting the control rod and drive shaft to the fully withdrawn

position.

BWR sequence of events is a little different. A control blade is inserted into the core and becomes decoupled from the drive mechanism. The drive mechanism is subsequently withdrawn. The control blade is assumed to be stuck in place, and at a later moment, the control rod suddenly falls free and drops to the control rod drive position.

MEMBER MARCH-LEUBA: This is Jose again. Sorry to be disrupting what you were planning to say. But we have also been talking in our meetings about especially on the PWR event what we call a missile. So if the control rod housing breaks and the control rod gets ejected, I think this new Reg Guide is mostly worried or concerned about the thermodynamics inside the core.

We were concerned, does that ejected control rod has to be treated as a missile and see what it hits? We know that some reactors have missile shields for such events. I'm going to say again that your Reg Guide is silent on this issue.

MR. CLIFFORD: Yes, this Reg Guide is silent on that issue because that involves the mechanical design of the reactor vessel itself. And that responsibility resides in the Division of

Engineering, and that's a different section. I believe that's SRP Section 394 or 395, and there's guidance in that Reg Guide on the -- what's the word I'm looking for -- the integrity of the vessel and the pedigree at which the vessel is manufactured. And it involves GDC 14 compliance. So once again, this Reg Guide doesn't get into GDC 14.

MEMBER MARCH-LEUBA: Yeah, the question we were -- I'm just bringing up things we've talked about in our previous meetings over the last couple of years. And we were not very sure whether -- who's in charge of reviewing this missile or even if you have to review it as a missile or not? This guy -- what we're reviewing today is not covered by that. That should be some other topic of responsibility.

MR. CLIFFORD: Yes, it needs to reside in the Division of Engineering because they are responsible for reviewing the reactor vessel.

MEMBER MARCH-LEUBA: And changing the topic but it's related. We also talked about when you open the hole on the vessel, you have the first five seconds are the most important ones when you have the power excursion and the Doppler turned it around. But you are also creating a LOCA.

Is this supposed to be covered by the

small-break LOCA analysis that was performed somewhere else in Chapter 15? I'm going to say again you're silent on it because you can have the first five seconds, the big power pulses. Those are the important ones. But then as you -- if you have a LOCA, you have to make sure you don't uncover the core, right?

MR. CLIFFORD: Right. So I would say that this reactivity initiated accident class of events was stylized from a point of limiting fuel rod design and control element design so that you would avoid the catastrophic failure and show compliance with GDC 28. A break in the reactor vessel that would result in a slow depressurization of the RCS would then require engineered safety functions and systems to limit the consequences of that type of very slow evolving loss of coolant accident. So ultimately, your ECCS would be actuated and your ECCS would be relied upon to ensure long-term cooling (telephonic interference) and the removal of decay heat.

MEMBER MARCH-LEUBA: I understand your point of view. What I'm worried -- I've been worried for several -- I've been a member for now three years, and I worry there is some discontinuity, some compartmentalization of the review. My job is to look

at the first five seconds. Somebody else will look at the LOCA.

I would have loved to see this RG point or assess and verify that the small-break LOCA spectrum analysis covers the break that you discussed today. And just a simple -- ensure that you -- because the guys that do the small-break LOCA -- look at pipes that connect to the vessel. That's where things break. And now we're postulating that a control rod housing is breaking and it's a different size, different location. How do we know it's covered?

And what I'm worried -- and I'm not saying this happened -- is Paul is telling me, I made sure the electronics were okay. My -- temperature didn't melt. I didn't violate any of my calorie per grams limit. And the small-break LOCA is on Chapter 15 and say, well, we look at all the pipes and when they break, nothing happens, and nobody looked at the housing breaking.

MR. CLIFFORD: I agree with what you're saying, and I understand where you're going. And that really comes from the discontinuity in 50.46 because 50.46 which is a regulation on -- which governs the design of your emergency core cooling system, specifically limits breaks to those in piping and not

in penetrations in the reactor vessel.

So there could be multiple penetrations in the reactor vessel. And if you were to break those, then you would have a loss of coolant accident. But it's outside of the regulatory requirements for the ECCS system. So in other words, you're not designing your ECCS to mitigate the consequences of a vessel break. And I don't know the history of all of that because, once again, that really resides in the Division of Engineering. But I believe it's because the penetrations are all designed to GDC 14 standards. And GDC 14, to meet those standards, you have to show that there's an extremely low probability of gross failure of those vessel penetrations.

MEMBER MARCH-LEUBA: By this, you are assuming it broke because you're saying my rod ejects.

CHAIR KIRCHNER: Again, Jose, this is -- because as Paul used the operative words, this is the stylized event specifically targeted at GDC 28 and hence sets a limit on things like control rod work or reactivity in the core, et cetera, for purposes of GDC 28.

MEMBER MARCH-LEUBA: Yeah. I'm not saying I see any problem anywhere. I'm just trying to make people think if you see there are holes, and I see

some discontinuities. I'm not seeing a problem. I mean, it depends on what people think how we're doing this stuff. Please do continue.

MR. CLIFFORD: Okay. So next, we will discuss a timeline and stakeholder comments. Now onto Slide 8. This timeline shows the evolution in regulatory guidance beginning in 1974 with the release of Reg Guide 1.77 and ending with today's briefing. Note the two public comment periods for DG-1327 which was subsequently designed Reg Guide 1.236. And now we'll go on to Slide 9 and talk about each of these.

Before we begin with the requirements or the guidance, it's important to know that in-pile prompt critical power excursion type testing is required to understand the phenomena, to identify damage mechanisms and failure modes, to calibrate analytical models, and finally, to establish analytical limits to ensure acceptable fuel performance. In 1974, the year Reg Guide 1.77 was issued, the empirical database consisted of ten in-pile property activity insertion tests conducted within the capsule driver core of the Special Power Excursion Reactor Test program, referred to as SPERT-CDC. These tests were conducted in 1969 and 1970.

Reg Guide 1.77 used the empirical

database, although limited at the time, to provide acceptable analytical methods and models and assumptions for modeling a PWR control rod ejection. In addition, it provided coolable geometry limits, limits on reactor coolant system pressure, and limits on radiological consequences, and these are provided here on the slide. Two hundred and eighty calories per gram was the peak radial average fuel enthalpy allowed. Reactor system pressure was limited to Service Level C, and offsite dose consequences were limited to well within the guidance in Part 100.

Now well within translates to 25 percent. So for the maximum hypothetical accident which is of a lower frequency, the allowance would be 100 percent of 10 CFR Part 100. But for this class of accidents which is considered more likely, the radiological consequences are limited to 25 percent of the limits.

MR. CORRADINI: Paul?

MR. CLIFFORD: Yes.

MR. CORRADINI: Can I just get a clarification? I think I know what peak radial average fuel enthalpy means. That means you're looking for the peak in the radial direction, the average within the pellet. Is that correct?

MR. CLIFFORD: Correct.

MR. CORRADINI: Okay. Which means you get some sort of spatial distribution in the pellet which has to be then -- I don't want to use the word back calculated but evaluated because there might be regions which are solid and there might be regions where there's melt.

MR. CLIFFORD: Correct. But as we'll get into, the allowable melt is very limited.

MR. CORRADINI: Okay, understood. And then the second part of this is with UO2, I'm just trying to remind myself of the numbers. With UO2, 280 calories per gram average over the pellet is a small amount of melt is in existence if I do just a thermodynamic balance, right?

MR. CLIFFORD: Correct. And that portion of molten fuel would increase with burnup as melting point diminishes.

MR. CORRADINI: Correct, right. Sorry, that's right because of the change of property. Excuse me. Got it. Thank you.

MR. CLIFFORD: Okay. So moving to Slide 10, we're going to 1980. So from 1978 to 1980, a series of in-pile tests were conducted at the Power Burst Facility in Idaho. This led to additional data, and this data suggested a need for new analytical

limits.

A paper was published which recommended changes to the existing criteria. The coolability criteria was reduced from 280 calories per gram to 230 calories per gram. Cladding failure thresholds were reduced from 170 to 140 for irradiated fuel. It was noted that the failure mode strongly dependent on prior irradiation history. And based upon advanced analytical methods and evaluations, it was concluded that there was no eminent safety concern for the operating fleet. The guidance in Reg Guide 1.77 was not revised. However, many plants voluntarily implemented reduced limits.

Moving on to 2004. By 2004, the empirical database had increased significantly with in-pile testing being performed in France, Japan, and Russia. In response to reported PCMI cladding failures at lower than expected deposition energies, the NRC completed a detailed safety assessment. Based upon the research which was conducted, the expanded empirical database, and advanced analytical methods, RIL-0401 concluded that there was no eminent safety concern in the current fleet. However, different from in the past, it was decided that the guidance would be updated based on the latest and greatest information.

So building upon the expanded empirical database, the NRC issued interim guidance in 2007. There were ACRS briefings before the interim went out. There was a public comment period before the interim guidance went out. The guidance is documented in Appendix B, the Standard Review Plan 4.2.

The guidance provides hydrogen-dependent PWR and BWR PCMI cladding failure thresholds, cladding delta P-dependent failure thresholds for high temperature failure modes, burnup-dependent coolability criteria, and finally, transient fission gas release. This guidance was classified as interim because the staff was awaiting further test results to help resolve some scaling issues with the data.

Moving on to today, Slide 13 presents the current empirical database. Tests conducted since the interim guidance in 2007 are highlighted in red. This data collectively was used in the development of the final guidance within Reg Guide 1.236.

MEMBER MARCH-LEUBA: Paul, this is Jose again. Can you spend a little more time on the difference between the black dots and the open dots? What are we looking at here?

MR. CLIFFORD: I'm glad you said that. I actually had a note to say something about that, but

I missed it. Okay. So the open dots -- well, first of all, there's different symbols based on where the tests were conducted. There's a different symbol for SPERT versus PBF versus BIGR and IGR which is -- let's see what that -- those are the test reactors in Russia. Impulse Graphite Reactor, I believe that's what IGR stands for, the French reactor CABRI and the Japanese reactor NSRR.

The close symbols are those that failed. The open symbols are those that did not fail. It's important to note that not everybody reported the same information for CABRI and for NSRR. They reported the enthalpy at the time of failure. But for IGR and BIGR, they just reported the maximum enthalpy. So you don't know when it failed. You only know that it failed, whereas NSRR and CABRI, you know when it failed. So it's a little more -- it's much more valuable from that perspective.

MR. CORRADINI: Paul, can you -- I guess I wanted you to talk through the red since some of the red failures are at low values, and I guess I want to understand the new data that you emphasized.

MR. CLIFFORD: Well, the red was just emphasized as being the most current data. We'll get through how I used this data because the data is all

used differently, depending on what the failure mode is. And we'll get into that in the coming slides.

MR. CORRADINI: Thank you.

MEMBER MARCH-LEUBA: So you have -- this is Jose again. You have looked at it, I mean, all your life and understand it. Do you see any correlation with new fuels, new materials, better cladding, or it's all the same?

MR. CLIFFORD: We can get into that. Certainly, there are burnup effects and there are corrosion effects and there are also fabrication effects. And all those are considered in this Reg Guide.

MR. CORRADINI: But I think what Jose is asking is if it were M5 versus ZIRLO versus Zirc-4 versus Zirc-2, is there a difference?

MR. CLIFFORD: Yes, and there are different analytical limits for each one of those cladding types.

MEMBER BALLINGER: This is Ron. In fact, most of the Zircaloy-2 and Zircaloy-4, you don't see those in cores anymore at all, right?

MR. CLIFFORD: Right.

MEMBER BALLINGER: The last batch of Zircaloy in a core I think has probably already been

discharged. So it's either all ZIRLO or M5, right?

MR. CLIFFORD: So for the PWRs, I don't believe there's any reactors that are taking fresh batches with Zirc-4. I think that's --

MEMBER BALLINGER: Yeah, that's what I was --

MR. CLIFFORD: -- the last one.

MEMBER BALLINGER: -- thinking. So the -- rates for M5, they're much, much lower than for Zirc-4. And the hydrogen pickup rate is also much lower than for Zirc-4. Am I right?

MR. CLIFFORD: Yes, and that is specifically accounted for in this guidance.

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: And I'll get to that.

CHAIR KIRCHNER: Just a quick break here, Paul. Members, when you ask a question or make a statement, please identify yourself. That will help for people who are not on Skype but are on a public line. Thank you.

MR. CLIFFORD: Okay. As we go through the guidance, I'll try to keep that in my mind to make sure I get into the alloy specific aspects of it and the fuel specific aspects of it. If I don't, just ask me some questions. So we're moving on to Slide 14,

and this is the first public comment period in 2017.

So there was an ACRS briefing in 2016. Shortly afterwards, DG-1327 went out for public comment. Comments were received from 12 stakeholders with a total of 124 comments. Over 100 comments were accepted and prompted changes to the guidance. This pie chart below shows the distribution of comments to the various sections of the Reg Guide.

And if you quickly look at it, you see the lion's share of comments were in the analytical methods section, the range of applicability and in the failure threshold curves. However, there were comments received in really all aspects of the guidance. And as I mentioned, 100 of the comments were accepted and the NRC made changes to the guidance in response to those comments.

In 2017, DG-1327 went out for a second round of public comments. There was comments received from seven stakeholders with a total of 54 comments. Over 30 comments were accepted and prompted changes to the guidance. Here's the distribution of the comments received in 2019.

It's worth noting that there a large portion of the comments were on the radiological source term information, and those comments were

mostly directed at removing that information from this Reg Guide and putting it back into Reg Guide 1.183. There were also comments on the failure threshold curves.

It's kind of worth noting that every time we've gone out for comments, we received recommendations on how to redraw the figures, one way or another. And each time we've gone out, we have redrawn the figures. And I'll get to that later on, but I think that leads to a little regulatory instability when you start continuous -- when you continuously redraw figures based on the same data set. And I'll get to what the staff's recommendations on that is.

MR. CORRADINI: Paul, this is Corradini. One thing that you don't have to answer now but whatever is logical to put in. When you identify peak radial average enthalpy -- fuel enthalpy, is there a spatial area over which this is analyzed? That is, are we looking at a single pin, at a fuel assembly, at a group of fuel assemblies, because I think that has evolved over time given the ability to do better resolution of reactor physics. And I can't remember where that sits now.

MR. CLIFFORD: No, you're 100 percent

correct. It really depends on the granularity of the physics model that's used to calculate that. Going back in time, you would see a physics node would be approximately six inches. But since then, they've grown shorter because the computational abilities are much stronger. So you can afford more detailed resolution when you're solving your equations. So I think it depends on what's approved and when it was approved.

MR. CORRADINI: In terms of the analysis. But what I was trying to get at is -- well, I'll ask it again later when you get to loss of coolable geometry. I'm trying to understand the spatial area over which loss of coolable geometry is considered. If I had a single pin or a single two or three pins for whatever that would undergo failure, cladding failure, that's different than would be a fuel assembly. I'm just trying to understand how that's done now with the new guidance.

MR. CLIFFORD: Okay. Yeah, we'll get to it in the new guidance. But to answer your question now, it is based essentially down to the pellet. You can now see the coolability criteria down to a pellet or whatever node that the limiting node is. That node might be three or four pellets that are you're

calculating, but it's down to that one single point in the most limiting rod.

MR. CORRADINI: Thank you.

MR. CLIFFORD: Okay. So --

MR. SCHULZ: Paul, excuse me. This is Steve Schulz.

MR. CLIFFORD: Yes.

MR. SCHULZ: You mentioned a radiological source terms and fission product release fractions that the number of comments there were disposition by referring them to Reg Guide 1.183, 1.195. Does that mean that those are in process of being redeveloped, and then therefore the information that was developed here that affected those guides is going to be addressed in the future?

MR. CLIFFORD: Yes, we are actively updating those Reg Guides now. As you may now, Reg Guide 1.183 has kind of a long history. It went out for public comment in DG-1199, I believe, many years. Lots of comments were received and resolved. However, the Reg Guide was never issued final. That was one of the reasons that prompted me to take the relevant information that was going to be in that revision to that Reg Guide 1.183 and move it into Reg Guide 1.236. Knowing that, it really didn't -- it shouldn't reside

there but just a means to get it out. But the comments we received were that it shouldn't reside here if it's confusing to the industry because you're now going to have a duplication of information in two different Reg Guides and they're not identical. So --

MR. SCHULZ: Right.

MR. CLIFFORD: -- the comments were move it back where it belongs to Reg Guide 1.182, and that's what we plan on doing. We're actively updating that and we're pushing to move that forward this year.

MR. SCHULZ: This year, okay. That was -- the second part of my question was on schedule.

MR. CLIFFORD: So we have a major action involving burnup extension. So we need to update that Reg Guide, not just to bring it to the current state of knowledge but also to update it to support the higher burnups.

MR. SCHULZ: That's right. And the elements that are associated with the newer fuel designs are also affecting that Reg Guide and that effort?

MR. CLIFFORD: Well, certainly, the burnup has a direct impact on the source terms.

MR. SCHULZ: Okay. Thank you.

MR. CLIFFORD: But the source term was not

just burnup specific. We needed a new source term that would have a higher power level at extended burnups so that it would support even current fuel management techniques. Right now, people can't meet the Reg Guide 1.183 applicability limit. I believe you can't be higher than 6.3 kilowatts a foot beyond 54 gigawatt-days. They can't meet that with modern fuel management.

So we needed to expand that power burnup window, and we did that in the early revision to Table 3 which is the gap fractions. But now we're going to extend that even further to encompass the extended burnup. So it's both extended burnup and extended power.

MR. SCHULZ: Good. Thank you.

MR. CLIFFORD: Okay. So in response to two rounds of public comments, we made many changes and I'm just going to go over the major ones. And we'll get into more detail as we get into the specific guidance. But we expanded the allowable fuel burnup or the applicability range of the guidance out to 68 gigawatt-days rod average burnup. That's in Section C-1.

We improved the analytical requirements. As I mentioned, there were lots of comments to on to

provide clarity to the requirements or the organization. For instance, we lumped them all together. And as a result of comments, we tried to put the PWR-specific guidance in one section and the BWR-specific guidance in a different section just to make it clear.

We revised the PCMI cladding failure threshold curves in Section C-3. We removed the radiological source term information which was put in for the second round, now being taken out for the final. And that included the analytical requirements, the fission product gap release fractions, and an acceptable analytical procedure for calculating design specific gap fractions which has not existed. We amended the implementation section to reflect the Commission's revised backfit guidance, and we added hydrogen uptake models to aid in the implementation.

MEMBER REMPE: Paul, this is Joy Rempe, and I have a question that's more related to knowledge-management of the agency. We followed the Reg Guide. We've been given these slides which have more of a technical basis for changes, and there was a memo that was passed to us yesterday.

But in some cases when I look at the comments you received and responses back, you made

some changes. Is there some sort of Technical Basis Document that underlies this Reg Guide for knowledge-management within the agency so they can understand more about what's in the Reg Guide and the basis for it?

MR. CLIFFORD: Right. So there was a memo made 40 or 50 pages long that provided the technical and regulatory basis for what was in the Reg Guide. And that was developed to support the interim criteria. And then that was revised to then support the changes in the first -- should I say the 2017 version of DG-1327.

And then further changes to the Reg Guides are document as part of a public comment response table which is also captured in ADAMS. And then we would have then, of course, the second round of public comment responses. We would show the changes to the documents there.

So you're right. It is kind of a piecemeal -- if you went back in history, it is a little piecemeal that you would have three sources defining the technical basis for the Reg Guide. It would be the initial memo, then it would be the revisions and the public comment response table from 2017, and then it'd be revisions from the public

comment response table in 2019.

MEMBER REMPE: So is that trail at least documented somewhere internally within the agency? I know it's in ADAMS, and I just don't know all the links and things. But if a staffer wants to figure this out, to at least know how to get to the piecemeal trail.

MR. CLIFFORD: Yeah, if you -- it's probably the introduction section of the Reg Guide, one of the first pages. There's kind of a history there, and it describes the basis, exactly what I said. And all three of those documents are in ADAMS and they're referenced in the Reg Guide.

MEMBER REMPE: I didn't see, like, your memo provided us references, but perhaps I've missed that somewhere in the introduction.

MEMBER BALLINGER: This is Ron Ballinger. Talking about the memo related to the 68 gigawatt-days for time?

MR. CLIFFORD: No.

MEMBER REMPE: That came yesterday --

(Simultaneous speaking.)

MEMBER BALLINGER: That's yesterday.

MR. CLIFFORD: That's yesterday.

MEMBER BALLINGER: But in a lot of these

Reg Guides, there is a technical basis that's in one place. And might we not consider instead of forcing the reader, because this is a fairly important Reg Guide, to go and have sort of ASME codes fingered in five different locations pages and have it in one place that kind of unifies everything. It's especially good for this knowledge-management thing, but just for people's sanity when they want to go look this up.

MR. CLIFFORD: No, I mean, that's a good recommendation. That could be a good recommendation for the ACRS to update that.

MEMBER REMPE: Thank you.

MR. CLIFFORD: So we're at a natural break here. Before we move on to the next topic which is walking through the guidance, are there any more questions on the timeline or the comments we received to date?

MEMBER REMPE: I'm not sure if this is a good place or later, and I'm willing to wait till later. But I had a question about the applicability for this Reg Guide. Is this a good place to ask this and your logic for it and why you have what you have there, or should I wait?

MR. CLIFFORD: I think if we wait and hold

that question because I'm going to be walking through the different failure modes and how the different analytical limits are applied to different fields. So maybe that'll answer it naturally as we go through it.

MEMBER REMPE: Okay. So this is, like, Section 1.1 of your Reg Guide is where I -- like, you ruled out greater than five percent.

MR. CLIFFORD: And so that's the question, why we ruled out greater than five percent?

MEMBER REMPE: There's a couple of different ones like that that I'm just wondering what the logic will be. Is everything going to just be case by case in the future? You've talked about the new fuel. Well, they may go to higher enrichment. What about if you have a modular reactor with PWR fuel operating at BWR pressures? Is that something that -- are these all just going to have to be case by case?

MR. CLIFFORD: Yeah. I mean, if you read the introduction paragraph of the applicability section, it does kind of state that this is applicable to the current operating fleet for the PWRs and BWRs. And the design-specific changes to modular reactors or advanced reactors would need to be addressed on a case-by-case basis. I guess the applicant would need to come in and say why the Reg Guide continues to be

applicable to their design given whatever changes they're making to that design.

It's difficult to try to make any blanket statements on how you could apply this to a different design because I just don't understand what that design is. I mean they could have an entirely different pulse width profile or different blade widths, or there could be a lot of different aspects of it.

MEMBER BALLINGER: This is Ron Ballinger again. I mean I think the limits, the way they're posed in terms of differential pressures and burnups and hydrogen content and those kinds of things kind of lay out the field. In an SMR, as long as the SMR fits in those fields, the fact that if one is operating PWR fuel with BWR differential pressure is naturally taken care of, right?

MR. CLIFFORD: Yes, in that particular --

MEMBER BALLINGER: Anything that would related to the source term is now going to be shifted to Reg Guide 1.183, right?

MR. CLIFFORD: Correct. As we go through the presentation, if we haven't addressed it, we can circle back.

CHAIR KIRCHNER: Okay. Paul, this is --

is this a logical break point?

MR. CLIFFORD: Yeah, sure.

CHAIR KIRCHNER: Okay. Then why don't we take a -- on my NRC computer, it says it's 2:00 p.m. Eastern Time -- Daylight Time. Let's take a 10-minute break and reconvene at 2:10, 10 minutes from now.

(Whereupon, the above-entitled matter went off the record at 2:00 p.m. and resumed at 2:10 p.m.)

CHAIR KIRCHNER: Okay. Let's reconvene our meeting. This is Walt Kirchner from the ACRS, and this is a meeting of the Thermal Hydraulics and Metallurgy and Reactor Fuel Subcommittee. We are addressing Reg Guide 1.236, and Paul Clifford of the staff is the presenter. So Paul, if you're ready, please continue.

MR. CLIFFORD: Okay. Welcome back, everybody. Next, we're going to walk through the guidance with emphasis on what has changed relative to 2016 version of DG-1327, the version that was presented to the ACRS. So we're not going to get into intermittent changes between the two public comment periods. We're just going to say this was a line in the sand in 2016, and here's the final.

So we begin with the RCS pressure analytical limit. The changes to the text are in

blue. We modified the text based on comments received, noting that not all existing plants use the same pressure criterion. It's consistent with the current guidance and the practice, and it protects the reactor coolant pressure boundary, hence satisfying a portion of GDC 28.

CHAIR KIRCHNER: Now Paul, just for the record, when one does your pressure calculations, you take no credit for the fact that you actually have a LOCA, in the case of the PWR of sizable LOCA. Does it have actual failure of the control rod drive housing? So that is not credited. Is that correct?

MR. CLIFFORD: That is true. Any leakage through the postulated break in the control rod drive mechanism housing is not credited. However, it's probably worth noting in the first few seconds when you get your peak pressure even if you did credit it, it probably would not have a big impact since it's -- I forget the size. I want to say 0.04 square feet is something that pops in my mind.

CHAIR KIRCHNER: Yeah, I think that it's probably in the order of a six-inch flange.

MR. CLIFFORD: I think it's smaller than that.

CHAIR KIRCHNER: Smaller than that? The

nozzles would be smaller. I wasn't sure about the flange size.

MR. CLIFFORD: I don't know. 0.04 square feet is something that just pops in my head.

CHAIR KIRCHNER: Okay. Thank you.

MR. CLIFFORD: Okay. Moving on to the next guidance, this is damaged core coolability. So here are some photographs of fuel rod test specimens from the SPERT-CDC test program which I mentioned took place in 1969 and 1970. When we define an analytical limit for damaged core coolability, we are trying to preserve the fuel pellet stack within the fuel rod cladding within a fuel assembly bundle array, essentially a known configuration which limits fuel-coolant interaction and also preserves a geometry that's amenable to coolant.

MR. CORRADINI: This is -- Paul, this is fresh fuel, Paul? This is Corradini.

MR. CLIFFORD: So if we went back and looked at it, I believe this was all fresh fuel --

MR. CORRADINI: Thank you.

MR. CLIFFORD: -- or really low burnup, within five or ten -- probably five gigawatt-days.

MR. CORRADINI: Thank you.

MEMBER PETTI: And Paul, the energy

deposition is an average number? This is Dave. It's not the peak is it, or --

MR. CLIFFORD: I believe it was the peak deposited in the fuel.

MEMBER PETTI: Okay. Thank you.

MR. CLIFFORD: You could almost see the line, how they got to 280 at this point in time. At 240, the cladding was failed but it was still a rod structure. It still had the same geometry which could be shown to be coolable. It wasn't -- it released its fission gas, but it didn't melt and the cladding maintained its structure.

So onto the guidance itself. Here's the wording that's in the guidance document. There's only editorial changes relative to 2016. The coolability limits remain unchanged. A loss of fuel rod geometry is limited to 230, and that's based on the earlier tests, not just SPERT but also reinforced with PBF and NSRR.

The limited centerline melt which is consistent with many plants FSAR design basis. Even though it allows melt, it's limited to 10 percent volume in the centerline region, hence avoiding molten fuel-coolant interaction. It should be noted that fuel melt criteria becomes more limiting at about 30

gigawatt-days per metric ton of uranium.

In other words, you'll stop melting fuel below 230 calories per gram as you increase burnup. So the higher burnup fuel will be limited by the melt criteria. Note that these limits are designed to preserve coolability and satisfy GDC 28. And as I mentioned --

MEMBER REMPE: Paul?

MR. CLIFFORD: Yes.

MEMBER REMPE: This is Joy. I'm sorry. I thought you were done. If you have to finish a sentence, go ahead and finish it first.

MR. CLIFFORD: I was just going to remind it's just editorial changes made since 2016.

MEMBER REMPE: Okay. When I was looking through this -- and I apologize if I asked you this at the meeting whenever last time we met on it. But why is there so much focus on melting versus liquefaction, because in the severe accident world, we know that fuel and cladding materials can become liquid at lower temperatures than melting.

MR. CLIFFORD: Right. So what we're trying to do is really limit the fuel-coolant interaction. So if we're limiting it to just centerline melt, then we're not going to have that

interaction with the coolant, even if the cladding fails.

CHAIR KIRCHNER: This is Walt. Joy, I think the other thing here is the time. This is almost instantaneous whereas in the severe accident world, you're setting time and temperature and you get changes in the phase diagrams, et cetera, et cetera. But this is -- here, we're dealing with a rather short time frame, and I think --

(Simultaneous speaking.)

MEMBER REMPE: So I think there's --

CHAIR KIRCHNER: -- focusing on melt versus other phase -- parts of the phase diagram.

MEMBER REMPE: I figured timing would be one of the big ones, but I just was wondering. I don't see any sort of lower temperature phenomena occurring because of the cladding and the fuel becoming liquid. I realize your point about timing is what I suspected the answer would be. But I never noticed that in any of these tests, I guess is what you're saying.

MEMBER PETTI: Joy, this is Dave. I can remember being in Idaho when --

(Simultaneous speaking.)

MEMBER REMPE: Dave, you're going to have

to talk louder. I just cannot hear you.

MEMBER PETTI: Oh, okay. I remember talking to the guys that were doing the fuel-coolant interaction analysis on one of the PBF tests that was particularly Korea. And the severe accident loss was just starting, so they knew about that stuff at the same time. But it never came in. It must have been the time to get an interaction on a phase diagram. It was the kinetics of, frankly, it's got to take some time to occur, and it's just not occurring in the millisecond time scale here.

MEMBER REMPE: Yeah, that's what I was wondering, if they never seen it at all. Thank you.

MEMBER PETTI: An interesting question, for sure.

MEMBER REMPE: I wish you could find a way to get the volume up, Dave. I can kind of hear you. But boy, it's hard compared to other speakers.

MEMBER PETTI: I'm talking directly into cell phone. Even on the computer, I can't -- nobody can hear me through my headphones for some reason.

CHAIR KIRCHNER: For those of us in Idaho, just yell out the back window.

(Laughter.)

MEMBER REMPE: I didn't think of that.

You're right.

MR. CLIFFORD: Okay. Moving on to Slide 22, we're going to get into fuel failure estimates and radiological consequences. So to perform a proper radiological consequence assessment, first you need a conservative estimate of the number of rods that fell, and then you need an estimate of how much fission gas is released from each of the rods that fell. In the Reg Guide, we provide guidance on estimating the total number of failed fuel rods from several different failure modes.

First, there's a high temperature cladding failure threshold which will be applied to prompt critical scenarios. And there's the traditional assumption of cladding failure if you violate your DNB critical power ratio design limits which are applied to non-prompt power excursions, then there's PCMI cladding failure. And finally there's a presumption of cladding failure upon centerline melt. We'll be getting into each one of these.

The guidance for estimating or providing a conservative estimate of the amount of fission gas which is released from each failed rod includes steady-state gap inventories, releases during the transient as a result of separation of grain

boundaries, and there's also additional releases if there's fuel melt. That guidance is being moved to Reg Guide 1.183 as we talked about.

So let's get into the guts of the guidance with respect to estimating fuel rod failures. We begin with high-temperature failure modes, and I'm just showing you the guidance. We're going to get into more detail when we get to Slide 25.

So the empirical database supporting this failure mode is comprised mostly of tests conducted in Japan NSRR test facility and Russia's BIGR and IGR test facilities. Tests reported which had failures reported due to PCMI were removed from the database. Also, tests that were conducted in the CABRI sodium loop were also removed since the cladding would not have experienced any high temperature excursion. There are two failure modes which are reported. The first is brittle failure due to post-DNB oxygen-induced embrittlement, and the second is ductile failure due to post-DNB balloon and rupture.

So sensitivity studies. Here's the database plotted not as a function of burnup but as a function of cladding differential pressure. Note that the tests reported peak fuel enthalpy and not enthalpy of failure. So you're looking for the intersection

between the failed solid symbols and the non-failed hollow symbols. And this red dotted line attempts to draw that kind of a line of demarcation between the failed and the non-failed.

As I mentioned in one of the earlier slides, sometimes the reported enthalpy at failure was provided. Sometimes it was just the peak enthalpy. And these particular events, it was peak enthalpy. So you don't know when it failed. You just know that it failed. So you have to look at the intersection between the two, the failed and the non-failed test segments.

The data exhibits a clear trend with decreasing failure enthalpy with increasing cladding differential pressure. It's also worth noting that any fission gas released during the transient would contribute to rod internal pressure and hence cladding DP, and the failure trend does not continue indefinitely, below 100 calories per gram. Reported cladding temperatures remained below 800 degrees F and is no longer sensitive to failure.

MEMBER MARCH-LEUBA: Hey, Paul, this is Jose. Reinforcing what you just said a moment ago, and I know you understand this. But I wanted to see if I understand it. Let's look at the one, two,

three, four, five block squares at three megapascal. It tells me if I read it correct that one of them failed at 140, the other one failed at 160, the other one 200, 240, 300. But what you said is that most likely, they all failed at 140. It's only that the transient was so large that even though it failed at 140, it continued to go all the way to 300. Is that what you said?

MR. CLIFFORD: Yes, that's exactly what I'm saying.

MEMBER MARCH-LEUBA: Okay. Thank you.

MR. CLIFFORD: Okay. So you guys can actually see my pointer on the screen when I move it?

CHAIR KIRCHNER: Yes, we can see it.

MEMBER MARCH-LEUBA: Yes, we can see it.

MR. CLIFFORD: Okay. I guess I didn't know that. Good, okay. Now we're moving on to Slide 27. Okay. This figure shows the proposed cladding failure threshold along with the supporting empirical database. This failure threshold has remained unchanged since 2016. So at the beginning of this curve, you're protecting against brittle failure, then you move into ductile failure. And then at the end, there'd be no failures because the enthalpy is so low that you don't experience a significant transient.

MEMBER BALLINGER: Paul, this is Ron Ballinger. Quite a long time ago now, there was a study done where they did burst tests. Admittedly, I believe it was on unirradiated fuel cladding. But depending on the temperature, the differential pressure, and that's what I'm talking about here, differential pressure, the actual failure geometry of the cladding was quite different. It ranged from just a little bit of a perforation kind of thing to a case where they had a fair amount of uniform expansion, almost superplastic behavior, and that would have a big difference, a big effect on coolable geometry. Is any of that built into this?

MR. CLIFFORD: No, we're not trying to define -- like in LOCA, for instance, you have balloon and burst models which predict the extent of the balloon because you're interested in the effect of the balloon on kind of the long term thermal hydraulics.

MEMBER BALLINGER: But you are trying to factor in coolable geometry. That's one of the requirements, right?

MR. CLIFFORD: Correct. With respect to GDC 28, you're just trying to limit the rate and the amount of reactivity insertion and how that corresponds to basically violently failing the rod.

Ballooning and bursting the rod, you still preserve a bundle geometry that's still felt to be amenable to cooling. And also, this event -- this class of event is very localized. It's not going to be a core-wide phenomenon like a LOCA would be where you're worried about ballooning a lot of rods and blocking flow for a large portion of the core. These are very localized activity events.

MEMBER BALLINGER: I remember photographs and things where the balloon section was basically six inches long.

MR. CLIFFORD: And realistically, if you look at the database for ballooning, there is a tremendous amount of spread in the size and shape of the balloon. And also those events are so slow compared to this prompt critical. You don't have a long time to heat up the cladding.

MEMBER BALLINGER: Okay. Yeah, I remember that. Okay, thanks.

MR. CLIFFORD: So as I mentioned, this is the basis for the high temperature cladding failure mechanisms. Here's the analytical limit, and it hasn't changed since 2016. One important item as I mentioned is the influence of transient fission gas release.

Now during normal operation, fission gas is released. And depending on fuel design and operating history, rod internal pressures will continue to increase as a result of this fission gas release. It may even exceed reactor system pressure. During the transient, additional fission gas may be released which would also contribute to rod internal pressure. Hence, this is a contribution which must be accounted for when applying this curve. Looking at the available database, staff came up with burnup-dependent transient fission gas release correlations which are provided in the guidance to help estimate rod internal pressure. These correlations are unchanged from 2016.

MEMBER MARCH-LEUBA: It took me a while to unmute. Can you explain the figure a little bit? Can you tell us what the figure says?

MR. CLIFFORD: Okay. So the figure provides measured transient fission gas release for each of the tests as a function of the peak enthalpy increase during the event. And the symbols are color coordinated based on the burnup. So the green symbols are the lower burnup, below 30 gigawatt-days. The blue symbols are intermediate burnup, between 30 and 50. And the red symbols are greater than 50 gigawatt-

days, so higher burnup.

And as with the other curves, the open symbols did not fail. So after the test, they would've been punctured and they would have measured the fission gas in the rod. And then the closed symbols are ones that failed, and they used different means to then measure the fission gas that was released into the test capsule.

MEMBER MARCH-LEUBA: So then I understand the red solid line is trying to bound all the failed rods with the red dots which are high burnup.

MR. CLIFFORD: Correct. Right. So when looking at the data, once again, there's a lot of spread in the data as there always is spread in the data when you're talking fission gas release. So transient gas release, not surprising to see this spread. But just from looking at the data, I guess we felt that there was an observation of higher fission gas release for the higher burnup rods. So at the same energy deposition, the higher burnup rods, about 50 gigawatt-days, would release more fission gas -- more relevant fission gas release.

MEMBER MARCH-LEUBA: And the blue solid line is trying to bound all of the green and blue dots --

MR. CLIFFORD: Correct, right.

MEMBER MARCH-LEUBA: -- which are the ones below 50? And fission gas release percent, is that a percent of the theoretical calculated maximum, calculated production?

MR. CLIFFORD: Correct. So that --

MEMBER MARCH-LEUBA: What evidence?

MR. CLIFFORD: So that is a percent of fission gas that was created as a result of the burnup history. So of course, the moles of gas increases with time and with irradiation. So it's not the moles of gas. It's the percentage of the gas that's released. So that would be multiplied by the moles of gas at --

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: So if I read 25 percent FGR, I mean, 75 percent of the gas is the fission gases where it's still retaining, say, inside the crystalline structure, inside a pellet?

MR. CLIFFORD: Correct.

MEMBER BALLINGER: Though we have to be -- this is Ron again. We have to be a little bit careful when we look at this because it's really -- it doesn't say it in the figure, but it's 25 percent of the gas released in the area for which the power excursion

occurs. It's not the full rod, right?

MR. CLIFFORD: And that's the reason if you look at the text right above it is 2.3.3. We say you should calculate it along the length of the rod.

MEMBER BALLINGER: Right.

MR. CLIFFORD: So --

MEMBER BALLINGER: Along the length of the rod. Okay, okay.

(Simultaneous speaking.)

MEMBER MARCH-LEUBA: Go ahead, Paul.

MR. CLIFFORD: -- calculated it so like in the area where the rod was ejected which would be the top of the core, you might have 20 percent release. But six inches below that, you'd have 0 percent release because the change in fuel enthalpy would be --

MEMBER MARCH-LEUBA: Yeah.

MR. CLIFFORD: You wouldn't see a transient. So it's only a high percentage of those pellets that actually experience the release -- I mean the transient.

MEMBER BALLINGER: So the actually impact of the differential pressure is probably not that large.

MR. CLIFFORD: It's not that large.

However, there was a least one test where it ballooned and failed when it shouldn't have, and that's really what focused the world's attention -- the nuclear world's attention on the potential for the transient fission gas release.

MEMBER MARCH-LEUBA: Just for my education, Paul. Is this the internal of the whole cycle burnup of production of gas, or is it only the gas that is produced during the peak during excursion as the rod is ejected? This is a transient meaning during those two minutes -- two seconds where the power increased, you produce some gas. And this is the fraction of that two second production that gets released, or is it the fraction of the gas that was produced since I started up?

MR. CLIFFORD: It's the fraction of gas since you started up, but --

MR. SCHULZ: It's the latter, yeah.

MR. CLIFFORD: Yeah, it's not -- if you were using this information for calculating doses, of course you would have to take into account the decay of each of the radionuclides and stuff like that that occur and how much activity was released. But here, we're just talking moles of gas going into rod internal pressure.

MEMBER MARCH-LEUBA: So it's mostly helium, but there is some --

MR. CLIFFORD: Mm-hmm.

MEMBER MARCH-LEUBA: Okay. Thank you. Somebody else had a question. I’m over and out.

MR. SCHULZ: Paul, this is Steve Schulz. It looks this is what again is being transferred to Reg Guide 1.183?

MR. CLIFFORD: So this was maintained in this Reg Guide because it feeds into the mechanical analysis of rod internal pressure. There would also be this information along with a breakdown of how to treat this for each of the radionuclide groups when calculating activity releases as a result of transient fission gas release. But that would be slightly different.

MR. SCHULZ: Would you not expect in terms of the fission gas release calculation from the event? Just in terms of the mechanics of the calculations, it's not trivial to implement the analysis of the steady state and now the transient fission gas release creating that pressure. So wouldn't you want to make sure that you have a consistent approach in this evaluation of gas release and the one that's used in 1.183?

MR. CLIFFORD: It will be consistent. The only difference will be this would be concerned mostly just about the moles of gas released, not the activity of each, because, for instance, if you have a long-term stable nuclide, krypton-85, it's not decaying. So whether it was born on day one or it was born a second before the transient, it's still going to be there and it's going to be released and you'll have to account for the dose count.

But if it was iodine-131 which has a much shorter half-life, the iodine that was created three weeks before the accident isn't active anymore. So it doesn't really affect it. So there has to be guidance on how to treat each of the radionuclides with respect to calculating the activity that's released or the additional activity that's released during the transient. But it's still using the same database to come up with a baseline. So you would see this figure in Reg Guide 1.183, but it would be focused more on how to use this information to determine the activity released from each of the radionuclide groups.

MR. SCHULZ: Yeah, that's post-accident, if you will, post-event in terms of the radiological release.

MR. CLIFFORD: Mm-hmm.

MR. SCHULZ: Okay. Thank you.

CHAIR KIRCHNER: Paul, I think -- this is Walt. I think you described it well. This is for the structural, the pressure loading calculation, not for the source term. But they would be consistent. You would start with this and then have to look at the speciation, half-lives, et cetera, et cetera. But for purposes of the PV=nRT, this is what is added to the existing rod internal pressure.

MR. CLIFFORD: That is correct. Okay. So let's move on to calculating additional failure modes, or using additional failure modes to calculate the total number of failed rods. So we get into the requirements associated with DNB and CPR.

Based upon comments received, the application of the high temperature cladding failure curve which we just talked about was changed from all zero power scenarios to all prompt critical power scenarios to be more consistent with the empirical database. So the way this is now worded was if you're at prompt critical excursion, you use Figure 1. If you're non-prompt, then you use your DNB, CPR thermal design limits to estimate cladding failure. Any questions?

(No response.)

MR. CLIFFORD: And the use of DNB as a cladding failure metric, that's the way it's always been done. Okay. Over on to the next failure mechanism which is PCMI, Pellet-Clad Mechanical Interaction. Here's the guidance that's in the Reg Guide for estimating the number of fuel rod failures due to PCMI. We'll get into a lot more discussion. What we're trying to protect is the failure of the cladding due to the mechanical interaction of the expanding pellet. So onto the details, Slide 31.

The PCMI empirical database as a function of fuel enthalpy rise versus cladding excess hydrogen. Hydrogen which is absorbed during normal operation forms zirconium hydrides which reduce the cladding's ductility. Hydrogen uptake depends on several factors, time and temperature, power history and fluence, alloy-specific corrosion and hydrogen pickup kinetics, proximity to dissimilar metals, and RCS chemistry. Low burnup and low corrosion fuel rods retain sufficient cladding ductility and will likely fail by the high temperature mechanisms we previously discussed.

Besides overall cladding hydrogen content, PCMI database exhibits sensitivity to hydride distribution and orientation. Hydride distribution is

influenced by thermal and mechanical treatment during manufacturing and the stress state prevailing during hydride precipitation. This force micrograph shows the cross section of a high burnup PWR cladding manufactured in a stress-relieve annealed state.

This OD oxide layer is clearly visible at the top. Hydrides preferentially precipitate in a circumferential direction in the SRA state. Hydrides preferentially reside along the cooler outside surfaces as shown in the micrograph.

MEMBER BALLINGER: This is Ron Ballinger again. The PWR cladding sample would be Zircaloy-4, and the BWR cladding sample would be Zircaloy-2?

MR. CLIFFORD: In this micrograph?

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: I believe VA-2 was the Japanese cladding MDA which is --

MEMBER BALLINGER: Okay. So it's not Zircaloy-2 or 4?

MR. CLIFFORD: VA-2 on here. Now this figure here is Zirc-2.

MEMBER BALLINGER: Okay. The one on the right is Zirc-2? Okay.

MR. CLIFFORD: And all BWRs currently use Zirc-2.

MEMBER BALLINGER: Right. The one on the left is not Zircaloy-4?

MR. CLIFFORD: No, I believe it's more like a ZIRLO alloy.

MEMBER BALLINGER: Okay, okay.

MR. CLIFFORD: But it's not the alloying that affects the hydride preferential orientation distribution.

MEMBER BALLINGER: But it does affect the amount?

MR. CLIFFORD: Correct, it affects the amount of oxidation and the amount of hydrogen uptake. But it doesn't affect the morphology of the zirconium hydride which is influenced by the manufacturing.

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: So if you look to the right, the second micrograph shows a cross section from a high burnup BWR fuel which is manufactured in a fully recrystalized state. RXA cladding exhibits randomly oriented hydrides. Note that the radial hydrides provides the pathway for crack propagation. As a result, the RXA cladding is more sensitive to hydrogen content.

MEMBER BALLINGER: This is a texture effect?

MR. CLIFFORD: Correct. So if you look at the sample on the left, the failure begins on the OD in a brittle manner through the oxide and the hydride rim on the OD of the cladding and then exhibits more of a ductile failure as it moves to the lower hydrogen region towards the ID, whereas if you look at the figure on the right where you have basically this pathway of radial oriented hydrides, you see it behave in both the brittle and a ductile fashion. It begins as brittle failure with a straight line, and then you see the classical 45 degree tearing in a ductile fashion as you move away from the hydrides.

Okay. So because of the sensitivity to hydrogen, separate PCMI failure threshold lines were established which account for the impacts associated with not only the amount of hydrogen or the texture or the cladding which affects hydride precipitation but also initial RCS temperature. So you'll see four PCMI curves, one for SRA material at high temperatures, one for SRA cladding materials at low temperature, and the same is true for RXA material.

So here we're accounting for -- and also it's a function of the initial hydrogen content of the cladding at the time of the accident. So you're accounting for alloy effects and the hydrogen pickup

fraction and the corrosion rate. You're accounting for fabrication effects for SRA versus RXA, and then you're accounting for temperature effects, whether the effect begins at cold zero power conditions or hot operating conditions.

So let's get into a comparison of these four curves. This plot shows the SRA high temperature cladding failure from the DG-1327 along with a revised curve from Reg Guide 1.236 and the supporting database. In response to comments, the algebraic form of the failure correlation was changed. The new correlation exhibits a steeper decline showing a higher sensitivity to initial hydrogen content but then a saturation effect.

To provide regulatory stability, the staff elected to use a more conservative fit as opposed to the prior best estimate fit of the failure data. Best estimate curves are susceptible to constant change as new data becomes available. And as we started in 2007, as more data became available, the staff found themselves redrawing the curves, trying to reestablish a best estimate fit.

And it's noting that in 2007, the rate at which data became available was very low. CABRI had been shut down for many years. Following Fukushima,

the Japanese were conducting less tests at NSRR. But now today's environment, we're getting more data from JEA. They're back up and running. CABRI just finished their qualifications and starting to run tests. We expect multiple tests a year. And now, we spent a lot of money starting up the TREAT reactor at INL.

So we expect more data to be available as we move forward, and we wanted to design these curves and draw these curves so it was more likely that the new data would confirm the continued applicability of the curve as opposed to necessitating changes to the curve because it was found to be non-conservative. So with regulatory stability in mind, we move from a best estimate fit to more of a lower bound drawing of the curve.

MEMBER PETTI: So Paul, I have a question.

MR. CLIFFORD: Sure thing.

MEMBER PETTI: At the high hydrogen pickup numbers, there's a bunch of solid squares that are below the curve. Why would you have drawn your curve to make one be above it --

MR. CLIFFORD: Right. I mean --

(Simultaneous speaking.)

MR. CLIFFORD: Right. There's a certain

amount of engineering judgment that goes into drawing these curves. We weren't statistically trying to come up with a 95/95 only because it was such a limited data set. So I think we're looking for the overall trends with an emphasis on trying to bound most of the data but not all of the data.

There are certain points on the curve which I show that were considered to be faulted either due to there was an incident where one of the more recent tests, the water was initially water-logged which meant it had failed before the event occurred. So the results aren't really applicable. And some of the earlier CAPRI tests had a lot of spallation and hence hydride lenses on the cladding which would fail at a much lower enthalpy and aren't allowed by the current design of fuel. So we didn't intend to bound all the data. We just bounded most of the data.

MEMBER BALLINGER: This is Ron Ballinger again. On Slide -- just pick one, Slide 34 which is what you're on and pick -- I don't know. I don't know. Pick 250 ppm hydrogen. This is all Zircaloy-4 data, or is there any M5 or ZIRLO data here? And is there a difference?

MR. CLIFFORD: Yes, yes. There is a difference. M5 is RXA. So the Zirc-2 -- the database

for the RXA consists of RXA Zirc-2 for BWRs and M5 for PWRs.

MEMBER BALLINGER: Right.

MR. CLIFFORD: This database consists of mostly Zirc-4 which is SRA.

MEMBER BALLINGER: Right. And so what I'm trying to be mindful of is that there is no more Zirc-4.

MR. CLIFFORD: That is true, but we don't expect that the failure enthalpy -- once you remove the corrosion dependence of the alloy which is accounted for in the x-axis here, the excess cladding hydrogen, that it's going to be the morphology or the orientation of the hydrides that has a first-order effect. And that's dependent on the manufacturing. So whether it's SRA ZIRLO or whether it's SRA Zirc-4, both of which are represented here, it doesn't have a first-order effect. They're both SRA.

MEMBER BALLINGER: But it's multivariate. For the same hydrogen concentration in, say, ZIRLO or M5 or whatever versus Zircaloy-4, you'd have a much higher burnup for Zirc-4.

MR. CLIFFORD: And that would be accounted on how you apply these curves. That is correct. Like for instance, if this was an SRA version of a good

niobium-based alloy like optimized ZIRLO, you wouldn't have more than 200 ppm at the end of life, so --

MEMBER BALLINGER: That's why I picked 250.

MR. CLIFFORD: So you wouldn't have to fall down this curve. You wouldn't --

MEMBER BALLINGER: Right.

MR. CLIFFORD: -- come down this slope. You would be up at 150 delta calories per gram for the first two cycles.

MEMBER BALLINGER: Right.

MR. CLIFFORD: So you're considering that in how you apply the curves.

MEMBER BALLINGER: Yeah.

CHAIR KIRCHNER: Yeah, Paul. Yeah, this is Walt. Thank you for that observation because I was going to ask you to just point out what a typical excess cladding hydrogen level would be at full burnup.

MR. CLIFFORD: Okay. So --

CHAIR KIRCHNER: It's well below 500. Isn't that correct?

MR. CLIFFORD: Well not for the legacy ones. Zirc-4 could get out to 700, and ZIRLO could get out to 500 or 600. But as I mentioned, we're not

really loading ZIRLO and Zirc-4 anymore. We're using optimized ZIRLO and M5 and other advanced cladding alloys. So you're starting to see end of life at about 200 or lower, so --

MEMBER BALLINGER: And that's why -- I mean a lot of this curve is irrelevant.

CHAIR KIRCHNER: Yeah, the reason I wanted to bring it up, though, is because Dave had pointed out some of the points that aren't necessarily well captured by the red line that's drawn here out at 600 or 700 ppm. But that probably, going forward, is irrelevant.

MR. CLIFFORD: That's true. So here's the first of the four curves. This is the SRA at hot, and there's an adjustment to account for the temperature scaling. So there's a slight change in the SRA. The RXA because I mentioned the radial hydrides that can form, the random orientation of the RXA hydrides, some of which would be radial. It shows that it's much more sensitive. So you're going to see a steeper, earlier decrease in ductility with excess hydrogen. So here's the curve.

What you'll notice here is in DG-1327. We didn't have any data beyond 300 ppm, so the curve was truncated there. We got some more data and noting the

saturation effect. We just drew the curve through that last data point that we had there at 700. And this is for the hot -- so this would be for -- hot would be, for instance, M5 in a PWR and cold would be RXA Zirc-2 and in a BWR for reactor startup conditions. So those are the changes in the curves.

MR. SCHULZ: Excuse me, Paul. I've got a -- this is Steve Schultz -- a general question, and it comes from your comment that experimental data now has the opportunity to come in, that there's more work that's going to be done. Can you help with understanding of, if you will, the purpose of this experimental work? When you look at what's been done here and all the experimental data that is available so far, in a way, you kind of narrowed in on the right parameters.

You've got some data that is being well captured in a conservative way by the curves that have been drawn. What do you see as the future benefit of this new data? You indicated that you wanted to have curves that aren't going to change in spite of it. That is to say, I presume you're suggesting that a Reg Guide review in the future might be five or 10 years out depending on data that comes in. But I mean what is the value of the new data that's going to be

created --

MR. CLIFFORD: So --

MR. SCHULZ: -- that you see?

MR. CLIFFORD: -- fuel designs don't stand still. They're continuously evolving. We've seen a lot of different zirconium-based alloys that have been introduced in the last 10 years, and each one of those is going to have different oxidation kinetics and potentially a different formation of hydrides along a different preferential orientation that have to be accounted for.

But we're starting to see tests that are being done for ATF designs. Doped fuel is a big thing that's coming our way. So there's a lot of tests that'll be run on doped fuel, and there's various doping agents. There's going to be coated cladding, whether they put chromium on the outside of the cladding or some other the metallic barrier. We could see that, and there would be testing done kind of to confirm the applicability.

Higher enrichment, we expect to see licensing actions for higher enrichment and higher burnups, first to 68 gigawatt-days. That's why we went to the trouble of evaluating the database, and I have a presentation on that upcoming. But then

further to 75 gigawatt-days, so you might see more of an intrinsic or inherent burnup phenomena that occurs that causes failure at lower enthalpies, maybe due to higher gaseous swelling, due to the availability of more fission gas along grain boundaries or within the grains. So there are changes that are occurring, and there are tests that are being done to try to capture those for either demonstrating that the existing failure limits are applicable or for doing new ones.

MR. SCHULZ: That's what I want to do to get to because -- and it's stated in the Reg Guide clearly that the expectation or the desire would be for the new cladding to be demonstrated to either match or be modified to this curve as a result of data being provided.

MEMBER BALLINGER: But the newer -- and this is Ron again. The newer cladding -- any newer cladding, in order for it to be used, would push things to the left in terms of hydrogen pickup and corrosion rates and things like that. And the only thing that would push things out to the right, which would be to higher hydrogen, would be burnup. And until we raise the enrichment limit considerably, we're not going to get burnup much higher, certainly not more than 100.

MR. SCHULZ: That goes back to Joy's earlier question too which we might come to later.

MR. CLIFFORD: Right. So when we -- I think that memo was distributed maybe yesterday or the day before, staff's memo on the applicability to higher burnup for Reg Guide 1.236. In that memo, we not only assess the sensitivity with burnup and the extent of the database. But we also identified data gaps.

So we've identified several important data gaps if we want to go to higher burnup to try to confirm whether or not there's intrinsic burnup effects or fill in the gaps that are in the existing database. For example, looking at this Slide 37, not a whole lot of data here, right? So it would be -- I think we're very limited in RXA data. And since a lot of reactors are moving towards say M5 which is an RXA material, it would be good to get more information.

So maybe we move on to -- also, we're sticking with PCMI but moving into a different topic. We're getting into the impact or the potential impact of barrier lining on the application of PCMI failure thresholds. So in response to numerous fuel rod failures during normal operation in the BWR fleet in the '70s and '80s, the industry developed barrier

cladding.

The root cause of these fuel failures was PCI stress corrosion and cracking. Barrier cladding consists of a natural or low alloy zirconium liner on the cladding ID of standard Zirc-2 cladding. It has been shown to be less susceptible to PCI SCC. One unexpected benefit of the liner was its higher affinity for hydrogen.

Micrograph on the bottom right shows a cross-section of a radiated Zirc-2 barrier cladding. Zirconium hydrides are clearly visible. Note the concentration of hydrides within the barrier liner along the ID. Hence, the liner depletes the base metal of the detrimental effects of hydrogen. And also, studies have shown that the liner will remain intact even at higher concentrations of hydrides.

Next, you'll see -- here's the plot for the RXA failure threshold at hot zero power and the supporting database. The blue symbols represent test segments consisting of Zirc-2 cladding with a barrier liner. Note that these barrier tests are the most important test results since they define the shape of the curve.

So the test results shown in blue are shifted to account for the presence of up to 30

percent of the hydrogen in the barrier lining. Using the scale test results, the staff developed RXA PCMI cladding failures for both fuel rod types with and without barrier cladding. Are there any questions?

CHAIR KIRCHNER: Five-second rule, Paul.

(Laughter.)

MR. CLIFFORD: Okay. So we're moving on to a different topic. So let me note something here, and I think it's important. So by having the barrier liner, not only were you protecting against PCI stress corrosion cracking and allowing plants to maneuver more aggressively, you were also basically absorbing all the hydrogen, hence forming less hydrides in the base material. So it was giving you the double benefit.

And if plants were to move away from zirconium liner, then they wouldn't have that second benefit, and thus at the same excess hydrogen level would have a higher impact on the cladding ductility, hence the shift in the curve. Okay. We'll go to the next slide. Now we're getting into a different agenda topic which is the hydrogen models.

In support of the now-dormant 50.46(c) rulemaking, the staff developed acceptable conservative cladding hydrogen uptake models. These

models were part of the public comment period for DG-1263 in 2014 and were published in a draft Reg Guide 1.224 in 2015. These models have been part of previous ACRS briefings on the 50.46(c) rulemaking.

CHAIR KIRCHNER: You picked up on that now-dormant wording.

(Laughter.)

MR. CLIFFORD: Given the status of 50.46(c), the staff decided to include these hydrogen models in Reg Guide to help facilitate implementation of the hydrogen dependent PCMI failure curves.

MEMBER BALLINGER: This is Ron again. I keep coming back to the difference between M5 and Zircaloy-4. The lower curve is for Zircaloy-4, and the upper curve is probably for Zircaloy-2. But there's no M5 data there on the lower one.

MR. CLIFFORD: Hold on. So you're looking at -- hold on.

MEMBER BALLINGER: Over at Slide 40.

MR. CLIFFORD: Right, okay. So the top curve is the BWR cladding, so that was Zirc-2. Below that is just the empirical database for Zirc-4. So for the BWR Zirc-2 where there is a noticeable impact of fluence --

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: -- we developed kind of a fluence-dependent correlation for hydrogen pickup fraction. For the PWR alloys where we don't see that, we developed conservative bounding pickup fractions which would then be combined with an approved fuel mechanical design code which calculates oxide thickness to then translate into hydrogen content which would then --

MEMBER BALLINGER: But the M5 slope would be shallower, right?

MR. CLIFFORD: Well, correct. M5 has the benefit of having both lower corrosion, a much lower corrosion rate, but also a lower pickup fraction.

MEMBER BALLINGER: Right.

MR. CLIFFORD: So yeah, the data to the bottom right is just Zirc-4 data. And what I'm showing here is just if you use a prediction of oxide and you use a 20 percent pickup fraction, you bound a majority of the data for measured hydrogen content. So it's just one acceptable model for translating on approved fuel mechanical design into an estimate of cladding hydrogen content just to implement those failure curves.

MEMBER BALLINGER: Are we being sure that we're not compounding conservatism, especially with

M5, to negate some of the advantages of M5?

MR. CLIFFORD: Compounding conservatism?

MEMBER BALLINGER: In other words, if you got a conservative hydrogen pickup curve where M5 is considerably below that, does that, in effect, penalize M5, because now that translates back into the allowable enthalpy in an insertion accident. Am I missing something here?

MR. CLIFFORD: So it just happened if you look at the last paragraph in that C-1, it says, the hydrogen pickup fractions should be used along with a best estimate prediction of peak oxide thickness from an approved model. So --

MEMBER BALLINGER: Right.

MR. CLIFFORD: -- it's using a conservative hydrogen pickup model but with a best estimate prediction of corrosion. So I don't think it's overly conservative. It's certainly more conservative than if the industry was to measure a lot of data, come in and get an approved hydrogen model. I think it would be less conservative than this, but --

(Simultaneous speaking.)

MEMBER BALLINGER: I keep -- if I go back to Slide -- where am I -- Slide 37 or 36 where the

knee in that curve is in some cases around 150 ppm. That conservatism can shift where you are in the knee in that curve, right?

MR. CLIFFORD: Correct.

MEMBER BALLINGER: Okay.

MR. CLIFFORD: I would say, though, this is excess cladding -- excess hydrogen. So you figure hot full conditions, you may have 70 ppm in solution and the remainder in being excessed in the form of hydride. So you would need -- see the curves falls at about -- and just eyeballing it -- falls at about 80.

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: So you would need more than 150 ppm before you started to see the detrimental effects. And I don't believe for M5 you would see more than 150 ppm at end of life. So it's not being overly conservative for the advanced cladding alloys.

MEMBER BALLINGER: Okay. All right. I've got to think about it. Okay.

MR. CLIFFORD: That's the problem with animating stuff. You've got to walk through each of these changes. Okay. So once again, these hydrogen models are there to aid in the implementation of the failure curves. You've seen them before in 2014 and 2015. Anymore questions on this topic?

Okay. Hearing none, good. We're moving on to the last topic which is the burnup extension. In support of the near-term licensing actions to extend allowable fuel rod burnup out from 62 gigawatt-days per metric ton of uranium to 68 gigawatt-days per metric ton of uranium, the staff completed a critical assessment of the empirical database supporting each portion of Reg Guide 1.236.

And there's the ML number for the -- so we investigated the sensitivity of the phenomena to burnup, and then we assessed the extent of the empirical database. And here's just a graph shown earlier that just shows all of the data as a function of burnup so you can see the extent of the data beyond the current burnup limits. It's important to note and this is kind of a source of confusion. When they report data -- sorry. When they report burnup, they're reporting a burnup -- the average burnup on the segment -- the test segment.

And the test segments can be four inches or they can be -- I think the largest were close to two feet or three feet in length. So when you're looking at the reported burnup, it's closer to a nodal burnup or even closer to a pellet burnup. But it is a rod average burnup, so you've got to kind of adjust

it based on that.

So what I'm saying here is the empirical data for the test segments, if you consider that a rod average 68 gigawatt-days which is the target we're trying to achieve here is approximately 75 gigawatt-days on a test segment. So when we're looking at the extent, we're saying, well, how much data is there out to 75 on the local fuel burnup? I just wanted to introduce that.

So we'll start with the high-temperature cladding failure threshold. Here's the database as a function of burnup with failure enthalpy. Or once again, this isn't failure enthalpy. It's the enthalpy and then the closed symbols -- solid symbols show which tests failed.

As a function of burnup, you'll notice if you go across at 150 here, you don't see a sensitivity to burnup. And it was determined that the cladding failure threshold was more sensitive to differential pressure. And of course, there is a relationship between differential pressure and burnup because higher burnup has more fission gas release, more fission gas release, higher delta P. So there is somewhat of a connection between burnup and differential pressure.

And if you look at the extent of the database of these higher burnups, you see these points here. You guys can see my thing here and I'll make it real fancy. Hold on a sec. Look at this, huh? My simulated laser pointer, it shows that these points survived above 100 calories per gram which is towards the end of life here. In fact, you didn't have failures up to 150 calories per gram, even in the higher burnup segments. So it was felt that the phenomena, in fact, was not sensitive to burnup and there was data out past higher burnups to support expansion in the applicability. Now --

MEMBER BALLINGER: This is Ron again. Back up a slide. What's the uncertainty? The differential pressure is a calculated number, right?

MR. CLIFFORD: Well, it would be on how you apply it. But in this database, it was something that was designed. It was measured.

MEMBER BALLINGER: Okay.

MR. CLIFFORD: They set the initial pressure to achieve a certain target.

MEMBER BALLINGER: Right. But in real life, it would be a calculated number?

MR. CLIFFORD: Correct.

MEMBER BALLINGER: Okay, which has

uncertainty, and I'm looking at the numbers. We're talking about three or four MPA. Well, okay.

MR. CLIFFORD: And realistically, this event, because you have more reactivity in a fresher bundle, by the time you get to high burnup, the reactivity in that bundle, hence the worth of the blade next to that bundle would be a lot lower. So this event is more limiting for fresher fuel, and fresher fuel probably don't have a DP because the RCS pressure is not decreasing during the event. It's just barely changing in the first couple seconds. So you're not seeing -- it's not like a LOCA where you also have pressure is dropping. So you have to have an initial pressure beyond system pressure to even move to the right on this scale over here. So --

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: -- chances are for the rods that are more susceptible, you're going to be in this realm right here.

MEMBER BALLINGER: But at the end of, say, third cycle, the fuel rod is actually probably being driven --

MR. CLIFFORD: Correct.

MEMBER BALLINGER: -- by adjacent assemblies, right?

MR. CLIFFORD: Right. But if it resides out in the core peripheral, there's going to be no worth --

CHAIR KIRCHNER: Actually, it's running on plutonium, Ron.

MEMBER BALLINGER: It's running on plutonium, yes.

MR. CLIFFORD: So your concern is the pedigree of the calculation of rod internal pressure that gets used to implement this?

MEMBER BALLINGER: Yeah. I mean, those burnups, especially with fission gas release, the starting differential pressure is going to be very, very low, right, because you've got system pressure on one side and you've got almost liftoff on the other side, right?

MR. CLIFFORD: Yeah, right. The limit on rod internal pressure from the design -- the mechanical design calculation is that you can't lift off. So right, you could be 800 pounds per square inch above system pressure and a very high burnup rod at end of life.

MEMBER BALLINGER: Right. Okay. I just am saying that things get a little bit muddled when these calculations become tenuous.

MR. CLIFFORD: Potentially. It depends on how fine they want to do the calculations. I've always felt that if you wanted to try to predict the exact number of rods that fail by each mechanism for each core loading, and since you have a differential pressure, a dependent failure threshold, and you have a cladding hydrogen dependent failure threshold, then that could -- it could be very difficult to find the worst possible scenario because you're no longer looking for the highest-worth ejection. So you just can't do a quick physics calculation saying, what's the maximum ejected rod worth, and do an analysis there, because there may be a rod worth that's of lower magnitude but is next to a cladding that has higher hydrogen content, hence a lower failure threshold.

MEMBER BALLINGER: And those calculations are basically almost full core calculations nowadays, right, rod by rod?

MR. CLIFFORD: I don't know. I don't know off the top of my head. I mean, there's been a lot of evolution in the analytical methods over the last couple of years as we're moving towards 3D space-time kind of kinetics as opposed to the 1D 1970s versions.

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: But where I was going with it, you want to do a very, very detailed consensus evaluation where you figure out how many rods fail for every mechanism and every point during a cycle for each reload. It could be very difficult, but remember, what we're trying to do here is estimate fuel failures that goes into a dose calc. This is not very limiting from a dose calc.

Even though you're only allowed 25 percent, you could still fail. You could survive even if you failed 20 percent of the core. You could meet your dose requirements, and there's no way to get to 20 percent fuel failure because the event is very localized.

MEMBER BALLINGER: Right.

MR. CLIFFORD: So you could do a bounding assessment once and say, look, I'm going to look at these failure thresholds and show that. I'm never going to get to this point because I've done a bounding dose assessment and I just can't fail any rods and then break down to some reload checklist item. But we're just providing the level of detail should somebody want to get into the weeds.

Okay. So where was I? Okay, PCMI. Now we're moving on to the burnup dependence of PCMI.

Okay. This plot shows the SRA PCMI failure threshold along the empirical database. The data suggests that PCMI is more sensitive to and better represented by a function of cladding hydrogen content as opposed to burnup.

Here's the same data, both rise and peak enthalpy but as a function of burnup instead of excess cladding hydrogen. It's important to note that you would expect some burnup effects on cladding PCMI performance under RIA conditions. For example, increased exposure leads to a closure of the initial fuel pellet to cladding gap which would have an effect on PCI performance. And also, higher exposure leads to enhanced gaseous swelling, would affect PCMI loading.

So you would expect some inherent burnup-dependence in PCMI. However, when you start looking at the data, at least up to 68 gigawatt-days, you don't really see it. Here's another plot. Here is the empirical database with cladding excess hydrogen as a function of fuel burnup. So it gives you an idea of the extent of the database.

So at least as far as this database is concerned, as you get into higher burnup, your failure mechanism is clearly related to hydrogen and not

related to burnup. You don't have a lot of data here. But when you start looking across -- this point right here is suspect. That's SIBQ (phonetic) which was monologued.

When you start looking here at about 125 calories per gram, you don't see failures at 40 and you don't see failures at 65. So you don't see a strong burnup dependence, and you have to consider this because you could have an M5 as we talked about that doesn't have the hydrogen uptake. So it doesn't fall down that curve, all the way down to 65 calories per gram failure threshold. It stays up at 150 even at end of life.

So that's why you need to try to understand the sensitivity of the failure with burnup. And here, going across the data, there's no obvious trend in decreasing. If you just looked at the right, you would think there would be a trend like you could draw a line like this. But the only reason that these are failing is because they're at high hydrogen content which we're capturing with the failure criteria.

MEMBER BALLINGER: This is Ron again. Do we have Slide 43, your Slide 43, that one? Well, wait a minute. Okay.

MR. CLIFFORD: So the other one, they're animated, so --

MEMBER BALLINGER: Okay. All right. I'm just trying to --

MR. CLIFFORD: You start here. This is 44, and you add this additional plot, then you change one of the plots out.

MEMBER BALLINGER: Okay. I think we're a little bit -- I don't think -- maybe I have the wrong set of slides, but I don't have that sequence.

MR. CLIFFORD: If you have it, just -- if you have the PDF version, the PDF version doesn't capture the --

MEMBER BALLINGER: Oh, okay, okay.

MR. CLIFFORD: -- information.

MEMBER BALLINGER: Yeah.

MR. CLIFFORD: So back, if you're looking at the data, there's not a pronounced sensitivity with burnup, at least to this burnup level. It's certainly more sensitive to cladding hydrogen content. And if you start looking at the database, remember that the kind of saturation effect on higher hydrogen contents. So there's a kind of asymptotic limit for SRA at 65 and RXA at 50 calories per gram.

And then when you start looking at both

burnup effects and failure of really high burnup test specimens, they're not failing below those limits here. So 65, 50 is right here. So you're not failing in these data points. They failed above those limits. So based on that information, we felt it was acceptable to extend the range of applicability out to 68 gigawatt-days for the PCMI. And we look at both RXA and SRA. I'm just showing you the SRA right here.

Next part is the damaged core coolability analytical limit. And here it is right here, the 230 calories per gram and the limited amount of melt. And here's the supporting database for these limits. It's really the entire. But realistically, you're looking at this cloud of data over here at very low burnups.

We identified a need for more data out here because if -- let's walk through the logic here. So while there's no data out here at very high burnups, the criterion itself since there's two parts of the criterion and the first one being the 230. It's kind of the upper -- the ceiling. But then the next part being the limited melt which decreases with increasing burnup.

You are capturing a burnup effect using the melt criteria. But we identified this as a data gap, that we needed to have tests run on higher burnup

rods, especially if you started to go to higher enrichment, because as you go to higher enrichment, you're going to have more reactive worth at higher burnup. So in that scenario, (telephonic interference) to here so that we could verify that the ten percent melt limit would be conservative because we just don't have data in this region here.

MR. CORRADINI: Paul, this is Corradini. I don't completely follow what you just said. Are you telling me the black dots below your little red star or your red pointer are ten percent melt fuel volume limits and you're worried that with higher enrichment and higher burnup, you'd actually fall below those? Is that what you're trying to get at? I'm still not clear about what you're saying.

MR. CLIFFORD: Okay. So right now, we have the two criteria. The first one is the 230 which is really based on fresh fuel.

MR. CORRADINI: Right.

MR. CLIFFORD: And the second part of it is no melting which is going to capture burnup effects on how you predict melting, both from a radio power profile and also a melting threshold which are both burnup dependent. But what we don't have is, for instance, what if you had a loss of coolable geometry

before you melted because of some intrinsic effect and really high burnup? Right now, you would expect that fresh fuel is going to be about 260 calories per gram is when you're going to melt. And high burnup fuel is going to drop down to below 200, somewhere down here --

MR. CORRADINI: Right.

MR. CLIFFORD: -- just based on radio power profile, plutonium production, and melting temperature. So it's going to drop down to about here. But what if you had a loss of coolable geometry up here? For instance, your gaseous swelling would be enhanced at high burnup. So maybe the pellet would essentially push its way out. Not melt but it would swell so violently at the higher burnup that it dispersed into the coolant. So we don't have the data to account for that phenomena. So --

MR. CORRADINI: Okay. I think I get it.

MR. CLIFFORD: Right. These dark symbols are just when the cladding fails, not when it melts and not when you lose the integrity of the cladding.

MR. SCHULZ: Paul, this is Steve Schulz. You're talking about a combination of phenomena that may not have been incorporated or considered at lower burnup.

MR. CLIFFORD: Well, let me rephrase this. So at low burnup, you're not going to run into the melt criteria until you're up above 260. But you have an empirically based threshold at 230 based on the destruction of the rod geometry. So at 230, it didn't melt, but you still violently failed the rod so that you couldn't guarantee coolability because it was no longer one or two pieces of rod. It was multiple sections of rod.

MR. CORRADINI: Even though you didn't melt, you would have a loss of coolable geometry in that region?

MR. CLIFFORD: Correct.

MR. CORRADINI: Okay.

MR. CLIFFORD: So as you go up and burnup, the melt criteria becomes more limiting and eventually drops down to 200 calories per gram at end of life. So it's much smaller than the 230. But who's to say that the other phenomena -- not the melt phenomena but just the phenomena of the expansion of the pellet and the response of the pellet and how it failed the rod in a very complex manner. Who is to say that doesn't become more limiting at higher burnup.

MR. SCHULZ: Or in combination. That's what I was getting to.

MR. CLIFFORD: Yeah, so that's just a data gap right now. But once again, for today's fuel, the way enrichments are, by the time you get out to high burnup, you have very low worth. The blade or the rod next to that or within that high burnup assembly is very limited in worth. So you just can't get enough power into that fuel assembly. So you're not going to go above 200 calories per gram in an end of life fuel rod. I know you're worried about this, but if you were to go to eight percent enrichment, you could.

MR. CORRADINI: Okay. I see your point now.

MR. CLIFFORD: So next section was the transient fission gas release and a burnup assessment. Here's the transient fission gas release database. This is measured fission gas release versus fuel burnup just to give you a feel for the extent of the database. Here is the extent of fuel rod burnup versus peak fuel enthalpy.

So it's important to look at both, how much high burnup fuel was included in the database and how hard they pushed that fuel. So here, you can see that they pushed the high burnup fuel up above 100 calories per gram which is good. In other words, if all this data for supporting high burnup was down

here, then you really wouldn't get a good accurate measure. But since you're pushing the high burnup fuel, it gives you -- it has more pedigree to it.

So the empirical database does show a sensitivity with burnup as we described before, and also the amount of deposited energy. And that's why the staff developed these burnup dependent correlations which is shown right here. So it's important to note that these four data points that are at or above 70 are these four data points here, one, two, three, and four. So all four of the high burnup specimens remained below the correlation. So based on the extent of the database and the fact that the correlations bounds the high burnup test specimens, we felt we could extend the range of applicability out to 68.

In conclusion for the burnup extension, based on the extent and the sensitivity of important parameters and phenomena of burnup, the staff found that Reg Guide 1.236 to be applicable up to a fuel rod average burnup of 68. This assessment is predicated on the use of approved core neutronics models and fuel rod thermal-mechanical models which are validated up to at least 68 gigawatt-days. In the course of the evaluation, we also identified two limitations.

The first is as you go march down towards higher burnup, you're going to have a longer residence time. A longer residence time could mean higher corrosion, cladding corrosion levels. If you have excess cladding corrosion, you could localize defects in the cladding such as a spallation or hydride blisters.

And tests conducted on cladding with these localized defects have shown significantly reduced cladding failure threshold. So therefore, the applicability of the failure thresholds is limited to fuel rod designs and cladding alloys and plants which control and limit oxide thickness to prevent these localized effects. And right now, when we approve a cladding alloy, we approve the alloy up to a specified oxide thickness, and that thickness is based upon poolside examinations which have shown that there's no localized effects, no spallation up to a certain thickness. So we've accounted for it, but we just wanted to reinforce it here in the Reg Guide.

And the second limitation was that fuel fragmentation, relocation, and dispersal is an important phenomena. It affects the ability to demonstrate coolable geometry, and it has been shown to be sensitive to burnup. This Reg Guide does not

provide an acceptable means of addressing fuel fragmentation and dispersal at higher burnups. So are there any questions on the burnup extension portion?

Hearing none, I will move on to the final slide, 3:30. So based upon the latest research data, revised research data and new analyses and international perspectives, the NRC has developed the guidance that's within Reg Guide 1.236. It represents a significant advancement in guidance which separately captures fabrication, burnup, and corrosion effects on fuel rod performance under RIA conditions.

We have been actively involved with the industry and the ACRS beginning well before 2007. There have been several previous ACRS briefings beginning with the interim criteria. Actually, beginning with RIL-0401 and then the interim criteria and then the draft guidance and now the final guidance. In addition, there's been numerous public workshops and three rounds of public comments if you consider the standard review plan which was the interim to be the first round. That's my presentation.

CHAIR KIRCHNER: Thank you, Paul. Very good. Excellent piece of work. So before we turn to the public, Members, have you any specific questions

to ask of Paul at this juncture, and that includes consultants?

Okay. Hearing none, when can we open the public line so that we can request any comments from the public?

MS. ABDULLAHI: Is anyone here?

CHAIR KIRCHNER: Go ahead. Please state your name and then make your comment, please.

MS. ABDULLAHI: Maybe your line is muted from last time unless it's a bridge line.

CHAIR KIRCHNER: I'm sorry. I couldn't understand the comment.

MEMBER MARCH-LEUBA: That was Zena trying to figure out if the line is open.

CHAIR KIRCHNER: Oh, okay. Thank you.

MR. EICHENBERG: Hello. Can you hear me?

CHAIR KIRCHNER: Yes, go ahead.

MR. EICHENBERG: Yes, this is Tom Eichenberg, Tennessee Valley Authority, and --

CHAIR KIRCHNER: Go ahead.

MR. EICHENBERG: -- I just wanted to ask a simple question, if there is any anticipated timeline for cleanup for SRP 4.2.

(Simultaneous speaking.)

CHAIR KIRCHNER: Paul, you don't have to

answer that. Hold on, Jose. Paul, we -- Tom from TVA, we normally take comments. If you have a question about a schedule or such, direct that to Zena who is the Designated Federal Official and we'll go through proper channels and give you an answer to your question.

MR. EICHENBERG: Okay. Thank you.

CHAIR KIRCHNER: Any other members of the public wishing to make a comment?

Hearing none, I think we can close the public line, and then I would like to turn to members. Right now, our plan would be for a letter at the June Committee meeting. Are there any other comments from members about that?

MEMBER MARCH-LEUBA: Hi, this is Jose. I just wanted to say how impressed with the whole presentation. I mean, this was very thorough, very informative, and it's impressive whenever -- Paul, whenever we ask you a question, you know what we're asking before we finish talking. So I wanted to thank you for a fantastic presentation, and I think this is a very good RG and I support writing a letter in June.

MR. CLIFFORD: Thank you very much.

MEMBER BALLINGER: This is Ron. I mean, I think this along with unfortunately the, what did

you say, now dormant 10 CFR 46(c) have taken so long that with the evolution of cladding, much of this stuff -- much of the information, while extremely valuable, is moot.

MR. CORRADINI: This is Corradini. I guess I do want to ask Paul something besides, as usual, telling him he did an excellent job of trying to explain to us when we were confused. Paul, if we go back to the slide -- now I'm going to get it wrong -- Slide 40 where you basically point in that the hydrogen uptake models now are included consistent with 50.46(c). Was that required, or does that just make it easier for the applicants to use it and clear to have it as part of this RG?

MR. CLIFFORD: It's not required, no. We're just providing it to aid in the implementation because not all vendors have approved hydrogen uptake models for each one of alloys, so no.

MR. CORRADINI: So you're basically making it consistent is the way I understood.

MR. CLIFFORD: Right. We're just giving them the option of using something that we've provided and that we already find acceptable.

MR. CORRADINI: Okay, fine. But other than that, as usual, I thought it was an excellent

presentation and you helped us a lot when we were confused -- when I was confused.

MEMBER REMPE: Paul, this is Joy, and I also find your presentations, as usual, very enlightening. But that's what motivated my question about knowledge management because I think you have -- your background did reference this earlier memo. But the later memo, I still could not find referenced in the Reg Guide. I also think that as you went through the presentation, you provided us some background that I'm not sure is easily found by other staff members in the future as they start trying to accommodate new fuels. And I would hope that our letter would mention the need for something that's a little more comprehensive that's a background document to support this Reg Guide.

(Simultaneous speaking.)

MEMBER PETTI: This is Dave. I have a question for Paul. I think about all of this, and I think about accident-tolerant fuels. And you look at how much time and effort it took to get this database. And even today, some of it has some holes in it. Have you given any thought to what you think it's going to take to get enough data from an accident-tolerant fuel perspective, whether it'd be the cladding or the new

fuels?

What I worry about is it could be a two-part answer. It could be, well, we just need some data to confirm. That leads you down one path. But what if something else shows up? When you make a change over here and you find out it causes a deleterious effect over there that you didn't think of, that could really take a lot of effort. Have you given any thought as you following the accident-tolerant fuel program from this perspective?

MR. CLIFFORD: Okay. So I think accident-tolerant fuel is a good example because there's both cladding -- design changes in the cladding and design changes in the pellet itself. And of course, the further you migrate away from the empirical database, the more difficult it will be to license. I mean, something like silicon carbide which does not possess significant ductility relative to a new metal, that's going to be a challenge and it's going to likely fail at a lower threshold.

So that's something that would have to be evaluated. But given -- if it's paired with the existing UO2 design, you understand the behavior of the UO2 for a given deposit of energy. So maybe there's separate effects that could be used to help

speed up the licensing process and limit how many in-pile tests that have to be done on an irradiated cladding material that's not Zircaloy.

But once you get to changes in the fuel pellet where you just don't understand the response of the fuel pellet to a certain energy deposition, uranium silicide and uranium nitride, there's going to have to be a lot of RAI tests done, in-pile testing done at TREAT or CABRI or NSRR to establish the failure modes for that fuel design because you've got significantly different compositions of the pellet, both from an alloy perspective and the microstructure of the pellet and the grain sizes.

And the melting temperature of the pellet could be significantly different. Thermal conductivity of the pellet is different. Swelling rates are different. So if you start changing from the UO2 pellet to something else, there's a lot of work that has to be done.

CHAIR KIRCHNER: Along those lines, Paul, there are a number of concepts in terms of changing the cladding or coatings. And so I was thinking back in your presentation the positive effect of the liner for the BWR fuel cladding. Has there been any preliminary look at what some of these coatings and

such might to in terms of changing either ductile or brittle fracture properties for the cladding under these kind of accident conditions?

MR. CLIFFORD: So for the coatings, well, certainly, the coating is going to ensure that there's almost no hydrogen present on the cladding at the beginning of the transient because the oxidation rates of chromium are so low compared to zirconium that you would not even begin to fall down that PCMI failure curve. You would always be at the very top which gives you tremendous benefit.

With respect to the actual performance, the strain capability of the cladding with the coating, we don't expect there to be much because the coatings are generally two to three percent thickness of the underlying substrate. So it's not going to have a significant impact on the mechanical performance of the bare cladding if you want to think of it as bare.

CHAIR KIRCHNER: Whether it's brittle or ductile. Okay.

MEMBER BALLINGER: This is Ron. The argument that is being made was made for putting these things in lead test assemblies or the like that are already in core.

CHAIR KIRCHNER: Yes.

MR. CLIFFORD: Yes, so I don't -- overall, I don't think coated cladding is a significant challenge from a licensing perspective. I think the challenges are going to be something that's a nonmetallic cladding or a change, ceramic pellet to a metallic pellet or a metallic bore or something else.

CHAIR KIRCHNER: Yes.

MR. SCHULZ: Paul, this comment follows Ron Ballinger's comment and your introduction today and that is that, as you stated, some of us have been involved with this issue for either near or more than three decades. And so as a result, a lot of the information and data that has gone into this technology is from cladding materials which are no longer used.

At the same time, I did want to comment that the NRC team has done a very excellent job in the handling and disposition and analysis of the public comments that were put in. And as you stated, there were many. But the thoroughness by which you address those comments, including some very specific evaluations that have been done of different and new types of cladding, are very helpful in both the issue of moving forward with how this can be applied to

current design, how it can be applied to future design, and also did contribute to an element of knowledge management and technical capture.

But I do agree with Joy that it would be good to somehow figure out or make sure that the information that's been developed from the responses to those questions is captured in the archives of the NRC. I did try to access one or two -- one of the referenced documents from the questions, and I couldn't get it in ADAMS. It said it wasn't available. So that's just an example. It's not a complaint, but it's an example that we need to make sure that the information is held for future use.

CHAIR KIRCHNER: Okay. Members, any further questions or comments at this point?

(No response.)

CHAIR KIRCHNER: I'm using a ten-second rule because often we have trouble getting our cursor back on the unmute the microphone.

(Laughter.)

CHAIR KIRCHNER: Okay. That's ten seconds. Well, then at this point, I want to thank you again, Paul, and thank all the members for participating. I ask Steve and Mike, our consultants, if you have any specific comments, would you please

get those to me within the next couple of weeks. And also, Members, I will be putting together a draft letter again for the June meeting. So with that, I think we are complete with our work today. Thank you all, also in the public attending. And with that, we are adjourned.

(Whereupon, the above-entitled matter went off the record at 3:45 p.m.)

May 5, 2020 Subcommittee Meeting

ATTENDEES

ACRS SUBCOMMITTEE MEMBERS

Walter Kirchner (Chairman)

Vesna Dimitrijevic

Ronald Ballinger

Dennis Bley

Jose March-Leuba

Joy Rempe

David Petti

ACRS Consultants

Michael Corradini

Stephen Schultz

ACRS Staff

Zena Abdullahi (Designated Federal Official)

Scott Moore (ACRS Director)

Lawrence Burkhart (Technical Branch Chief)

Christina Lui

Alesha Ballinger (Branch Chief, PMDA)

Thomas Dashiell

Makeeka Compton

Paula Dorm

Joanne Johnson

Shandeth Montgomery

Hossein Nourbakhsh

Janet Riner

Tammy Skov

Derek Widmayer

Weidong Wang

Quynh Nguyen

Christopher Brown

NRC staff

Paul Clifford (presenter)

Edward O'Donnell (RG 1.236 Senior Staff)

Donald Agama

Andrew Bielen

Andrew Proffitt

Ngola Otto

External Attendees

Alex (Guest)

Gregory Broadbent

Stephen Geier

Kent Halac (GEH)

Charles Heck (GEH)

Nathanael Hudson

Shawn Lamb (GEH)

Christan McElory (GEH)

David Mitchell

Brian Mount

Kurshad Mufruoglu (Guest)

Nathan Peck (GEH)

Scott Pfeffer (GEH)

Eric S. Scott

Eric Thomas

Ken Yueh

Telephone Attendees

Electric Power (Guest)

GE Nuclear Energy (Guest)

RNT G

Wireless Caller (Guest)

+467XXXXXXXXX (Guest)

201XXXXXXXXX (Guest)

571XXXXXXXXXX (Guest)

910XXXXXXXXXXX*Guest)

Draft Regulatory Guide 1.236
PWR Control Rod Ejection and BWR Control Rod Drop Accidents

ACRS Metallurgy and Reactor Fuels Subcommittee
May 5, 2020

Paul M. Clifford
Division of Safety Systems
Nuclear Reactor Regulation

Agenda

1. Regulatory Requirements
2. Timeline and Stakeholder Comments
3. Revised Guidance and Analytical Limits
   a. Reactor Coolant System Pressure
   b. Damaged Core Coolability
   c. Radiological Consequences
4. Cladding Hydrogen Models
5. BU Extension

Reason for Concern

- Fatal accident at Army's prototype modular reactor – Stationary Low Power Reactor (SL-1)
- Improper withdrawal of central control rod resulted in prompt critical power excursion and steam explosion

Protection Against Violent Explosion and Loss of Pressure Boundary

Idaho National Engineering and Environmental Laboratory, INEEL 81-3966

Regulatory Requirements

- 10 CFR 50, Appendix A GDC 28 requires reactivity control systems to be designed with appropriate **limits** on potential **amount** and **rate** of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core.
- 10 CFR 100.11 and 10 CFR 50.67 establish **radiation dose limits** for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone.

RG 1.236 provides an acceptable means to meet requirements

Reactivity Insertion Accidents

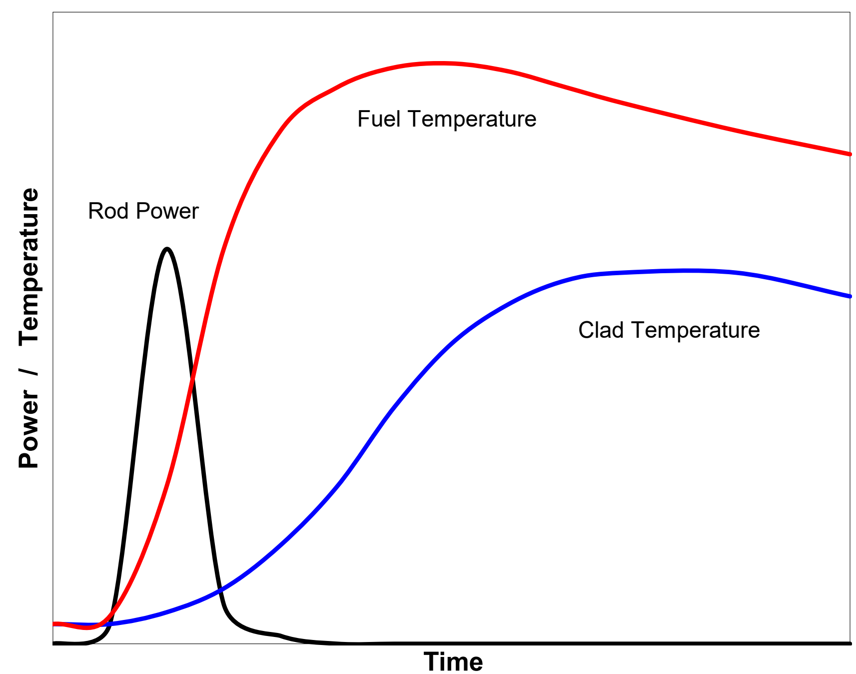
- Reactivity insertion accidents are safety significant because of their potential ability to challenge fuel rod integrity, fuel bundle geometry, and the integrity of the reactor pressure boundary

- The uncontrolled movement of a single control rod out of the core results in a positive reactivity insertion that promptly increases local core power
  - Considered the limiting reactivity insertion accident

- Of the various postulated single failures of the CRD system which may initiate an uncontrolled movement of a single control rod, PWR CRE and BWR CRD are considered the most limiting scenarios for the current operating fleet

Postulated Accidents

- A PWR CRE event is postulated to occur because of a mechanical failure that causes an instantaneous circumferential rupture of the control element drive mechanism housing or its associated nozzle. This results in the reactor coolant system pressure ejecting the control rod and drive shaft to the fully withdrawn position.
- A BWR CRD event is postulated to occur because of the following sequence of events: a control rod (blade) inserted into the core becomes decoupled from its drive mechanism, the drive mechanism is subsequently withdrawn, the control blade is assumed to be stuck in place, and at a later moment, the control rod suddenly falls free and drops to the control rod drive position.



Agenda

1. Regulatory Requirements
2. Timeline and Stakeholder Comments
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   a. Reactor Coolant System Pressure
   b. Damaged Core Coolability
   c. Radiological Consequences
4. Cladding Hydrogen Models
5. BU Extension

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Timeline

1974 RG 1.77, Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors

1980 *Nuclear Safety* article (MacDonald et.al.) suggests need for new analytical limits for coolable geometry and failure threshold

2004 RIL-0401, An Assessment of Postulated Reactivity-Initiated Accidents (RIAs) for Operating Reactors in the U.S.

2007 SRP 4.2, Appendix B, Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents

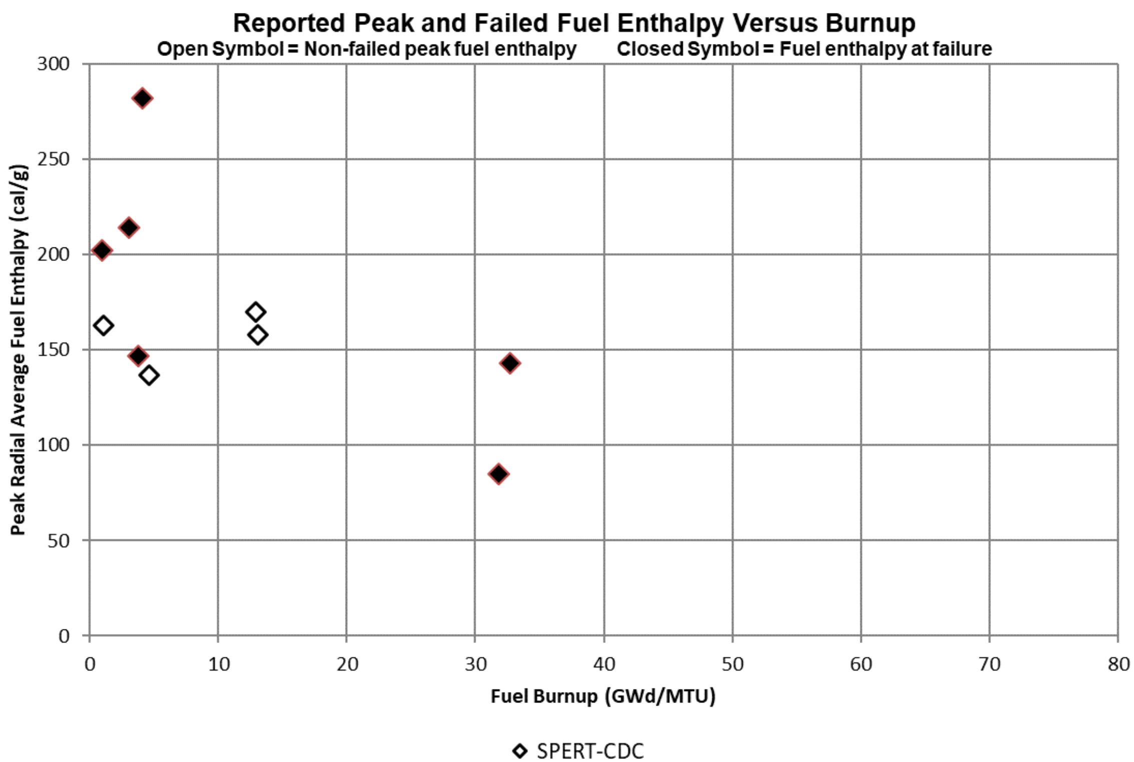
2017 DG-1327 1st public comment period

2019 DG-1327 2nd public comment period

Timeline - 1974

1974 RG 1.77, Assumptions used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors

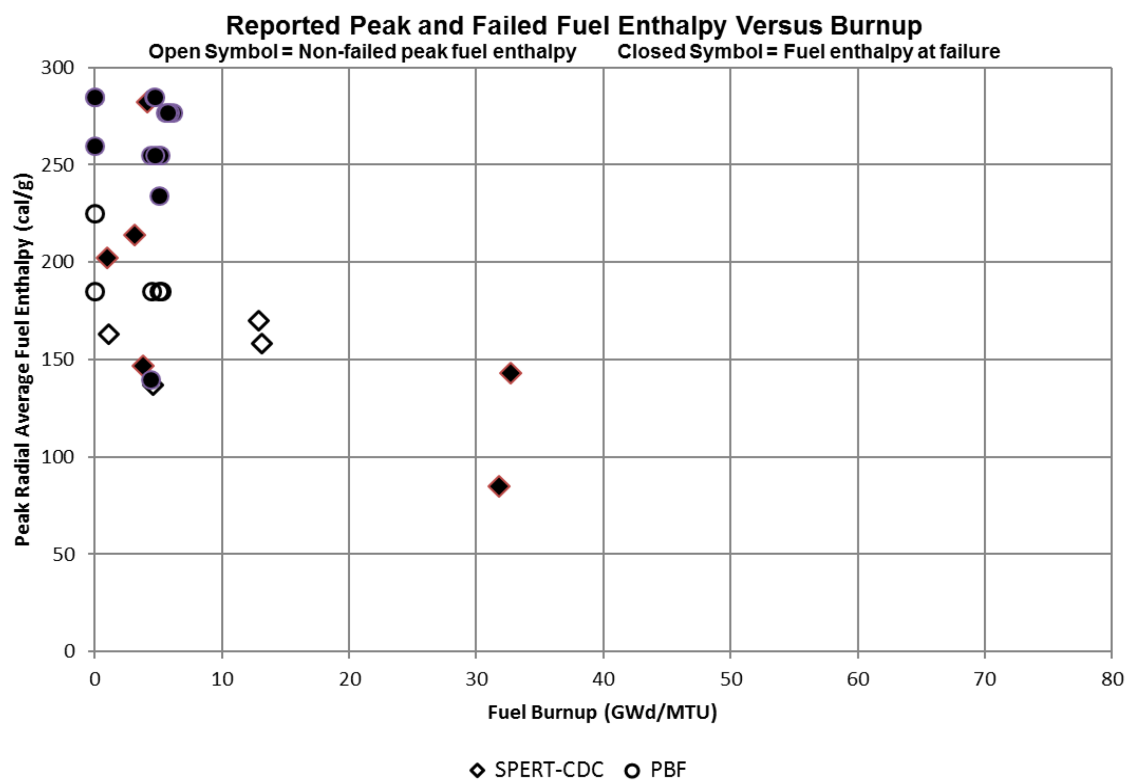
- Acceptable PWR analytical methods and assumptions
- Fuel radial average energy density limited to 280 cal/g at any axial node
- Maximum reactor pressure limited to ASME B&PV Service Level C
- Offsite dose consequences limited to "well within" the guidelines in 10CFR Part 100



Timeline - 1980

1980 *Nuclear Safety* article (MacDonald et.al.) suggests need for new analytical limits for coolable geometry and failure threshold

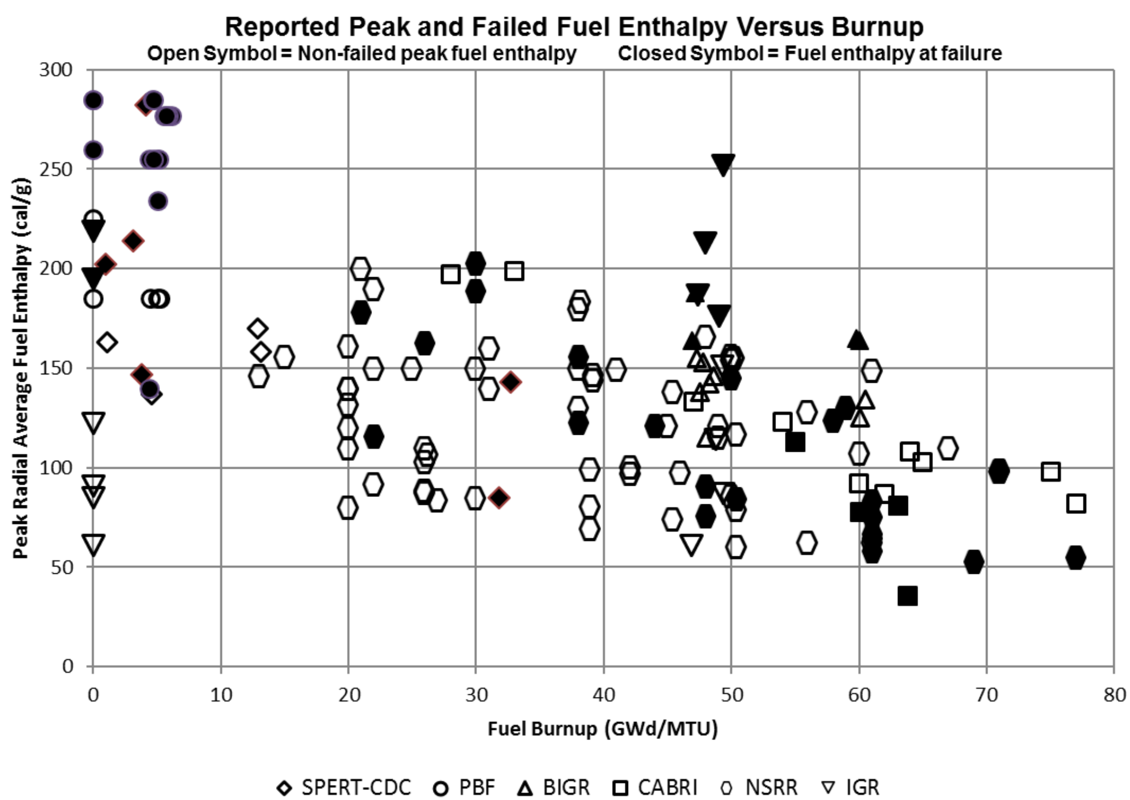
- Coolability criteria reduced to 230 cal/g at any axial node
- Cladding failure threshold reduced from 170 cal/g to 140 cal/g for irradiated fuel rods
- Failure mode strongly dependent on prior irradiation history
- Based on advanced analytics, no imminent safety concern



Timeline - 2004

2004 RIL-0401, An Assessment of Postulated Reactivity-Initiated Accidents (RIAs) for Operating Reactors in the U.S.

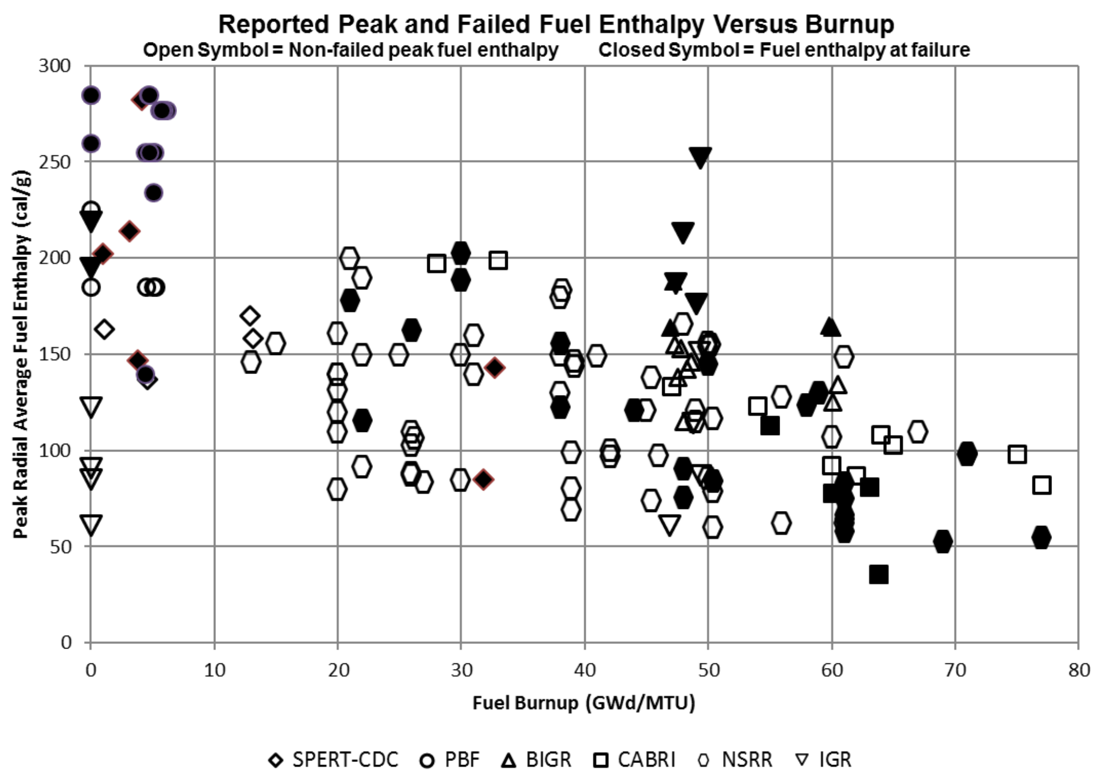
- PCMI cladding failure at much lower fuel enthalpy
- Based on advanced analytics, no imminent safety concern



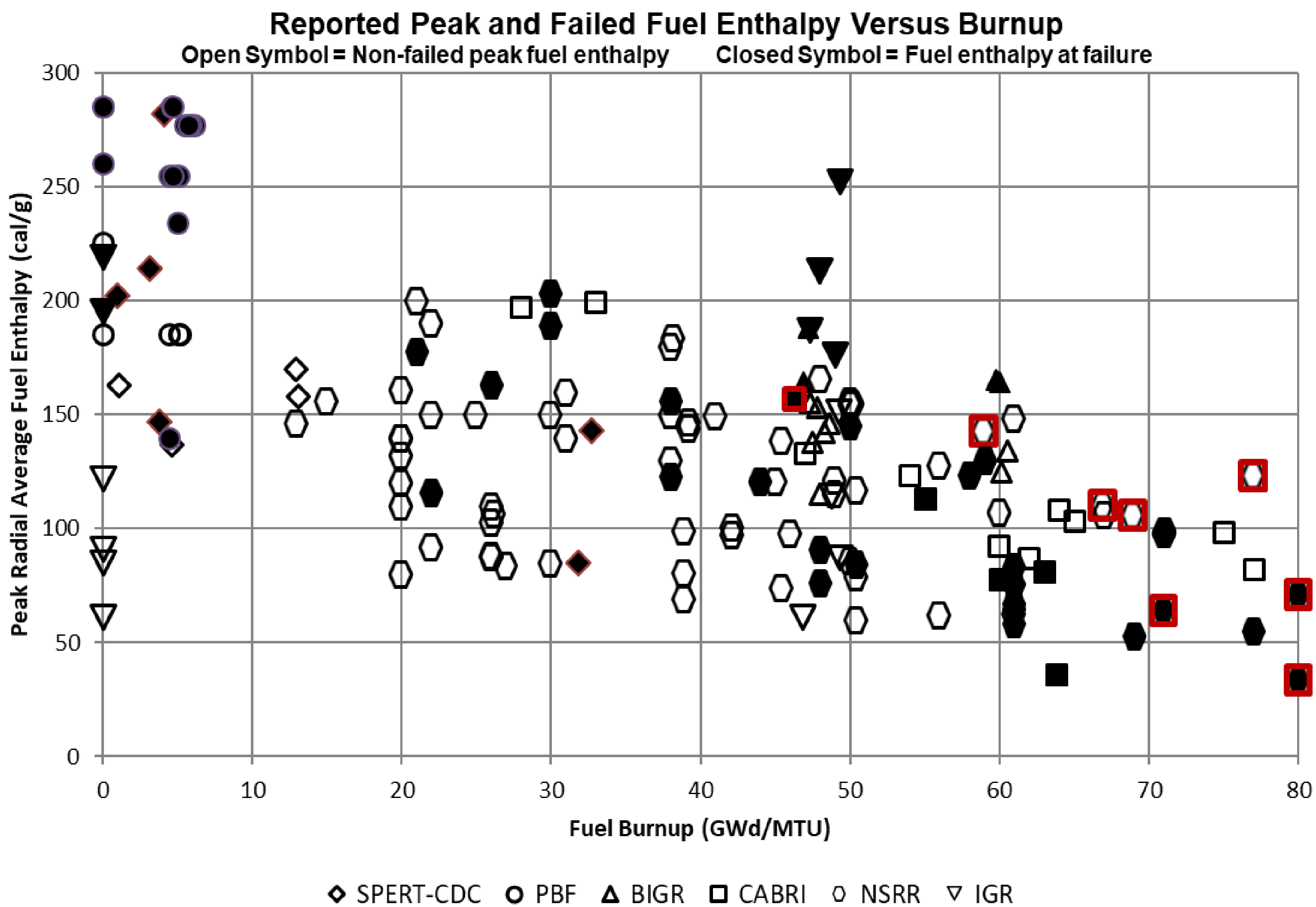
Timeline - 2007

2007 SRP 4.2, Appendix B, Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents

- Hydrogen-dependent PWR and BWR PCMI cladding failure thresholds
- Cladding ∆P-dependent cladding failure thresholds
- BU-dependent coolability criteria
- Transient FGR

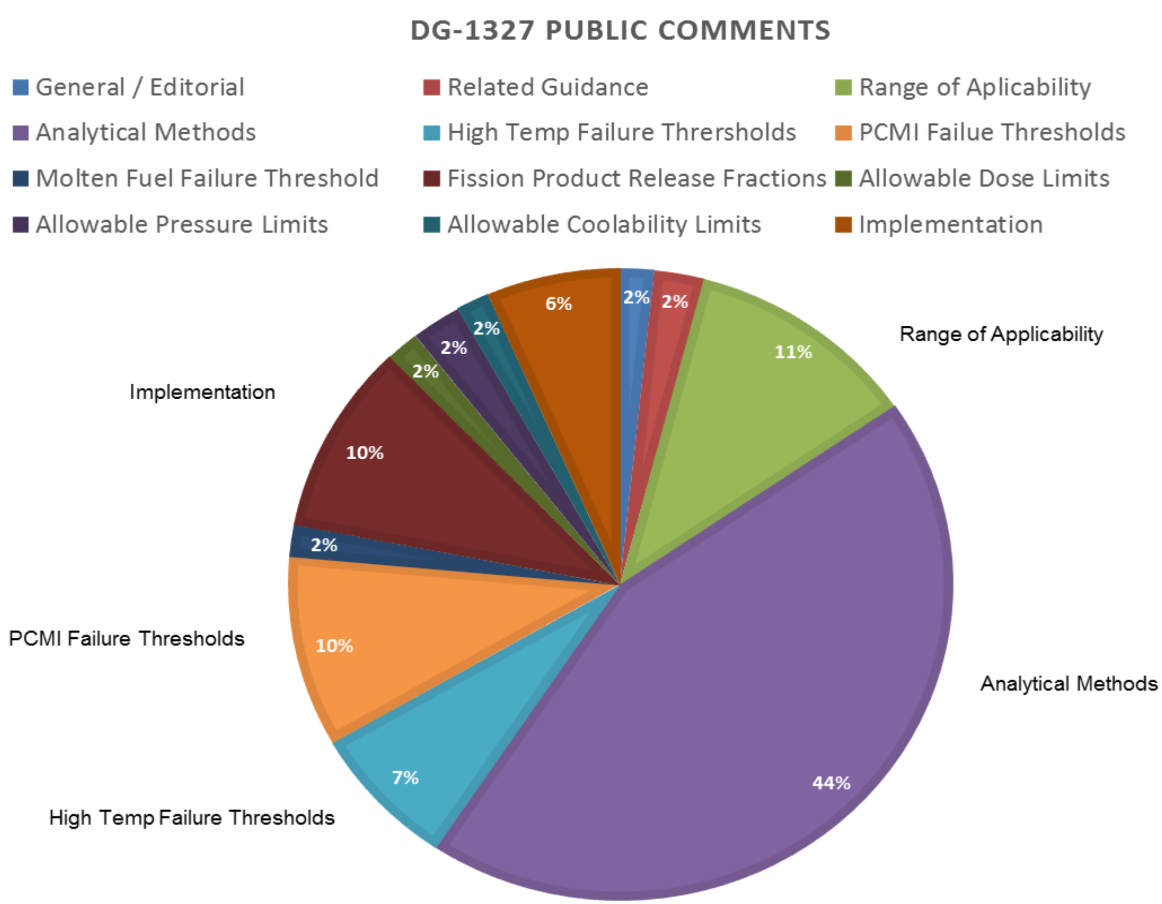


Empirical Database Circa 2019



Public Comment - 2017

- **Comment submissions received from 12 stakeholders with a total 124 comments**
  - **Over 100 comments were accepted**



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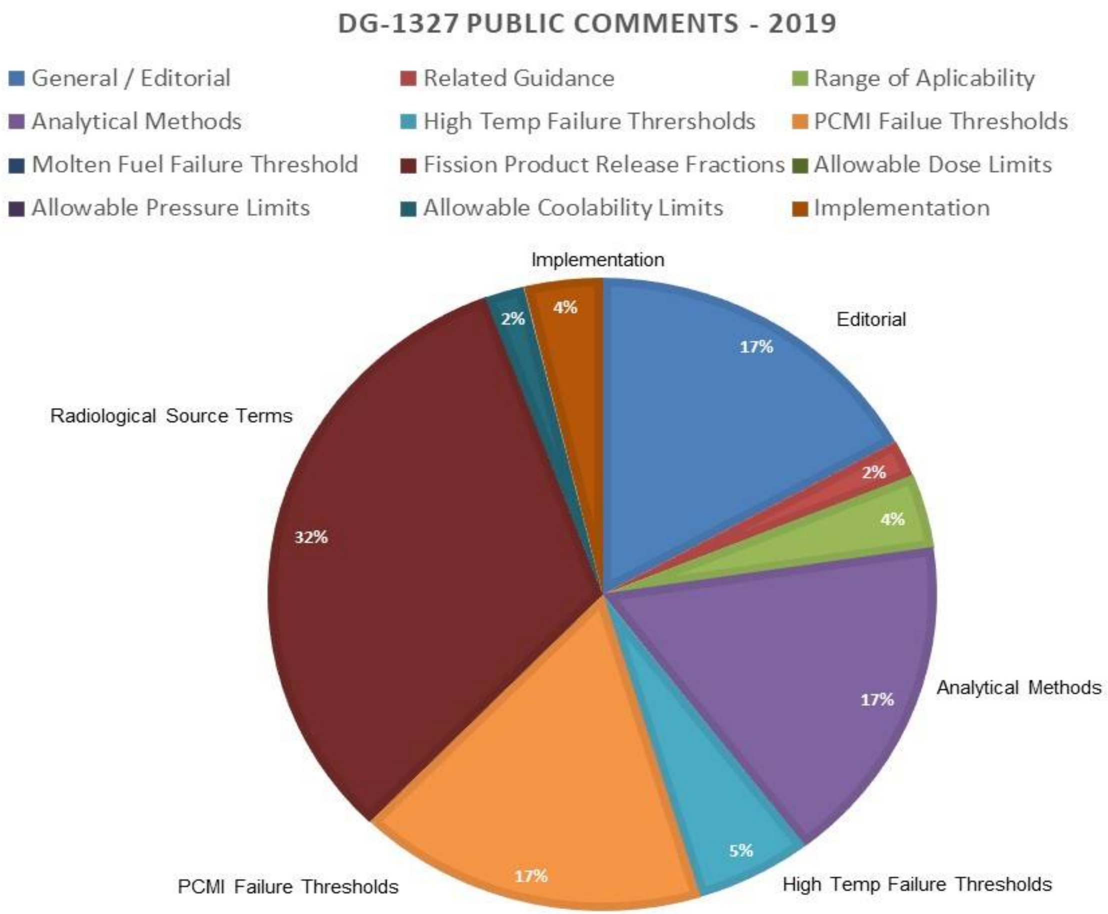
Protecting People and the Environment

Public Comment - 2019

- **Comment submissions received from 7 stakeholders with a total 54 comments**
  - **Over 30 comments were accepted**



Major Changes

- Expanded fuel burnup range to 68 GWd/MTU (Section C.1)
- Improved analytical requirements (Section C.2)
- Revised PCMI cladding failure threshold curves (Section C.3)
- Removed radiological source term information (Section C.4)
  - Analytical requirements
  - Fission product gap release fractions → Future revision to RG 1.183
  - Analytical procedure
- Amended implementation to reflect revised Backfit guidance (Section D)
- Added cladding hydrogen uptake models (Appendix C)

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1. Regulatory Requirements
2. Timeline and Stakeholder Comments
3. Revised Guidance and Analytical Limits
    a. Reactor Coolant System Pressure
    b. Damaged Core Coolability
    c. Radiological Consequences
4. Cladding Hydrogen Models
5. BU Extension

RCS Pressure Analytical Limit

5. Allowable Limits on Reactor Coolant System Pressure

For new license applications, the maximum reactor coolant system pressure should be limited to the value that will prevent stresses from exceeding Emergency Condition (Service Level C), as defined in Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 14). For existing plants, the allowable limits for the reactor pressure boundary specified in the plant's updated final safety analysis report should be maintained.

- Consistent with current guidance and practice
- Protects reactor pressure boundary - satisfies GDC-28

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Protect Against Loss of Fuel Bundle Geometry

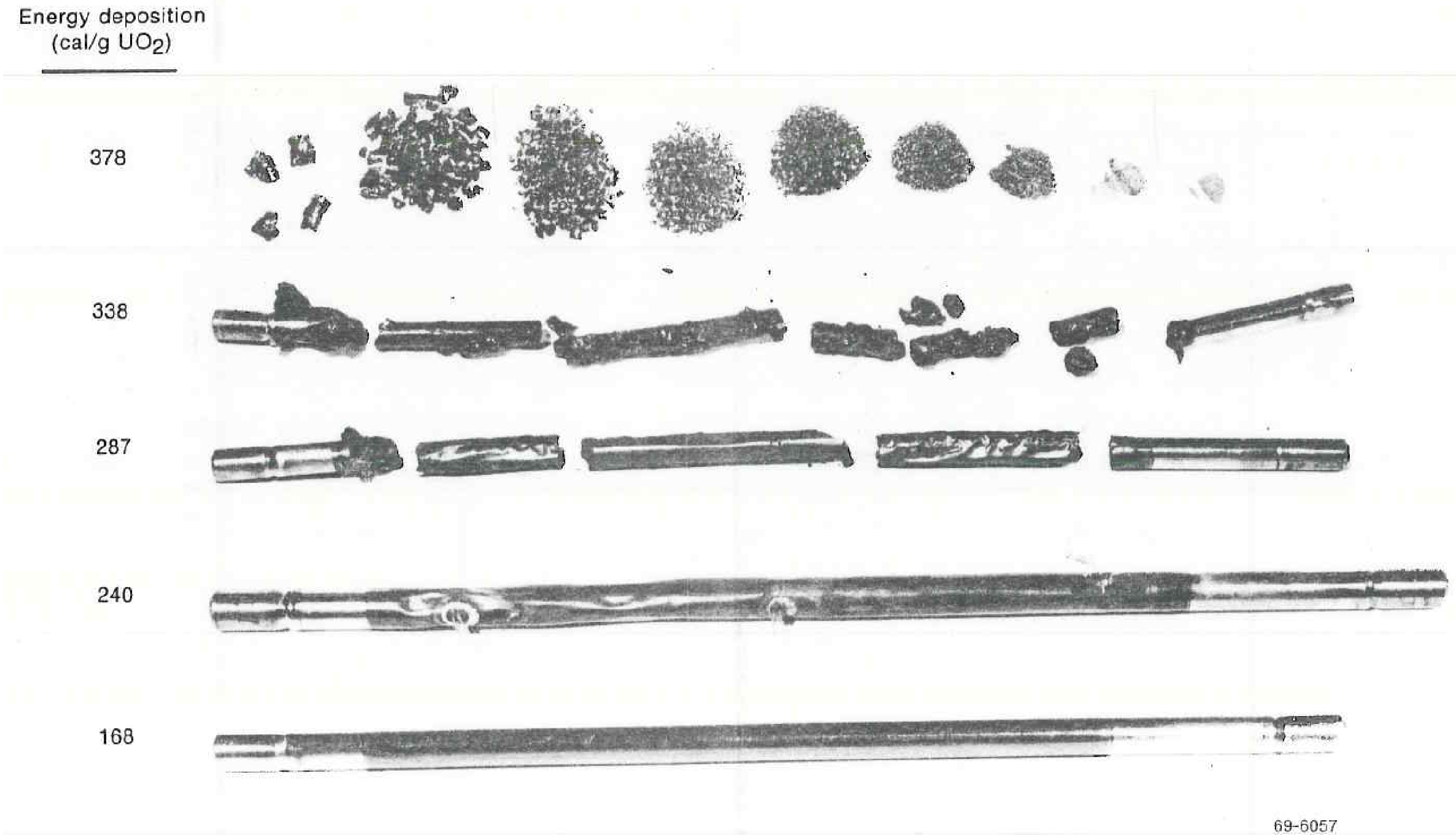

Fig. 3 Posttest photographs of SPXM rods tested in the CDC.

Coolability Analytical Limits

6. Allowable Limits on Damaged Core Coolability

Limiting peak radial average fuel enthalpy to prevent catastrophic fuel rod failure and avoiding molten fuel-coolant interaction 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.

- Loss of fuel rod geometry limit (230 cal/g) based on earlier SPERT, PBF and NSRR prompt critical experiments
- Limited centerline melt (current license bases) avoids molten FCI
- Fuel melt becomes more limiting at ~30 GWd/MTU
- Preserves coolable geometry - satisfies GDC-28

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Radiological Consequences

1. Requires a conservative estimate of the total number of failed fuel pins from all failure modes
   - Prompt critical high temperature cladding failure
   - Non-prompt DNB/CPR cladding failure
   - PCMI cladding failure
   - Centerline fuel melt cladding failure

2. Requires a conservative estimate of fission product release fractions
   - Steady-state gap inventories
   - Transient fission gas release → Future revision to RG 1.183
   - Fuel melt fission gas release

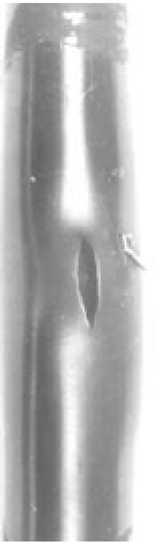
HT Cladding Failure Threshold

3.1 High-Temperature Cladding Failure Threshold

Figure 1 shows the empirically based high-temperature cladding failure threshold. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events with prompt critical excursions (i.e. ejected rod worth or drop rod worth greater than or equal to $1.0). Because ductile failure depends on cladding temperature and differential pressure (i.e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (megapascals (MPa)).

For prompt critical scenarios which experience a prolonged power level following the prompt pulse, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

For non-prompt critical excursions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits.



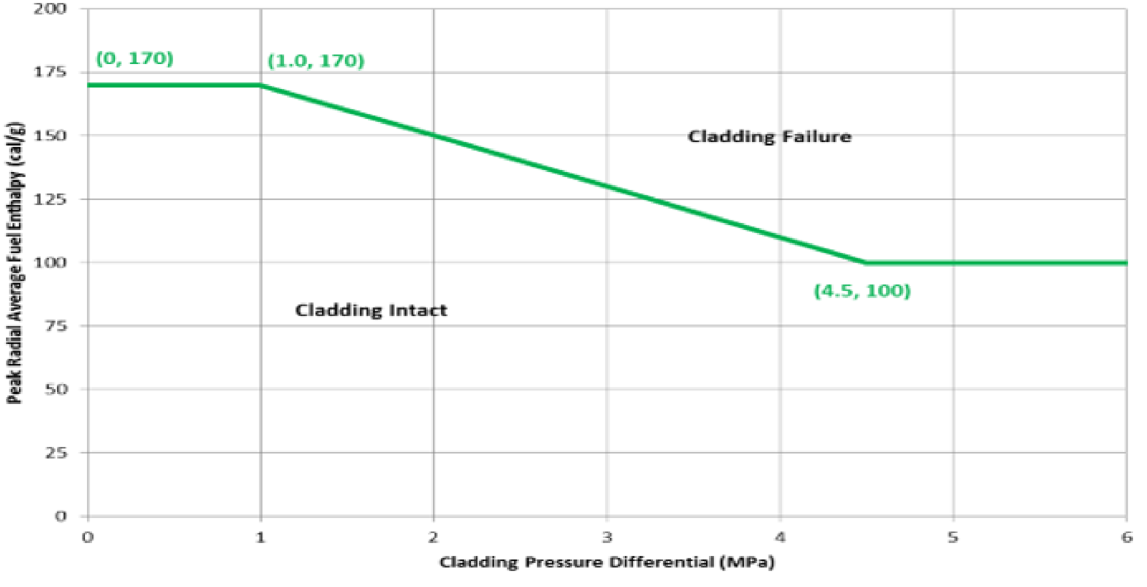
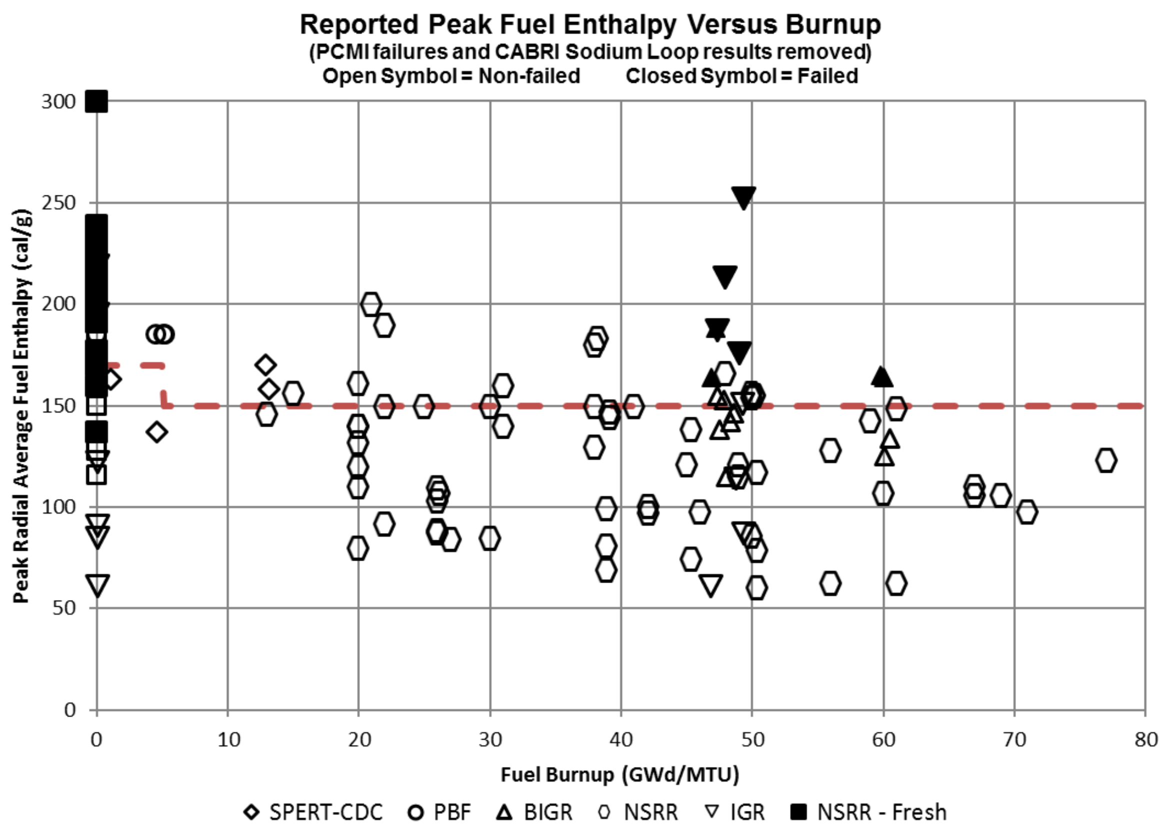


Figure 1. High-Temperature Cladding Failure Threshold

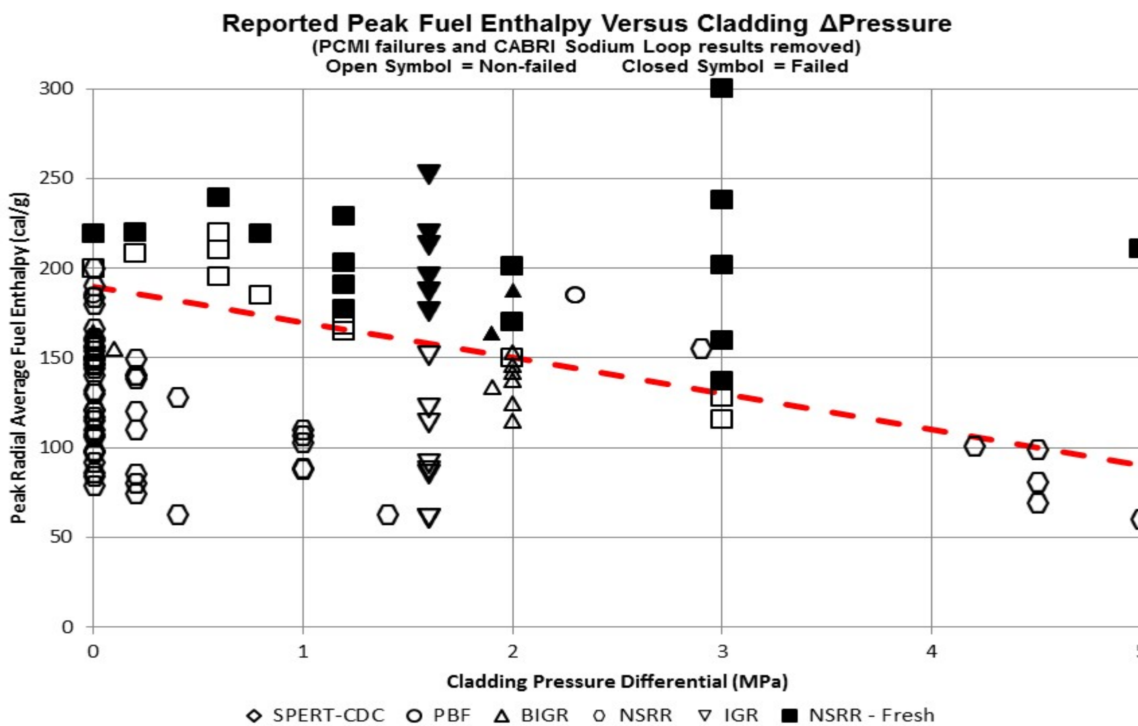
HT Cladding Failure Mechanisms

- High temperature cladding failure mechanisms
  - Brittle Failure: High-temperature post-DNB (film-boiling) oxygen-induced embrittlement and fragmentation
  - Ductile Failure: High-temperature cladding creep (balloon/rupture)

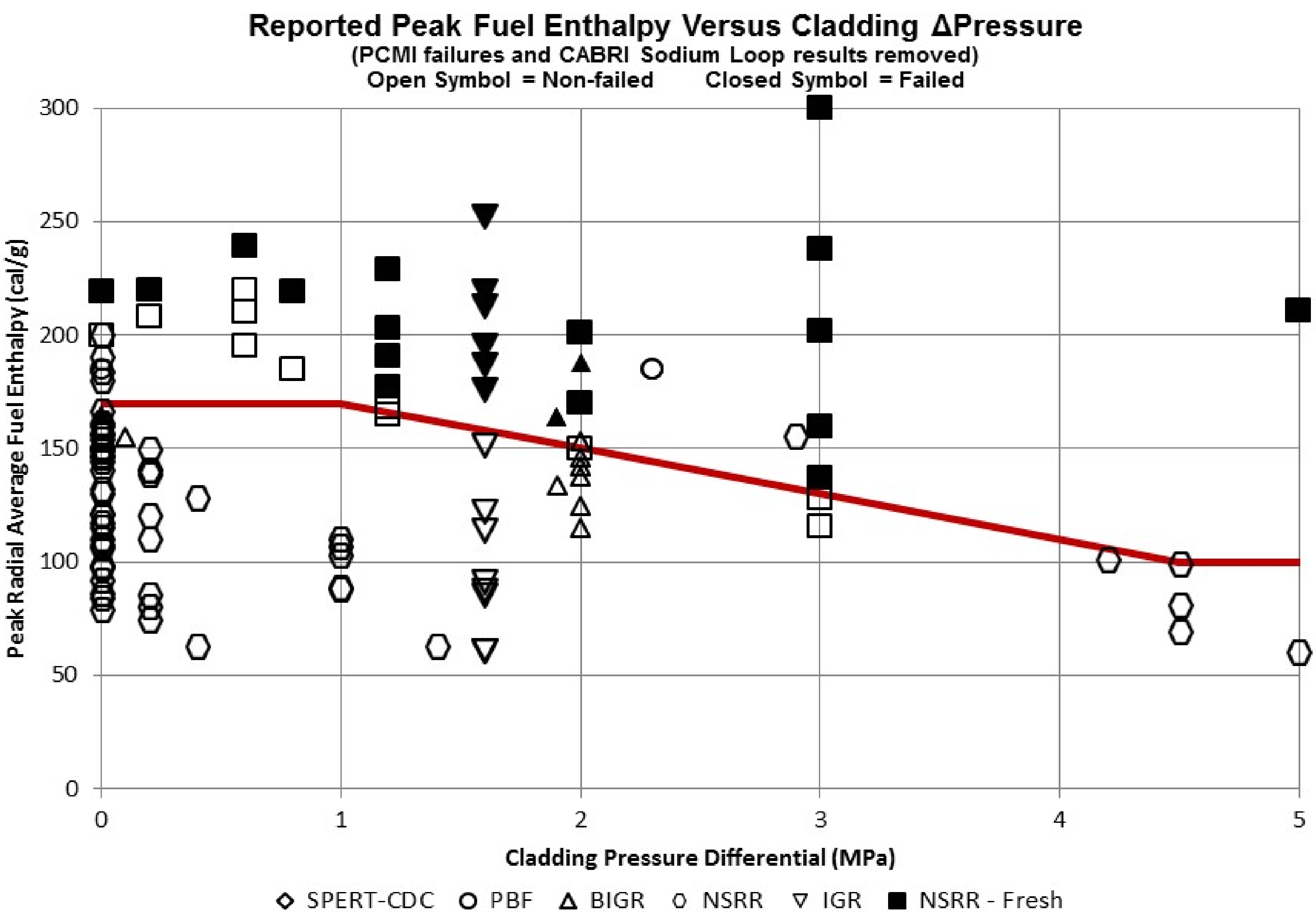


Sensitivity Study

- Database suggests ductile failure sensitive to cladding ΔP
  - Transient FGR contributes to rod internal pressure
- Below 100 cal/g fuel enthalpy, cladding remains below 800 ˚F and is no longer susceptible to HT failure
- Minimal BU effects



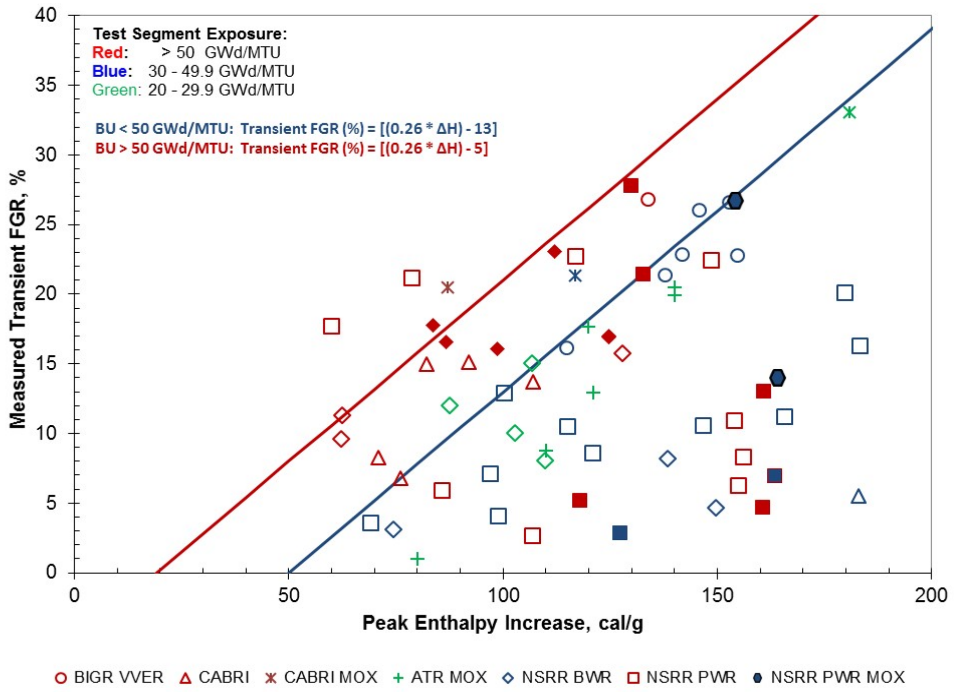
HT Cladding Failure Mechanisms



Impact of Transient FGR on RIP

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.



- Unchanged from DG-1327

DNB/CPR Cladding Failure

3.1 High-Temperature Cladding Failure Threshold

Figure 1 shows the empirically based high-temperature cladding failure threshold. This composite failure threshold encompasses both brittle and ductile failure modes and should be applied for events with prompt critical excursions (i.e. ejected rod worth or drop rod worth greater than or equal to $1.0). Because ductile failure depends on cladding temperature and differential pressure (i.e., rod internal pressure minus reactor pressure), the composite failure threshold is expressed in peak radial average fuel enthalpy (calories per gram (cal/g)) versus fuel cladding differential pressure (megapascals (MPa)).

For prompt critical scenarios which experience a prolonged power level following the prompt pulse, fuel cladding failure is presumed if local heat flux exceeds thermal design limits (e.g., departure from nucleate boiling and critical power ratios).

For non-prompt critical excursions, fuel cladding failure is presumed if local heat flux exceeds thermal design limits.

- Consistent with current guidance and practice

PCMI Cladding Failure

3.2 PCMI Cladding Failure Threshold

Figures 2 through 5 show the empirically based PCMI cladding failure thresholds. Because fuel cladding ductility is sensitive to hydrogen content, zirconium hydride orientation, and initial temperature, separate PCMI failure curves are provided for RXA and SRA cladding types at both low initial cladding temperature conditions (i.e., below 500 degrees F down to BWR cold startup) and high initial cladding temperature conditions (i.e., at or above 500 degrees F). The RXA cladding failure threshold is further refined for cladding designs with and without a barrier liner (e.g., sponge or low allow cladding inside diameter liner). The SRA cladding failure threshold is applicable regardless of the presence of a barrier liner. The PCMI cladding failure threshold is expressed in peak radial average fuel enthalpy rise (Δcal/g) versus excess cladding hydrogen content (wppm). Excess cladding hydrogen content refers to the portion of total hydrogen content in the form of zirconium hydrides (i.e., it does not include hydrogen in solution).

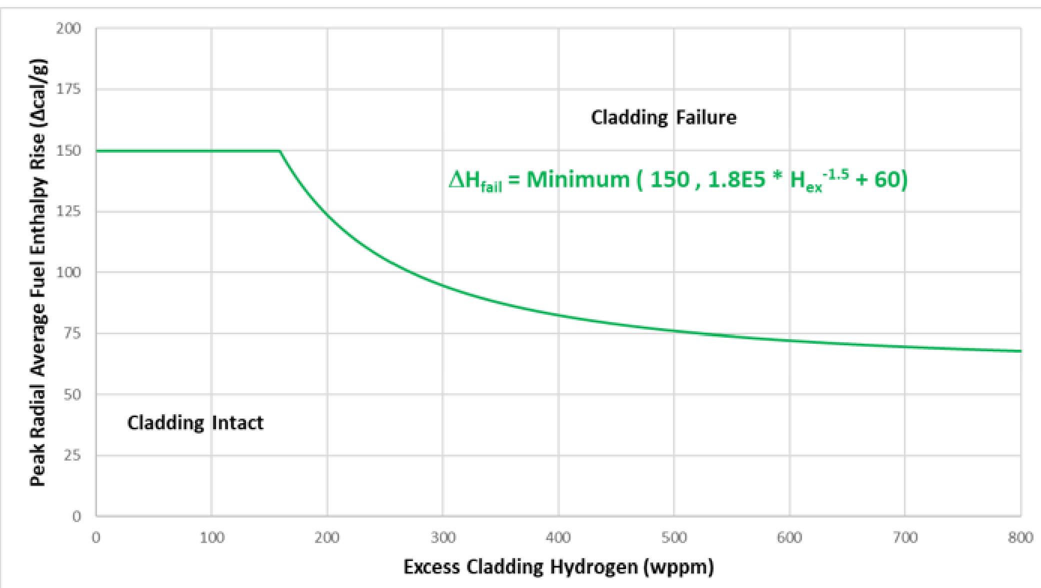
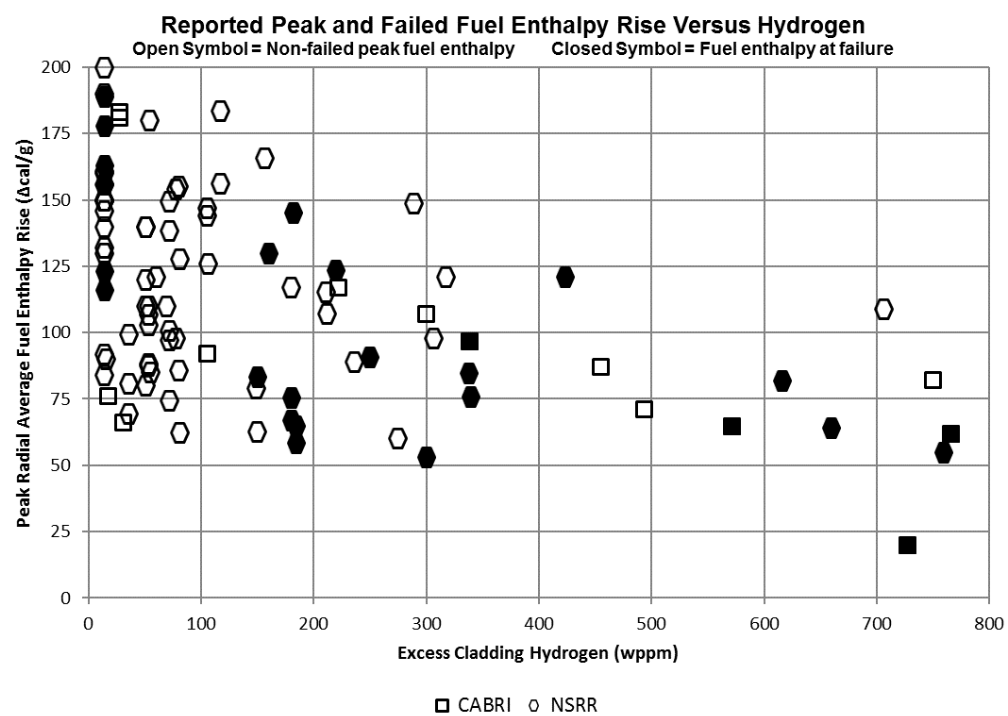


Figure 3. PCMI Cladding Failure Threshold—SRA Cladding at or above 500 Degrees F

Hydrogen-Enhanced PCMI

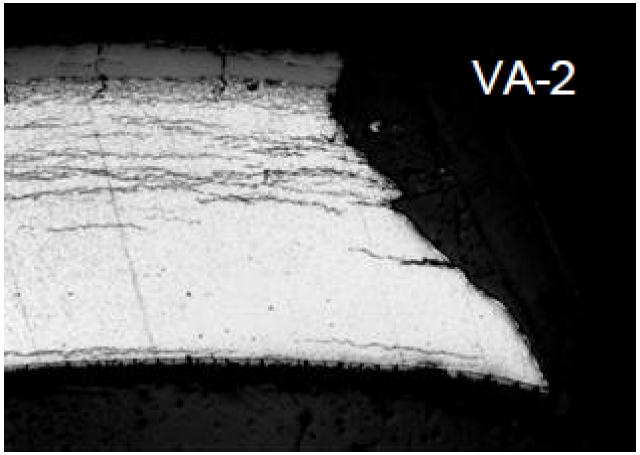
- Hydrogen, absorbed during normal operation waterside corrosion, forms zirconium hydrides which reduce the cladding's ductility
- Hydrogen uptake depends on several factors:
  - Time-at-temperature (residence time)
  - Power history and fluence
  - Alloy-specific corrosion and hydrogen pickup kinetics
  - Proximity to dissimilar metals
  - RCS chemistry
- Low burnup and low corrosion fuel rods retain sufficient cladding ductility and will likely fail by high temperature mechanisms before PCMI



Hydride Orientation

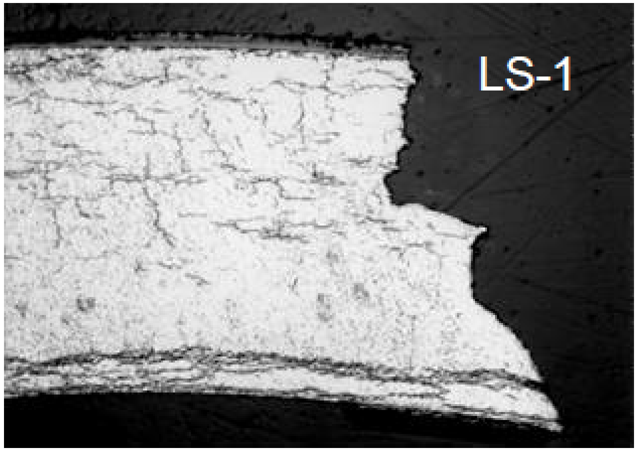
- Besides overall concentration, PCMI sensitive to hydride distribution and orientation which are influenced by:
  - Thermal and mechanical treatment during manufacturing
  - Stress state prevailing during hydride precipitation

PWR cladding (stress-relieve annealed)



Burnup: 79 GWd/t
H content: 760 ppm
Failure at 55 cal/g

BWR cladding (recrystallization annealed)



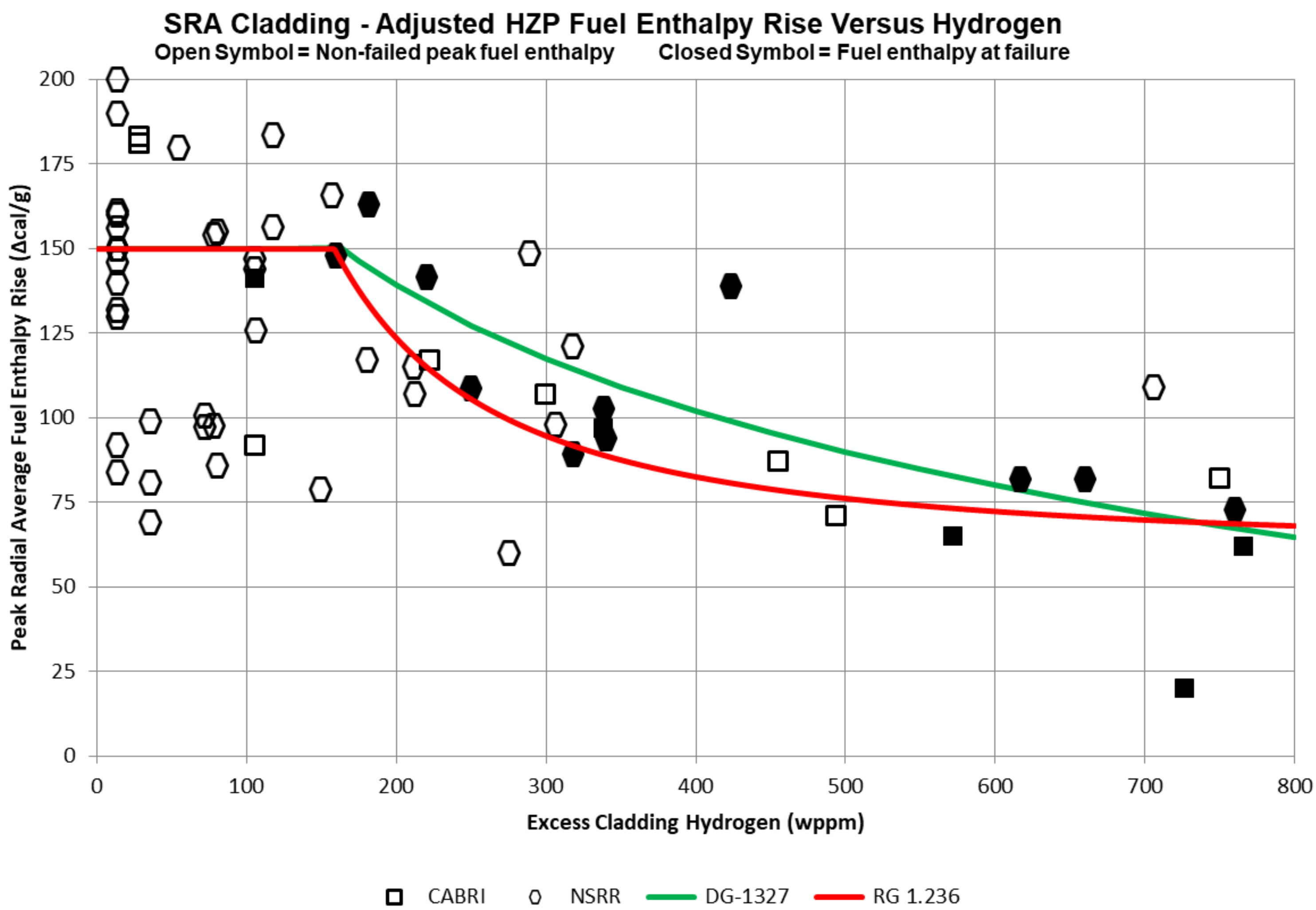
Burnup: 69 GWd/t
H content: ~300 ppm
Failure at 60 cal/g

T. Sugiyama, JAEA, "PCMI failure of high burnup fuels under RAI conditions," Fuel Safety Research Meeting, Tokai, Japan, May 2007

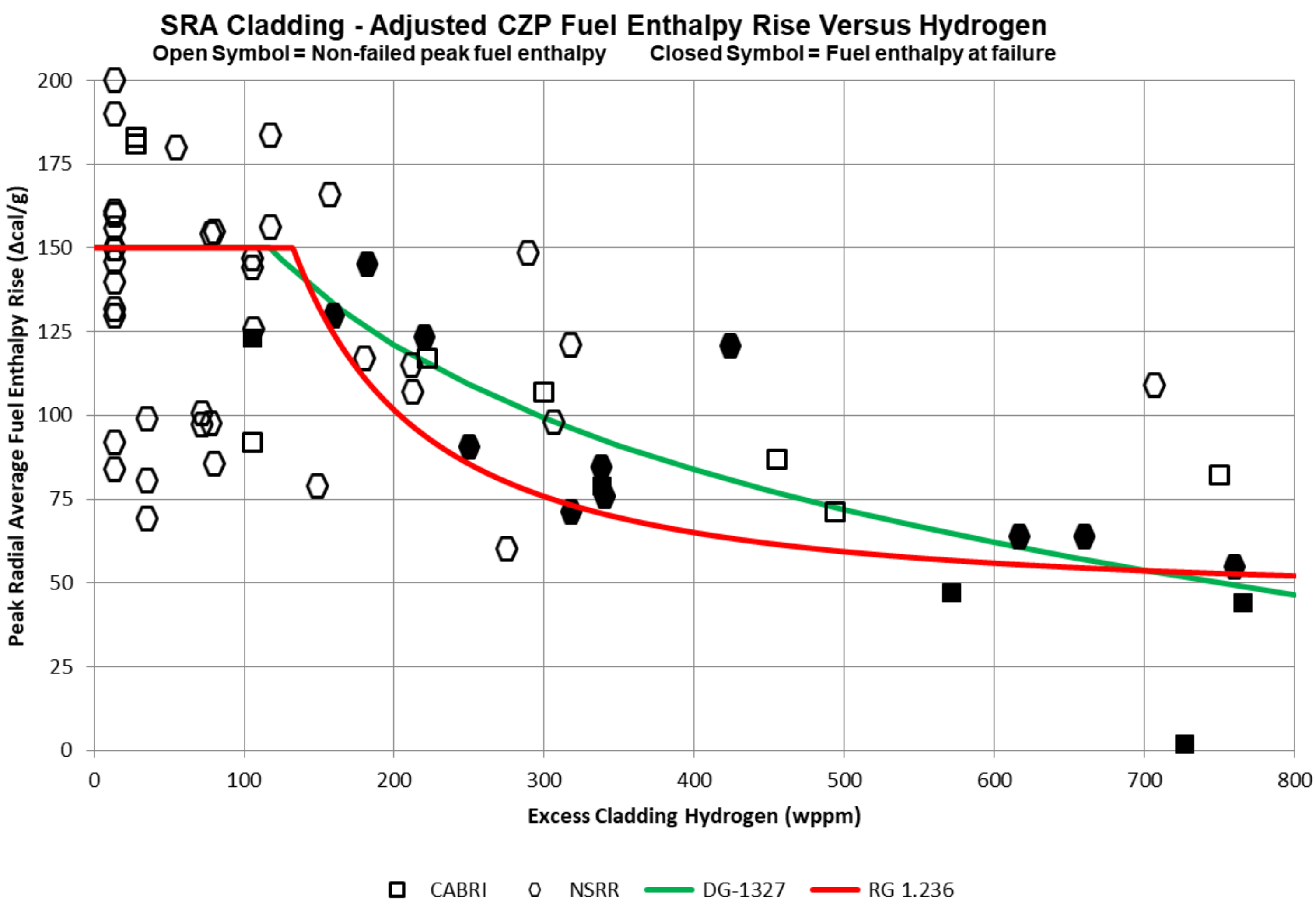
Separate PCMI Failure Thresholds

- Separate PCMI cladding failure threshold lines established which account for impacts associated with initial RCS temperature, excess hydrogen, and hydride sensitivity
  1. SRA cladding at high RCS coolant temperature
  2. SRA cladding at low RCS coolant temperature
  3. RXA cladding at high RCS coolant temperature
  4. RXA cladding at low RCS coolant temperature

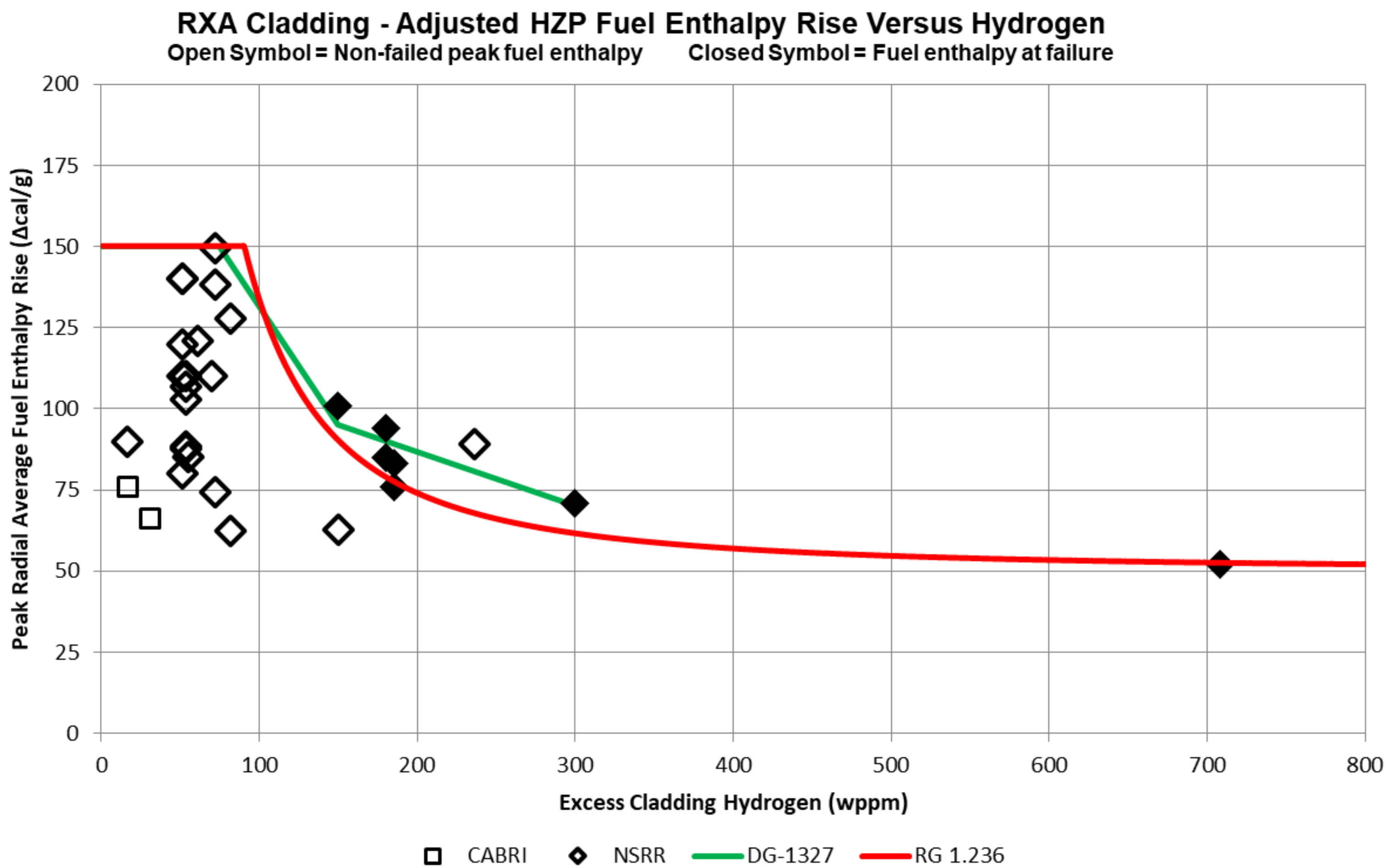
SRA Hot PCMI Failure Threshold



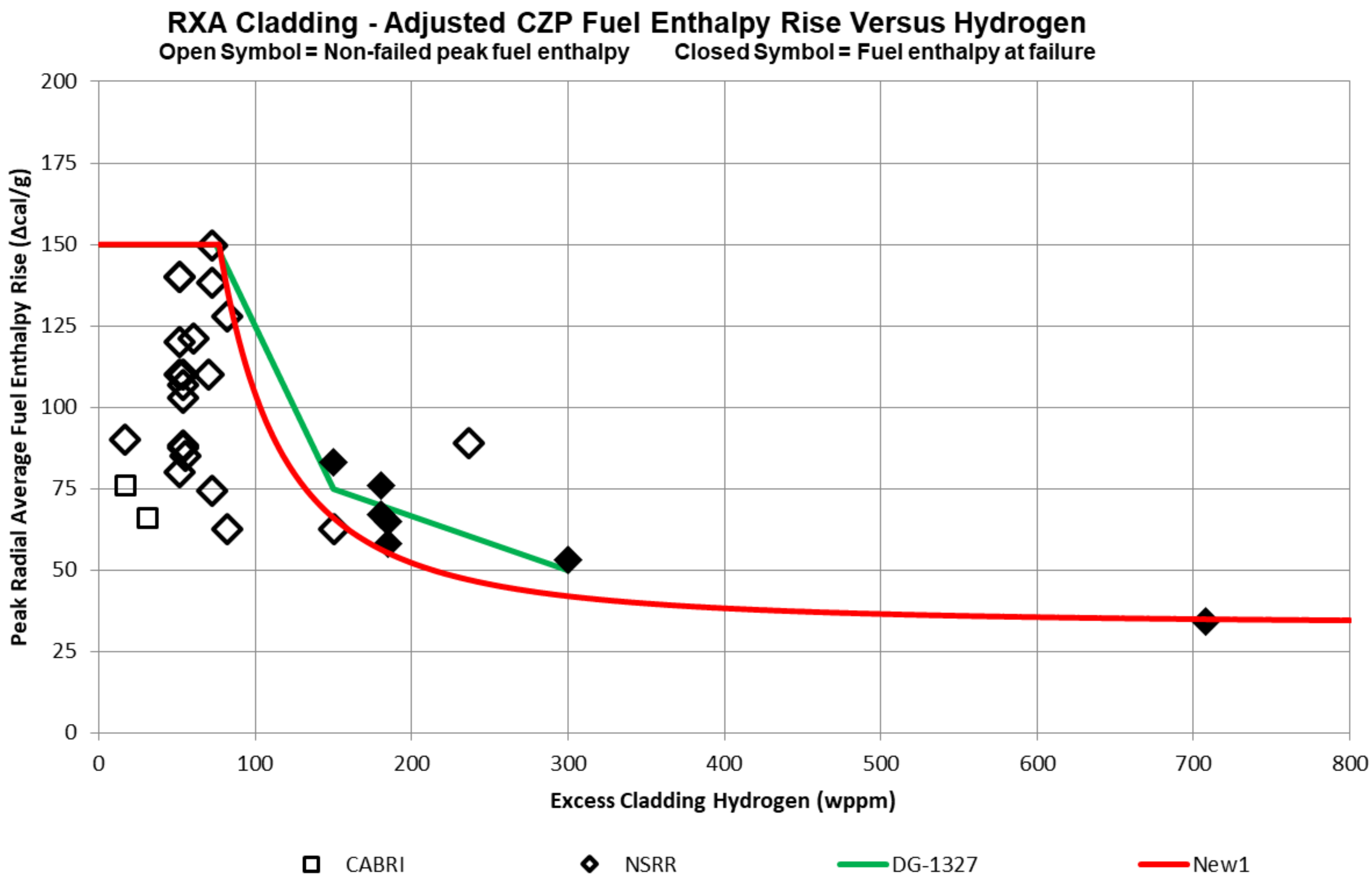
SRA Cold PCMI Failure Threshold



RXA Hot PCMI Failure Threshold

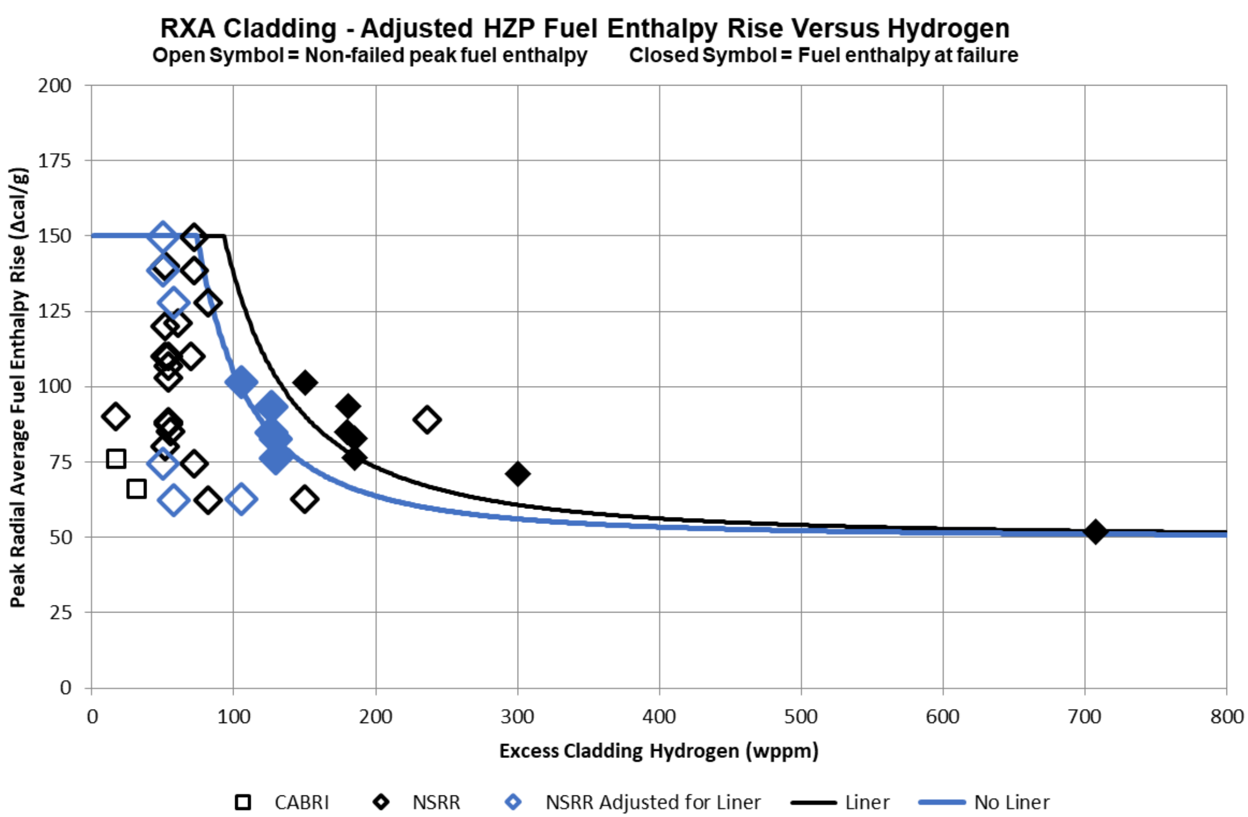


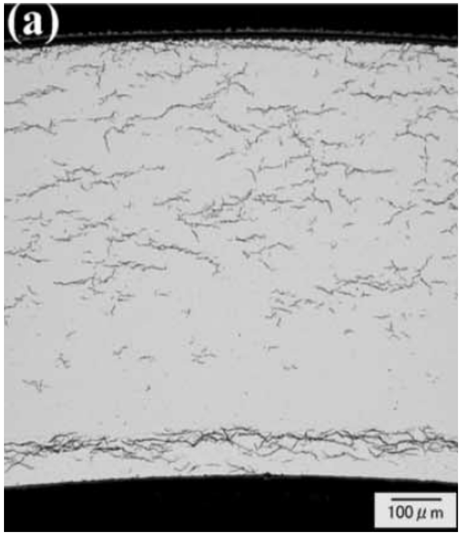
RXA Cold PCMI Failure Threshold



Impact of Barrier Liner

- In BWR liner (i.e., barrier) fuel, the natural or low alloy zirconium liner acts as a sponge for hydrogen.
  - Depletes base metal of detrimental effects of hydrides
  - Liner remains ductile even with high concentration of hydrides





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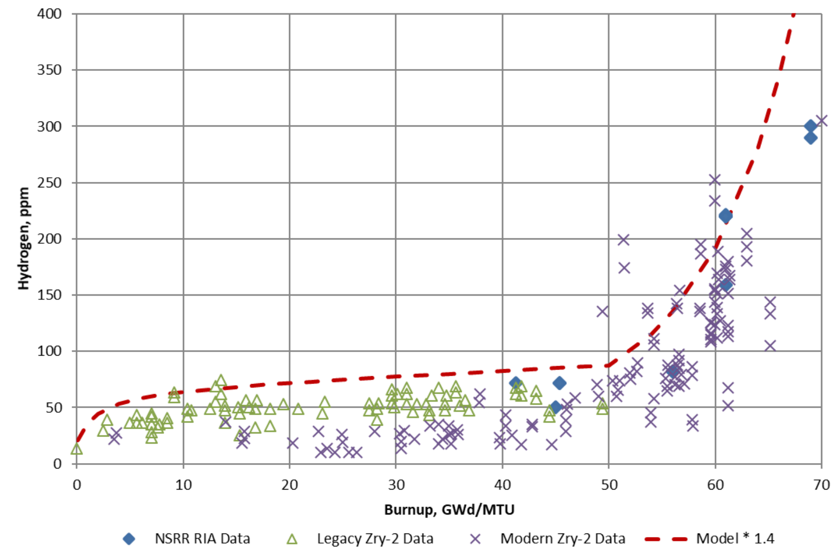
Hydrogen Uptake Models

- Originally published in draft RG 1.224 (2015) in support of 50.46c rule
  - DG-1263 public comment period in 2014



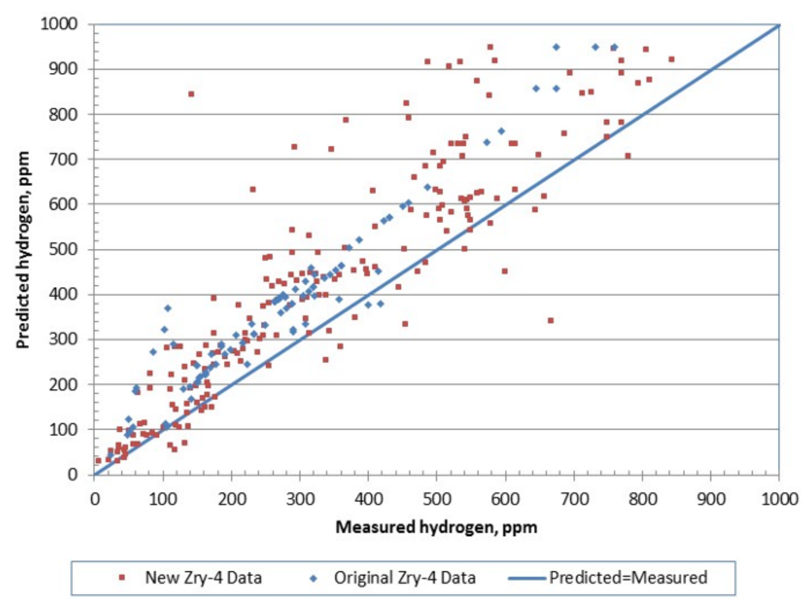
C-1. Zirconium Cladding Alloys in Pressurized-Water Reactors

Corrosion rates and the amount of corrosion at fuel discharge vary widely across the pressurized-water reactor (PWR) fleet because of alloy composition, operating conditions, and residence time (i.e., effective full-power days). Fuel vendors have approved fuel performance analytical tools along with corrosion models. In general, these corrosion models can predict a best estimate corrosion thickness as a function of effective full-power days and local operating conditions (fuel duty).

An examination of the empirical database of measured cladding hydrogen content for the current commercial zirconium alloys reveals that PWR cladding alloys do not exhibit the same breakaway hydrogen uptake at higher fluence levels as observed in Zircaloy-2 data for boiling-water reactors (BWRs). However, the pickup fraction does appear to be alloy specific. With consideration of the extent, uncertainty, and variability of the supporting database, the staff developed the following upper bound pickup fractions:

Zircaloy-4	20% hydrogen absorption
ZIRLO®	25% hydrogen absorption
Optimized ZIRLO™	25% hydrogen absorption
M5®	15% hydrogen absorption

These hydrogen pickup fractions should be used, along with a best estimate prediction of the peak oxide thickness using an approved fuel rod thermal-mechanical model, to estimate the cladding hydrogen content.



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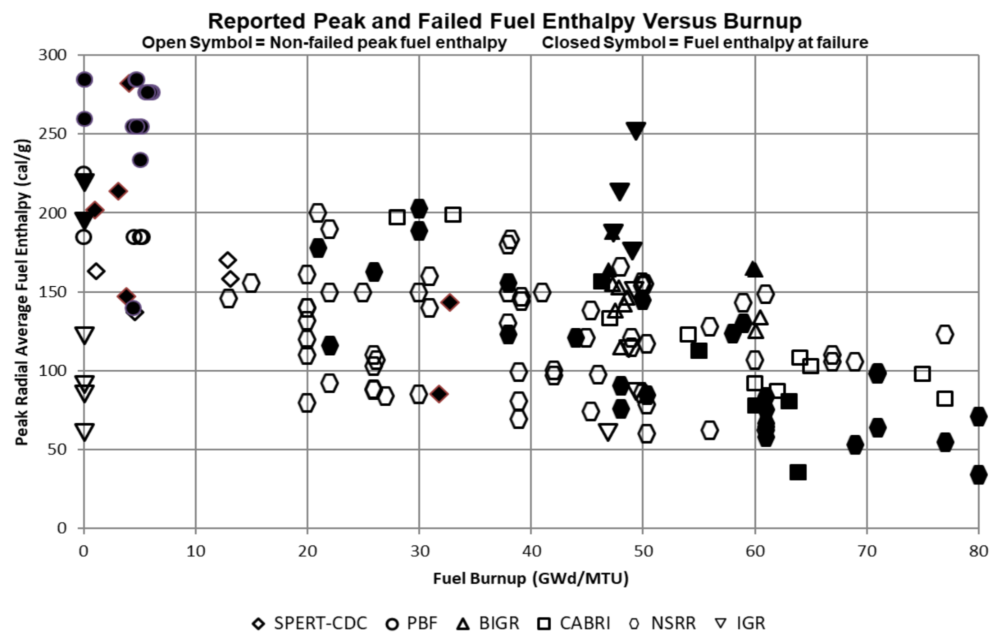
Burnup Extension

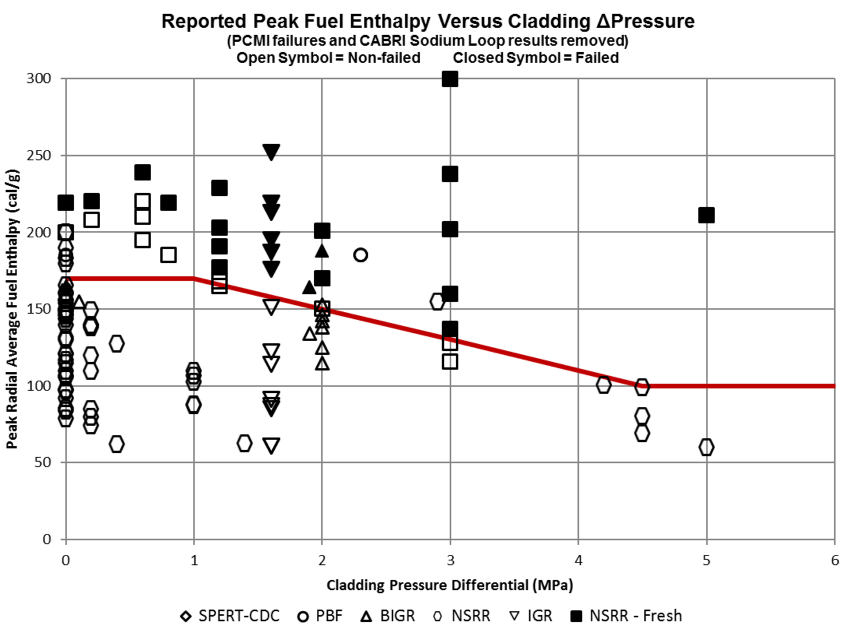
- In support of near-term licensing actions to extend allowable fuel rod average burnup to 68 GWd/MTU*, staff completed a critical assessment of the empirical database supporting RG 1.236 guidance (ML20090A308)
- For each portion of the guidance, the staff (1) investigated sensitivity with burnup and (2) assessed the extent of empirical database

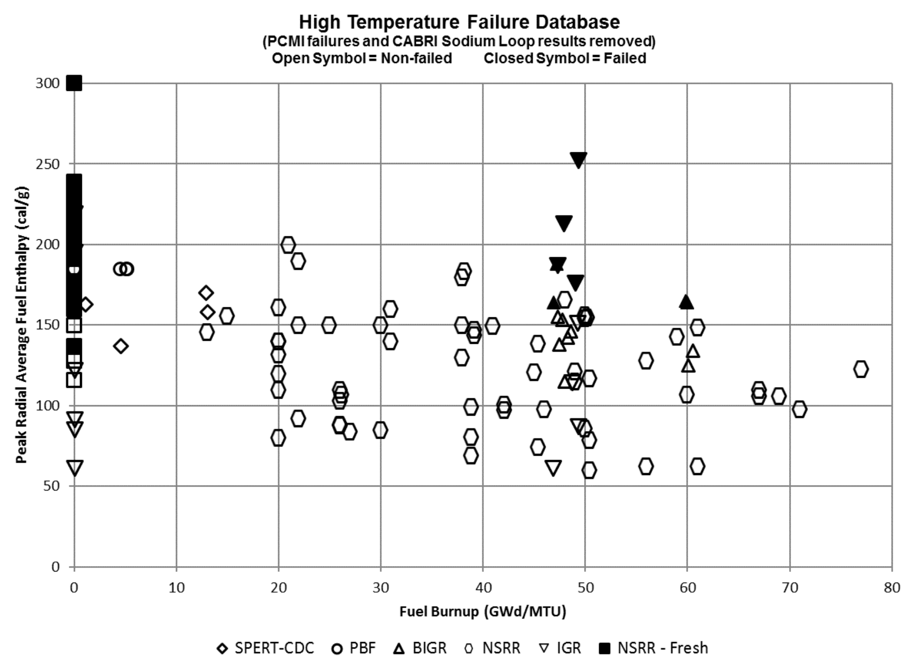


* Empirical database reports test segment (i.e., local) burnup. Rod average 68 GWd/MTU equivalent to approximately 75 GWd/MTU test segment burnup.

HT Cladding Failure Thresholds

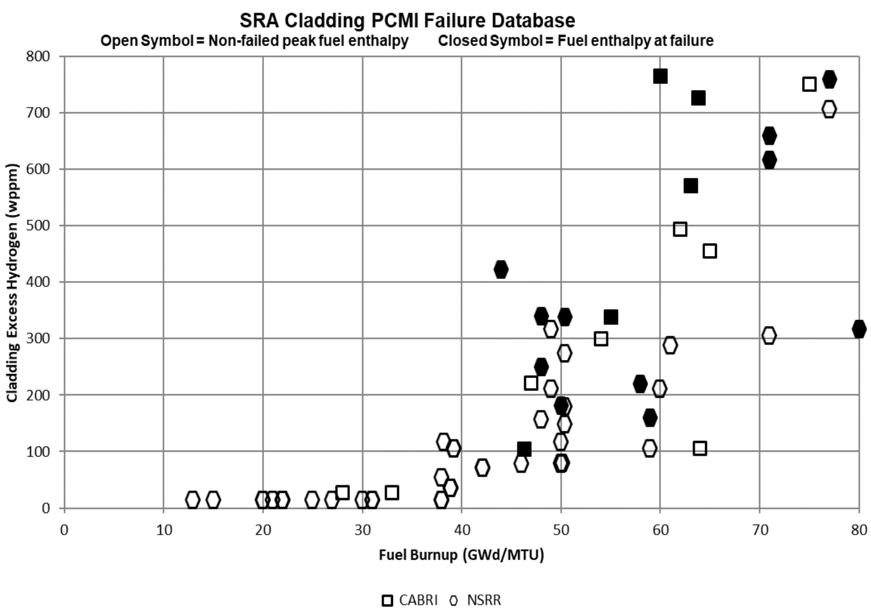
- Empirical database does not exhibit sensitivity with burnup
  - Burnup effects captured in rod internal pressure (cladding ΔP)
- Extended burnup test specimens survived above 100 cal/g

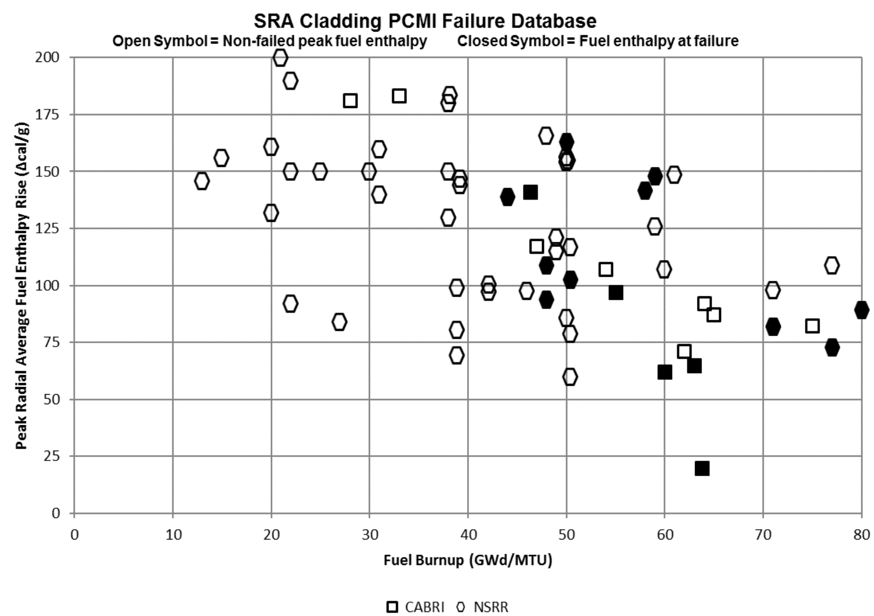




PCMI Cladding Failure Thresholds

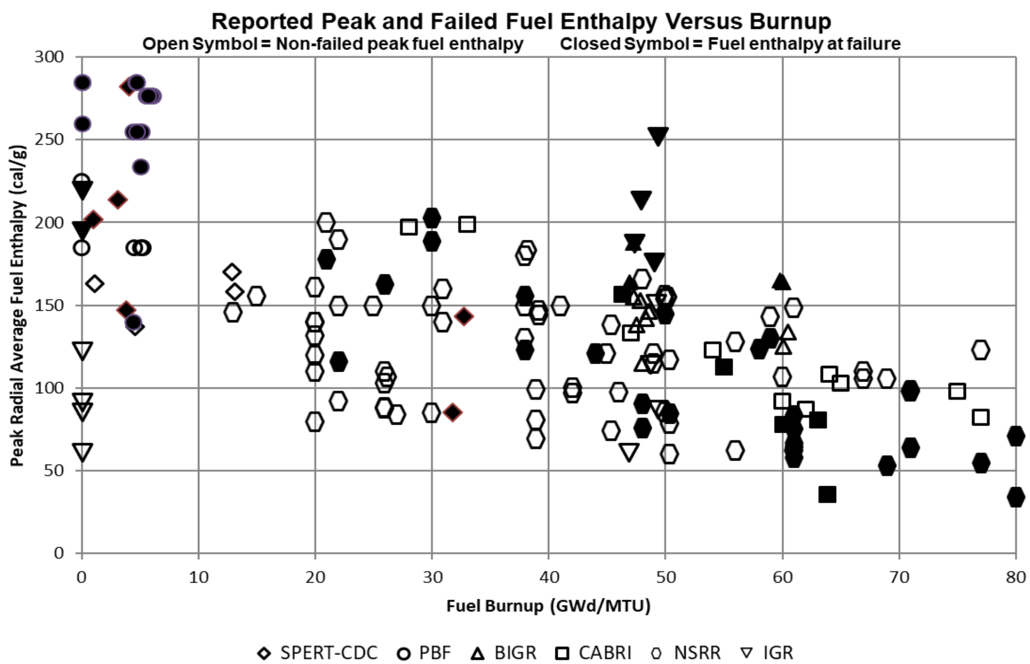
- Empirical database does not exhibit pronounced sensitivity with burnup
  - PCMI failure more sensitive to cladding hydrogen content
- Extended burnup test specimens survived above asymptotic limit (SRA -65 Δcal/g, RXA 50 Δcal/g)





Damaged Core Coolability

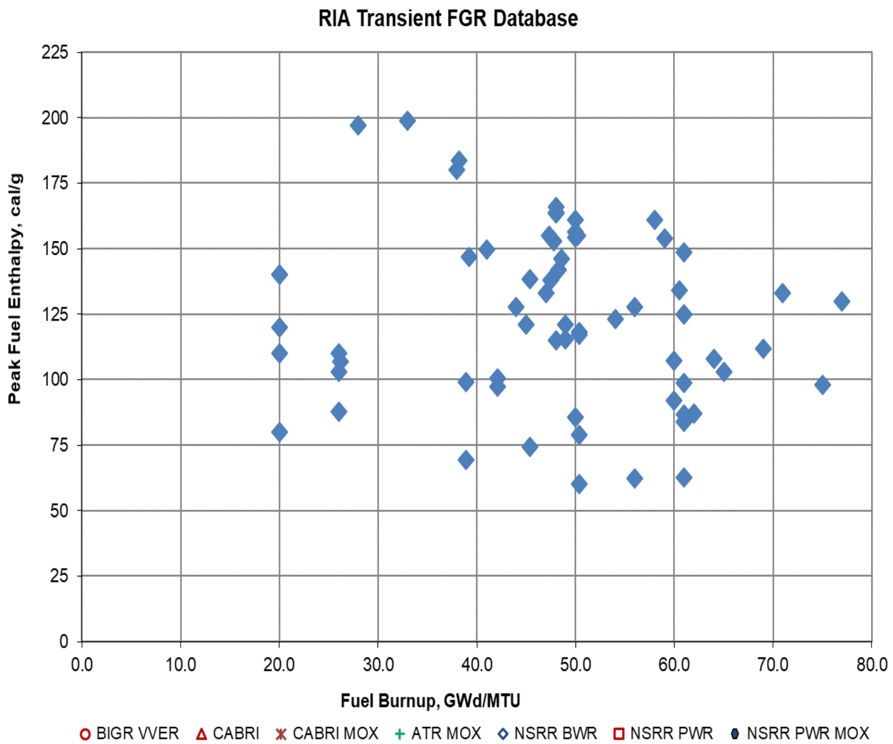
- While the damaged core coolable geometry empirical database does not include any high-burnup fuel rod segments, the detrimental effects of higher burnup must be accounted for to satisfy the limited fuel melt restriction.

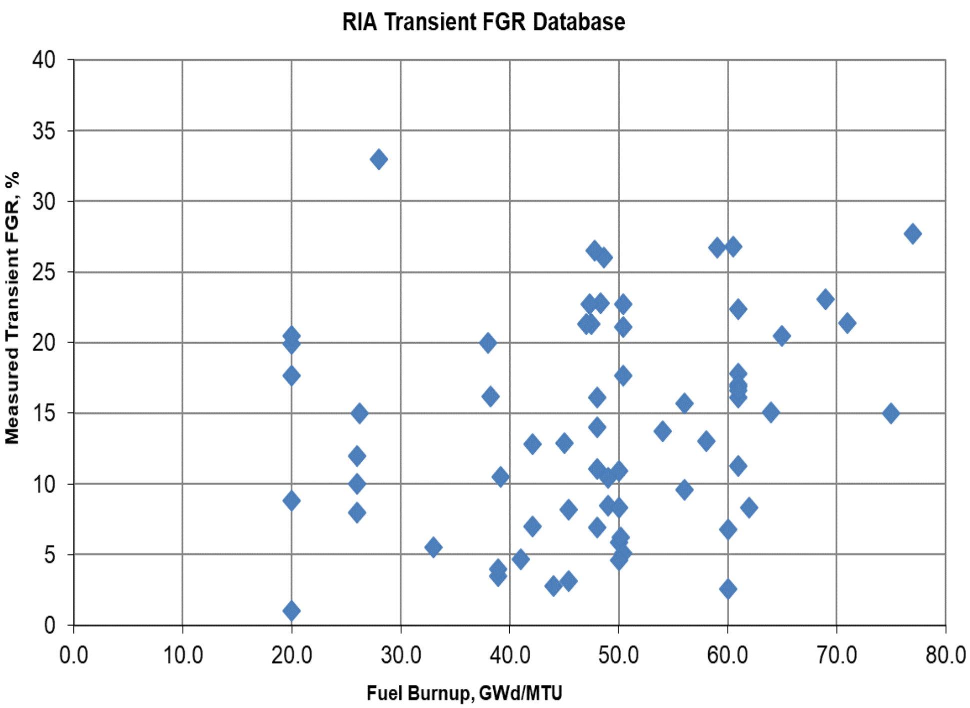


- Peak radial average fuel enthalpy should remain below 230 cal/g.
- A limited amount of fuel melting is acceptable provided that it is less than 10 percent of fuel volume. If fuel melting occurs, the peak fuel temperature in the outer 90 percent of the fuel volume should remain below incipient fuel melting conditions.

Transient FGR

- Empirical database shows sensitivity with burnup and deposited energy
  - BU-dependent FGR correlations
- Extended burnup test specimens fall below high burnup correlation





Burnup Extension

- Based upon the extent of the empirical database and sensitivity of important parameters and phenomena to burnup, the staff found RG 1.236 to be applicable up to a fuel rod average burnup of 68 GWd/MTU.
- This assessment was predicated on the use of approved core neutronics and fuel rod thermal-mechanical models. The following limitations were identified:
  - 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.
  - 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.

Conclusions

- Based upon latest research data, revised research data, new analysis, and international perspectives, the NRC staff has developed CRE/CRD guidance in RG 1.236
  - Represents a significant advancement in guidance
    - Separately captures fabrication-, burnup-, and corrosion-effects on fuel rod performance under RIA conditions

- ACRS and stakeholder involvement starting prior to 2007
  - Several ACRS briefings beginning prior to Interim Guidance
  - Numerous public workshops and 3 rounds of public comments