Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title: ACRS RPNM Subcommittee

Docket Number: n/a

Location: Rockville, Maryland

Date: July 10, 2012

Work Order No.: NRC-1739 Pages 1-158

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NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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SUBCOMMITTEE ON RADIATION PROTECTION AND NUCLEAR MATERIALS

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TUESDAY

JULY 10, 2012

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ROCKVILLE, MARYLAND

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The Subcommittee met at the Nuclear Regulatory Commission, Two White Flint North, Room T2B3, 11545 Rockville Pike, at 8:30 a.m., Michael T. Ryan, Chairman, presiding.

COMMITTEE MEMBERS:

MICHAEL T. RYAN, Chairman

J. SAM ARMIJO

DANA A. POWERS

HAROLD B. RAY

JOHN D. SIEBER

GORDON R. SKILLMAN

NRC STAFF PRESENT:

CHRISTOPHER L. BROWN, Designated Federal Official

MERAJ RAHIMI

DREW BARTO

BRIAN WAGNER

ALSO PRESENT:

JOHN WAGNER

ALBERT MACHIELS

MARCUS NICHOL

DALE LANCASTER

P-R-O-C-E-E-D-I-N-G-S

8:30 a.m.

CHAIR RYAN: Alrighty, good morning everybody. The meeting will now come to order. This is a meeting of the Advisory Committee on Reactor Safeguards, Subcommittee on Radiation Protection Nuclear Materials. I'm Michael Ryan, Chairman of the Subcommittee.

Subcommittee members in attendance are Dana Powers, Sam Armijo, Harold Ray. Let's see. Dick Skillman is here, Steve Schultz is not yet here.

MEMBER POWERS: No, he's not going to be here.

CHAIR RYAN: And Jack Sieber is here and Dana Powers. The Subcommittee will hear presentations by and hold discussions with representatives of the NRC staff, the Electric Power Research Institute and the Nuclear Energy Institute on ISG-8, Rev 3, burnup credit and the criticality safety analysis of PWR spent fuel in transportation and storage casks.

The Subcommittee will gather information, analyze relevant issues and facts, and formulate proposed positions and actions as appropriate for deliberation by the full Committee. Christopher Brown is the Designated Federal Official for this meeting.

The rules for participation in today's meeting have been announced as part of the notice of this meeting previously published in the *Federal Register* on June 26, 2012. A transcript of the meeting is being kept and will be made available as stated in the *Federal Register* notice.

It is requested that speakers first identify themselves and speak with sufficient clarity and volume so they can be readily heard. We ask at this time that you silence your mobile phones and other electronic devices.

The ACRS full Committee briefing is scheduled for September 6th, 2012. I might just make a short note, that this Subcommittee meeting follows onto work done by the former Advisory Committee on Nuclear Waste and Materials in 2007 and 2008, and I know they had a number of subcommittees and a couple of letters, and that work is available for reference, if anybody needs it.

We will proceed with the meeting, and I call upon Meraj Rahimi, branch chief of NMSS, to begin. Meraj, welcome and thanks for being with us.

MR. RAHIMI: Thank you very much, Dr. Ryan, thank you gentlemen, and my name is Meraj Rahimi. I'm the chief of the Criticality Shielding Dose Assessment

Branch in the Division of Spent Fuel Storage and Transportation in NMSS.

What we're going to talk about today is burnup credit, this new revision of ISG-8. This is the, I think, the work that we believe that has been completed on burnup credit, the work in making over 20 years, and this is the, really the closing chapter, we believe, on burnup credit for spent fuel storage and transportation cask in terms of providing complete guidance to the staff.

So this was actually quite a bit of work. I really want to acknowledge the people who really helped this work to come to fruition. Of course, this work was done in cooperation with NRR, NRO, Office of Research and the heavy lifting of the technical work, of course, Oak Ridge National Lab.

I want to acknowledge, you know, John Wagner from, you know, Oak Ridge National Lab, and especially I want to thank my team in the Criticality Branch, that we put together a team a couple of years ago to pull all the work together.

So again, I want to thank everybody for completing this work. So I think we're going to have a very good meeting, and especially we'll hear from the industry and with that, then let's go to the first

slide.

MEMBER ARMIJO: I did have one real background question.

MR. RAHIMI: Yes.

MEMBER ARMIJO: And this is this is titled for PWR Spent Fuel. But the approaches should be applicable to BWR fuel as well. Now what is the staff going to do about that or what has the staff done about BWR fuel?

MR. RAHIMI: Right. This work is only about PWR, and we will address towards the end, we have just started the work on BWR. Some preliminary has been -- some preliminary work has been done in the past few years on the BWR, a scoping study.

But as Drew will go over it at the end of his presentation, there is a plan for to write a separate ISG or revise this ISG for BWR burnup credit.

MEMBER ARMIJO: Okay, thank you.

MR. RAHIMI: So that's our agenda for this morning. So after I give the opening remarks, we'll give you a little bit of background, and Drew will cover really the overview of the ISG-8, in terms of the methodology and approach, and Brian Wagner from Office of Research will give a discussion on the misload probability, because this was one of the areas

that, with the help of Research, we wanted to focus on, and I know that the Committee's very interested in this area. So we should have a very good discussion on the mislead probability.

And John Wagner from Oak Ridge, he will cover the detail of the technical basis for the ISG, which mainly focuses on the code validation, depletion code and the criticality code validation, the method.

And Drew will cover the public comments that we've got so far, and our strategy in responding to those public comments. We certainly one of the things we did not want to finalize this ISG. We did want to get the input from the Committee also on the ISG, and our strategy in responding to the comments that we've received.

I believe we will hear from industry, from Al Machiels, EPRI, later on, and Marc Nichol and the industry views on the ISG-8. So with that, let's go to the next agenda.

I just wanted to give you a little bit of background in terms of this is a public meeting, you know, in terms of any members of the public, you know, are here. Really going back to the 70's, you know, why burnup credit for casks. I mean that first bullet really goes back in the 70's, the older generation

casks, the way they were designed. It was for younger fuel.

Basically, the idea was as soon as the fuel is discharged from reactor, it stays in the pool for a few months, or be shipped out to a reprocessing facility. So those casks in the 70's really were designed based on that concept.

So it was designed for a more younger fuel, and really subcriticality didn't come into play. It was mainly radiation and heat transfer. That was the driving parameters for the design.

MEMBER POWERS: That was because there wasn't much fuel in the cask?

MR. RAHIMI: I'm sorry?

MEMBER POWERS: There wasn't much fuel in the cask?

MR. RAHIMI: Yeah exactly, because we go back, analyze 1 PWR, 2 BWR casks. That was the old cask design. The rail cask design was, you know, transnuclear. It was, you know, 7 PWR. That was the 60-70 ton cask, but there was flux trapping there. So really heat and radiation was, you know, driving the design.

I guess in the 80's and the 90's, with the new generation of cask design, I realize reprocessing

is out of the pictures. They have to go to, you know, storage. So it's longer-cooled fuel. Therefore, the design is looking to increasing the payload capacity.

So with that subcriticality, now we've got more fuel than the cask. Subcriticality became one of the, you know, primary design drivers. Next slide.

So to achieve these high capacities, cask designers eliminated; basically, they wanted more real estate in that cask diameter. So they said okay, let's get rid of this flux trap. Flux trap is the spacing within fuel. Neutronics, you know, it helps you a lot, because you slow down the neutrons and you've got spaces, and they get absorbed in the poison plates.

So first thing they did, they got rid of the flux traps, so no spacing in there, and to get rid of that, the flux trap was based on the fresh fuel assumption. So they moved away from the fresh fuel assumptions into assuming the fuel is burned, which is the burnup credit comes into the picture.

What is burnup credit? It's just, you know, in designing your criticality controlled system of the cask, you take credit for reduction of reactivity that occurs with fuel burnup, due to depletion in the reactor core. That means that there is a net

reduction of fissile isotopes.

I mean you reduce, you know, U-235. Of course, you produce Pu-239. But overall, there's a net reduction in the fissile isotopes, and mainly you produce a lot of neutron-absorbing isotopes, actinides and fission products. So that's basically really the burnup credit.

Let's go to the next slide. So back in 2002, the staff, based on available data, started receiving application for burnup credit, and so we needed to put out a guidance for the staff to review these applications.

Based on the available data back then, we could only give credit, allow credit for actinides, because there were a lot of assay data, there were a lot of critical experiments that involved actinides, major actinide isotopes including U-235, plutonium.

So we had enough data, a lot of critical experiments. So that's what the staff could do at that time. So we issued ISG-8 Revision 2. Of course, you know, we used Revision 0 1998 and from 1998, you know, we made some modification, and in 2002, we should issued the guidance for to take credit for major actinides.

Then subsequently we had a SECY paper. We

had an SRM from the Commission, and there was a general direction from the Commission saying that the staff should focus its effort on using burnup credit as a means to insert more realism into spent fuel transportation cask criticality analysis.

So we take that as going beyond actinides only. Though you may not have all the data in there, but you know, take a risk-informed approach. Of course, in 2008, from ACNWM, that the letter went to Chairman Klein.

That was in regard with another exclusion really leading, but the burnup credit came into play into discussion, and the recommendation from the Committee was that the staff take a risk-informed approach in evaluating burnup credits.

So since then, we've been in the pursuit of that approach, and what has resulted today, that we now issued, in May, we issued the ISG Rev 3, which now includes, provides guidance to taking credit for fission products, and we have addressed a number of other items, including the burnup verification measurement, because we got a lot of feedback, especially from industry, in terms of the physical measurement.

So now we've provided an alternative to

physical measurement verification, and also we look at the other parameters. Like in ISG-8 Rev 2, the maximum burnup credit you could take up to 45 gigawatt-days, now we've extended all the way to 60, because since from 2002, a lot of chemical assay became available.

So that gave us the ability, you know, to extend the range of the burnup credit, which Drew will go over those. So with that, I'll turn it over to Drew, to provide an overview of what we've done in ISG-8 Rev 3.

CHAIR RYAN: Thanks, Raj. Drew.

MR. BARTO: Thanks, Meraj. I'm Drew Barto. I'm a senior nuclear engineer in the Division of Spent Fuel Storage and Transportation in NMSS. I'm going to go over what are the major changes to ISG-8 from the last revision.

Probably the major change is that ISG-8 Rev 2 recommended credit for only the major actinides for which we had sufficient data for co-validation for the depletion and criticality codes. With ISG-8 Rev 3, we are recommending additional credit for minor actinides and fission products. So that's an additional 18, actually I guess 20 isotopes that we'll get into, that we're recommending credit for.

Additionally, as Meraj just mentioned, there was sufficient data to justify allowing credit up to 60 gigawatt-days per MTU assembly-average burnup, and additionally, ISG-8 Rev 2 had a recommendation that in order to credit burnup, that cask users should perform a confirmatory burnup measurement prior to loading, to prevent a misload.

We've evaluated the whole misload issue, and provided an option in this revision for a misload analysis, accompanied with additional administrative loading procedures that may replace that measurement.

MEMBER SKILLMAN: Drew, by how much percentage or by how much hold-down does crediting the proof of minor actinides and fission products increase margin of criticality?

MR. BARTO: The rule of thumb is that the major actinides that we previously recommended credit for, represented about 75 percent of the reduction in k-effective that comes with burnup, and that these additional minor actinides and fission products represent about 25 percent.

So it doesn't sound like a lot, but if you look at how it moves your loading curve, it basically takes you from being able to load about somewhere in the neighborhood of 25 percent of the discharged fuel

population to more like 75 or 80 percent of the discharged PWR fuel population. So it's a significant increase in capacity basically, increase in the percentage of the discharged fuel population that you can transport in a high capacity cask.

MR. RAHIMI: I want to add something here that we should bear in mind, that at any time, you can transport all the fuel. But we're talking about the fully-loaded high capacity, you know, if you want it.

So even with no burnup credits, you know, you can transport -- at 32 PWR, let's say that's what you're talking about, getting designs, 32 PWR, 37 PWR cask, you can always derate a checkerboard pattern. You can load, you know, any fuel.

So what we're talking about, in terms of if you want to fully load it, that they do need, you know, burnup credit.

MEMBER SKILLMAN: I understand.

MR. RAHIMI: Yeah. So I think for the, especially members of the public, I want to clarify that, that the -- I mean you can transport really.

MEMBER SKILLMAN: All right, thank you.

MR. BARTO: I guess an additional note is the earlier generation of transport casks that Meraj was talking about earlier still exists, and are used

today, and they are lower capacity, but they can basically transport just about anything.

MR. RAHIMI: Fresh-fuel assumption. That's what they transport.

MR. BARTO: Okay. So getting into, you know, why we felt we were able to expand -- expand burnup credit, and this is an abbreviated list, really, of the work that's been done in the past 10 or 12 years on burnup credit. But these are kind of the most recent studies that have helped us the most.

In 2008, Oak Ridge evaluated a set of French critical experiment data that's known as the HTC critical experiment data, and it's critical experiments that were designed, as much as possible, to look like the major actinide distribution that would be present in 37-1/2 gigawatt-day for MTU burned PWR fuel.

So this was valuable, in that it helped us really validate well criticality predictions for the major actinides, which I've already mentioned are the largest component of the reduction k-effective of burnup.

NUREG, the second bullet there, NUREG/CR-7012 is a summary document of, I think, five other NUREGs that detail new sets of radiochemical assay

data for validation of the depletion codes.

That's part of why we felt we could raise the burnup limit up to 60, is that there's a great deal more high burnup data than was available in 2002.

Then the last two bullets there are two studies that were done by Oak Ridge National Lab, that are really pretty heavily referenced in the ISG. The first one develops an approach for -- actually several approaches for depletion code validation, and then actually provides some reference bias numbers that can be used directly by the applicant, and I'll get into how that's applied a little bit later.

Then the last one there, 7109, is an approach for criticality validation for burnup credit, and this report also provides reference bias number that can be used by applicants.

MR. RAHIMI: I want to add something here, and I think that I forgot to mention one of the really big change that it is, in my opinion, very beneficial to the industry.

Now we have provided an option that we have calculated the bias uncertainty. They don't have to do all the rigorous benchmarking. If they use the same code, the same cross-section single code that can take the number. I mean that's what is involved.

So I mean that's really a major change from the previous revisions that makes it much easier.

MR. BARTO: And I can follow up on that, in that we've already approved, I believe three burnup permit applications. I guess just off the top of my head, we're probably averaging 40 for review. They've all -- all of them have been different.

All of them have been beyond our Revision 2 guidance in some respect, and it's been quite a bit of back and forth with the various applicants on what's acceptable.

So we're anticipating that this revision will make a good deal of that back and forth, at least about the validation part, will make that go away.

So with respect to code validation in ISG-8 Revision 3, we felt that the availability of the high quality actinide critical experiment data that we were able to obtain from the French gives a greater degree of confidence in the actinide criticality validation than was previously available.

Additionally, there's new sets of chemical assay data that expanded the available database for, particularly for fission products, for depletion code validation, and also extended the burnup range.

So all of this new data was folded into

NUREG/CR 7108 and 7109, where Oak Ridge developed some alternative isotopic depletion and criticality code validation methodologies, wherein the applicants can either use those methodologies themselves or can use the reference values, and I'll discuss in a little bit how that can be done.

So for validation, Revision 3 recommended crediting both actinides and fission products up to 60 gigawatt days for MTU. John Wagner from Oak Ridge is going to go into a lot more detail about these particular reports. But we did coordinate with NRR and NRO on this, which is kind of new.

The criticality safety components of the agency, up until about five years ago, did not really talk to each other that much. So there were some disparate methodologies used for kind of similar analyses in the agency and, you know, since about 2005-2006, we've tried to coordinate a lot more.

We've actually started a criticality safety technical advisory group that's agency-wide now, and we meet quarterly. So there's a lot more communication, and hopefully in the future we'll see a lot more coordination on these types of research efforts.

But just to reiterate, these two reports

develop new isotopic depletion code methodologies and reference bias and bias uncertainty values.

There's also new minor actinide and fission product criticality code validation methodologies, as well as referenced bias values, and these reports provide recommendations on how to use the referenced values, and how one would use the methodologies to develop those values.

So the ISG basically references these two reports, and states that applicants can use the reference bias and bias uncertainty numbers developed by Oak Ridge, without performing explicit validation, provided they use the same code and cross-section data that were used in the Oak Ridge analyses.

It's very important that they do that, because everything that's done in the Oak Ridge reports points to the fact that these biases and bias uncertainties are really driven by errors in the data, at least for criticality. On the depletion side, it's somewhat driven by the uncertainties in the actual experimental measurements themselves.

MEMBER SKILLMAN: Do all licensees have access to these codes and cross-section data?

MR. BARTO: The codes, the only code system that was used in these NUREG reports to develop these

numbers was the SCALE code system, you know, primarily because that's the code system developed by Oak Ridge National Lab. But also some of the methodologies are -- the SCALE code system is geared -- there are components of the code system that are ideal for performing these sorts of analyses that aren't available on other code systems.

But we believe we have it worked out in the ISG, to where you're not limited to using the SCALE code system, and you may use other codes. For criticality, we're primarily talking about MCNP as for another code. It's very important that they use the same cross-section data. So you couldn't come in with a code, a European code that uses the JEF or JENDL cross-section libraries, because these reference bias and bias uncertainty numbers wouldn't, aren't applicable.

MEMBER SKILLMAN: Is there an upper limit on enrichment, for which these analyses are valid? For instance, licensees that went to 24-month fuel cycles, applied for a license amendment and received it for five weight percent fuel. I see in the write-up the span of data is between 4.6, I mean 2.453 and 4.657 weight percent 235.

So if an applicant has five weight percent

fuel as part of its discharge batches, are those covered by these analyses?

MR. BARTO: We believe they are. ISG states that the upper limit on the applicability is five weight percent.

MEMBER SKILLMAN: Is five weight percent? Thank you.

MR. RAHIMI: Yeah, and also, just to complete the answer to your first question yes, this SCALE code is widely available. Any vendor that needs it, they can call the Radiation Shielding Information Center at Oak Ridge and they can send them, you know, the code.

So it's widely available actually, you know. Most everybody, even overseas, you know, other countries, you know, they call in to get the code.

MEMBER SKILLMAN: Thank you.

MR. BARTO: And as far as the depletion validation data, let me back up a little bit here, the second bullet there, NUREG/CR 7012, like I mentioned, that's a summary document. There's five other NUREGs that feed into that document, that in each detail a set or a group of sets of radiochemical assay data.

They're described in pretty good detail. There should be enough information in those reports to

be able to model those chemical assay measurements.

MEMBER SKILLMAN: Okay, thank you.

MR. BARTO: So again, very important that the applicant use the same code and cross-section data, as was used in the Oak Ridge report, in order to use the reference bias and bias uncertainty numbers.

It's also one of the criterias that the applicant's storage or transportation system is demonstrated to be similar to that evaluated in the NUREG/CRs. There's a number of ways they can do that. There's sort of a standard way that critical experiments are compared, through a number of criteria.

Materials, geometry, some physics parameters like EALF and things like that. But there's also sensitivity uncertainty analysis that is able to compare systems, and the system that was used in the Oak Ridge reports is sort of a theoretical cask system.

It's known as the GBC-32, and it's set up to look like the kind of high capacity casks that we're seeing in our applications. So we don't expect that they'll be anything widely different from the theoretical cask system. So it ought to be relatively simple for an applicant to demonstrate similarity.

And then the other key criteria for using these reference values is that the applicant still performs traditional criticality code validation for the major actinides, using the data that is available.

The HTC data, the French actinide critical experiment data has been demonstrated to be very applicable to validate criticality validation for major actinides.

CHAIR RYAN: Forgive me for asking this question for you, but have you done any comparisons or results comparisons with other countries other than France, other folks that have used the same tools?

MR. BARTO: It's, these methodologies are relatively new. Actually they're kind of completely new in the NUREGs. So we have sort of tentative plans to communicate with the international community, to see if we can't get them to do similar analyses. It's nothing that's been discussed with anyone, because it's a tentative plan.

CHAIR RYAN: I'm just curious. It's clearly been a good coordination with the French program. I just was curious if it has gone beyond that or not at this point. Thanks.

MR. BARTO: So in summary for code validation, it's often helpful to think about code

validation for burnup credit in four parts. You can divide it into validation for the major actinides, and also for the minor actinides and fission products, and it needs to be done for the criticality analysis and also for the isotopic depletion analysis.

So in the, I guess upper left, for the major actinides, the criticality validation, this is the, what I just mentioned, how the applicant can perform this analysis using applicable fresh UO2, MOx and the HTC experiments, using more or less a traditional criticality validation approach.

For the minor actinides and fission products, for the criticality validation, the applicant can use the Oak Ridge-supplied reference bias number. Then for the isotopic depletion analysis, for both major actinides and minor actinides and fission products, applicants have the option of using the Oak Ridge-supplied bias and bias uncertainty numbers, or using the Oak Ridge-developed validation methodologies that are detailed in the NUREG.

With respect to burnup measurements that were recommended in ISG-8 Rev 2, we've had Oak Ridge do some work on, related to misloads. The first bullet there, NUREG/CR 6955, was basically a consequence analysis.

In other words, how much change in k-effect could you get with a misload, various misloads. NUREG/CR 6988 was for an overall look at measurement technologies, what's available to use in, you know, how do in-pool out of core measurements compare with in-core measurements, and also a review of, you know, how in-core measurements are done, versus how the out of core measurements are done.

We also had the Office of Research look into probability of a misload in a spent fuel cask, and Brian Wagner from Research is going to talk about that in a moment.

So we've collected all this information on misloads, and we've determined that a misload is a credible event that must be considered. However, we believed it was appropriate to allow an alternative to the measurement, which is basically performing a misload analysis to demonstrate that your system is not overly sensitive to a misload, and then also to have additional administrative loading procedures that are geared towards preventing misloads.

So the ISG has a number of recommendations for how to perform a misload analysis. We recommend that applicants look at a single severely underburned fuel assembly misloaded into the worse location, and

this misload should be chosen such that the reactivity of the fuel bounds 95 percent of the underburned fuel population with 95 percent confidence.

I've got a graphic on the next slide that will kind of show where that is, and based on the misload events that we've seen so far, the majority of them have been multiple assemblies, not just a single assembly.

So we recommend that a misload analysis be performed that also looks at multiple moderately underburned fuel assemblies, and the criteria that we recommend for that is that half the cask is filled with a fuel assembly that bounds the reactivity of 90 percent of the total discharge fuel population, and this will be more clear on the next slide, when we show the graphic.

There's also, we recommend a reduced administrative margin for this analysis. The criticality administrative margin is typically .05 for a criticality analysis. But we consider the misload to be sort of an upset condition, and we are -- the administrative margin can go as low as .02 for this analysis, and that's consistent with upset conditions that we've looked at in other regulated areas for criticality safety.

And again, we recommend a set of additional administrative loading procedures that I'll talk about a little more. We got a lot of industry comment on our proposed procedures, and just a sampling of them here, ensuring that there's no fresh fuel in pool at time of loading, since our misload --

The misload analysis that we recommend doesn't include fresh fuel, or and also independent third party reviews of cask loading. Again, that's just a handful of -- we've recommended, I believe, seven in the ISG. That's not intended to be an all-inclusive list, but I'll go through that in a little more detail later.

MEMBER SKILLMAN: Drew, on your second bullet there, you said that when you looked at this, the bulk of the misloads were moderately underburned. Is that a large population of events or a small population of events?

MR. BARTO: It's a small population of events, but as we'll see in Brian's presentation, there's a small population of casks that have been loaded. So it doesn't take many misloaded casks to get you into credible space.

Now an important point about the misload events is that no storage casks are being loaded right

now under the burnup credit assumption. So the misload events that we've looked at are misloads with respect to other criteria.

So there's really no data on or we didn't glean any information about how they might be misloaded with respect to a, you know, maximum, a minimum burnup requirement. You know typically, we're talking about maximum burnup requirements now for the radiation heat transfer.

So the misload events that we've seen have failed those criteria, and there's really not information about, you know, if they come in on the low side, which is what you're concerned about, criticality safety.

MEMBER SKILLMAN: Thank you.

MR. BARTO: So this is just sort of a visual of the recommended misload analysis criteria. This sort of cloud of numbers in the background there represents the entire discharged, entire permanently discharged PWR fuel population as of 2002, which is the last time DOE collected complete data on discharged fuel.

The green line represents what you might expect to get with a loading curve for a cask system. The blue line is a representation of where you might

expect 90 percent of the total population, a reactivity that bounds 90 percent of the total population.

The red line is, represents a reactivity that we would expect that would bound 95 percent of the underburned population. So you know, there are handful. If you look at the graph there, you know, this is actual data, actual permanently discharged fuel, and there are a handful of assemblies that are below that line.

But we have, we've recommended some procedures that we think will prevent, you know, any kind of --

CHAIR RYAN: There's nine of them below the red line.

MR. BARTO: What's that?

CHAIR RYAN: There's nine below the red line.

(Simultaneous speaking.)

MR. BARTO: Probably nine boxes, but there's multiple assemblies in each box.

MEMBER SKILLMAN: Okay.

MEMBER ARMIJO: Each number represents the number of assemblies in that burnup enrichment box.

MR. BARTO: Yes, the number of assemblies.

(Simultaneous speaking.)

MEMBER ARMIJO: Oh I see, okay. So those would be ones that we're worried about.

MEMBER ARMIJO: You'd worry about --

CHAIR RYAN: Yeah. I mean you really have to make sure those were isolated from the rest of the population.

MR. BARTO: Right, and one of the recommended procedures that we have is that if you have these assemblies in your pool, you identify where they are and, you know, after you load, you'd go back and make sure that they're still there.

It's important. You know, if you actually go and look at this data, you'll find more often than not if you see a grouping of assemblies like this, they're all from one plant. So you know, certain utilities may be able to make the argument, you know, I don't have any -- I just don't, I've never discharged a fuel assembly that's that low for burnup.

MEMBER SKILLMAN: The appearance of the even number in each of those cells in the lower right-hand corner suggests that it might even be test assemblies or very, very fresh fuel assemblies --

MR. BARTO: Right.

MEMBER SKILLMAN: --that had some mechanical

or other failure. Hence, they're discharged right into the box.

MR. BARTO: Yeah.

MEMBER SKILLMAN: And so they're very underburned and very reactive.

MR. BARTO: And I haven't looked at those specific boxes down there, but we have looked at some of the, some of the boxes that are higher up, that might even be into the yellow range, and just almost every time you look at one of those, it's either entirely from one utility or from one unit, or possibly from two different units.

But it's usually limited to where they're from, because the practice now is, at least from what I understand about how fuel assemblies are used, it's not an ideal situation to discharge a fuel assembly at that high of an enrichment, and to never put it back in the core.

If it comes out because it's damaged, it's now usually trying to reconfigure it, to get it back into the core. So I sort of, without having actually looked at this specific data that's below that red line, I would suspect that that's fairly old.

MR. RAHIMI: Yeah. Some of those assemblies, actually I looked at them many years ago.

It was, you know, of the stainless steel variety, you know, the older version, that really they didn't do well and they had to be discharged. I believe Yankee Rowe, you know, made some of those assemblies. As you said, it was from earlier assemblies that --

MR. BARTO: So this -- I believe DOE is gearing up to do this survey again, and I think if you, again, I can only speculate. But I would anticipate that you will see much more growth into the high burnup range at the higher enrichment, and I would doubt that you're going to say many, if any at all, in that little corner there.

MEMBER ARMIJO: Does each utility have, each plant have this curve that's unique to their specific plant?

MR. BARTO: There's a curve that's unique to the cask design. But you could do this for each utility essentially. You could --

MEMBER ARMIJO: They have the data. They just have to put it together.

MR. BARTO: Yeah, right.

MR. RAHIMI: The lines that you see, for example, the green line, this is for the cask, GBC-32 cask. Depending on the cask, that loading curve, you know, falls into fuel population. I mean the fuel

population you see, that's the entire inventory in the pool right now, I mean, as of 2002.

CHAIR RYAN: So getting back to maybe something Drew said earlier, let's say each plant did have their own, you know, data on this block. They could then come up with a risk for misload test at their plant, is that right?

MR. RAHIMI: That is true, that is true, that if each plant, they say okay, this is the population in my pool, and this particular cask is coming for shipment. This is the loading curve for this cast, superimpose it on the population, and yes, they could, you know, identify the ones that are below the loading curve.

CHAIR RYAN: I may be off base, but it seems like that would be very helpful information for a plant to evaluate.

MR. RAHIMI: But of course, you know, the certificate that we issue for the cask is not -- it could be used anywhere, at any plant.

CHAIR RYAN: No, no. I understand that, but I think that implementing that particular plant, you know, it's particular to their fuel pool and what's in it; right?

MR. RAHIMI: Yeah, yeah, yeah.

CHAIR RYAN: It seems like there's a --

MR. RAHIMI: Actually we did this -- we approved TN40, which was right now there's currently a bunch of TN40 casks in storage, in fact for Prairie Island.

They came in, they made a transport certificate request, and the way they did the misload analysis, I mean before we developed these criteria to their pool, Prairie Island pool, we put the TN40 loading curve on that. Based on that, what you just said, I mean we have them to do misload analysis.

CHAIR RYAN: So that way it would be a big campaign of some sort to remove fuel, or pull it out? That's when that would get done.

MR. RAHIMI: Yeah.

CHAIR RYAN: So they're going to probably do that as a matter of course anyway, or is that a requirement that they do a misload analysis?

MR. RAHIMI: Right now, it is the recommendation in the ISG, that you need to do a misload analysis. It is a recommendation, and in there, applicant is free to come with a different proposal.

CHAIR RYAN: Okay, fair enough.

MEMBER ARMIJO: I just want to make sure I

understand this curve. Let's say in your plant you had all of your fuel had a burnup enrichment -- had burnup enrichment values that were above, let's say, the blue line along this curve.

MR. RAHIMI: Uh-huh.

MEMBER ARMIJO: And you could prove that. NRC was satisfied that that was actually accurate. Would a burnup -- would a misload analysis be required in that situation?

MR. BARTO: I think for a, you know, if a specific utility came in for a, I guess site-specific transport certificate, we would certainly accept that without a misload analysis. If they could demonstrate again that they've never discharged fuel event with low burnup.

(Simultaneous speaking.)

MR. BARTO: So, you know, we would certainly consider that.

MEMBER ARMIJO: Great, thank you.

MR. BARTO: Okay. We've gotten a little bit behind schedule. I'm going to let Brian talk a little bit more about misloads and particular misload probabilities.

MR. WAGNER: Brian Wagner, NRC Research for the general risk analysis. So in our report we did

three things. We reviewed actual cask misload events to determine underlying causes, and to identify common failure modes and see what insights could be gained from them.

The main purpose was to calculate the probability of one or more casks being misloaded, and we did that using two separate methods. First, we just looked at the empirical data and calculated it.

Second, we used an event tree model to model the process for loading a cask, and tried to theoretically calculate the probability of misload. That was done to give us additional insights.

And finally, we considered the impacts that burnup would have on the probability of an assembly being misloaded, and we basically concluded that the burnup of an assembly isn't likely to affect the probability that the assembly would be misloaded, with a few exceptions like for fresh fuel assemblies, which are visually different from other casks, I'm sorry, from other assembly.

And that if an assembly is misloaded, that its burnup -- that it will be basically be chosen at random from the spent fuel population.

MEMBER POWERS: Why did you make that random assumption?

MR. WAGNER: Looking at the empirical data, we didn't see really any pattern, and you know, couldn't really think of a mechanism for what would cause a correlation for the burnup of the assemblies being misloaded. So generally --

MEMBER POWERS: I think -- it seems to me that the potential for correlation of misloads is extremely high, because if you develop one in the selection of assemblies, and you bring it through sequence, then every single one of them after that would be in error.

MR. WAGNER: Yes. Certainly the correlation for assemblies being misloaded is high. But as far as the --

MEMBER POWERS: Well, you're saying --

(Simultaneous speaking.)

MEMBER POWERS: --is uncorrelated. It's completely random. But once you get off --

MR. WAGNER: Then you're off.

MEMBER POWERS: Then you're off, right.

MR. WAGNER: Right, yeah.

MEMBER POWERS: Okay, I agree.

MR. BARTO: I'd like just to point out that you've got to think about it looking at the population. If you have a misload, you're more or

less selecting from this pot of discharged fuel, and it would be a smaller pop if you're looking at a specific site.

But the idea is that you will look at the misload events that have happened, and I'll just let Brian talk about those a little bit more. But it's more or less kind of a loss of control of the assemblies that you're selecting. So I think what Research found is that you're more or less selecting at random from this pot of fuel assemblies.

MR. RAHIMI: Right, I mean 20 gigawatt-days versus, you know, 15 gigawatt-days. So the probability of misloading 20 gigawatt-days, if they're both underburned. So there is no correlation. So you could pick any of the underburned fuel.

MR. WAGNER: And it's certainly possible to imagine situations, I guess, where there maybe could be a correlation. But it's hard enough to predict what that would be, that you all just have assume that it's going to be random.

So we identified seven misload events and one-year misload event. I wasn't going to go through the details of each of them individually. But all of them stem from errors in the initial planning process, or the procedures used to load the cask, and they all

involved multiple assemblies and multiple casks. So I have some of the numbers listed there.

One error in particular actually led to casks at different sites owned by the same licensee being misloaded. All events were discovered after all the standard reviews had been done, sometimes years later and after the casks had been sealed.

We included the near-misload event, just because it's the only example we had of a single assembly almost being misloaded. So we wanted to include that, just to show what mechanism could lead to assembly misload.

MEMBER ARMIJO: Yeah. That is what I always envisioned was the issue, in that someone intended to pick up this assembly, but it somehow picked up the wrong assembly. What you point out here is there's systems in identifying what needs to be opened, and errors in those systems, at least more prevalent than just an isolated wdo;l;sdkf9:21.

MR. WAGNER: Right. That was surprising to us too.

MEMBER ARMIJO: That to me is one disturbing event, somebody misreading a serial number marking on an assembly.

CHAIR RYAN: Are these four examples, or are

these, is this the whole population of misloads?

MR. WAGNER: This is the whole population of what we identified. As Drew mentioned, well as he at least alluded to, these are the ones that were eventually caught so far. So presumably there could be more, and there's also an issue of reporting requirements.

If, you know, there's what we might consider a misload that still satisfies the requirements, then it wouldn't necessary be reported to us.

MR. RAHIMI: Yeah. If they haven't violated a tech spec, you know, or anything they don't report. But these are the ones that involve specifically casks. I mean you've got misload in the pool, you know, the racks. We haven't even included all those misload events.

CHAIR RYAN: Okay. Yeah, just the casks. All right.

MR. WAGNER: Just the casks.

CHAIR RYAN: Okay.

MR. WAGNER: And I'll also make the point now that one of the events seemed to actually be caused by the fact that the requirements were complicated.

CHAIR RYAN: It's interesting to think

though for a second that if you have a misloading in a pool, like if say one position in the pool as opposed to some other way and that error is brought forward, then you could be thinking that a cask loading is just terrific, and it's not.

MR. WAGNER: It's not, yeah.

CHAIR RYAN: So I would think you'd want to pull that string a little harder, and go back to look at misloads in fuel pools, to see what that population looks like.

MR. RAHIMI: Yeah, we're going to -- I mean we did look at actually some INPO reports. We even looked at that. But we focused on the casks, since --

CHAIR RYAN: I understand that.

(Simultaneous speaking.)

CHAIR RYAN: I think that should be on your homework list, to think about is there something there that needs to be further evaluated.

MR. WAGNER: We'll talk about that a little bit.

CHAIR RYAN: Okay, and Jack, do you have a question?

MEMBER SIEBER: That's all based on the serial number and a misload some place along the line.

MR. WAGNER: Right.

MEMBER SIEBER: If it's possible to catch that because of the serial number on the fuel --

MR. WAGNER: Right. Our understanding is generally they're checking the actual serial number as they're loading the casks --

CHAIR RYAN: And that's kind of the backstop, you know, against which those, what's called misallocation in a pool can be overcome. But I think it's helpful to pursue that, just to see what is that population, how big is that?

Is it real small or is it prevalent, you know. That gives you some idea of the intensity with which we should continue to look at serial numbers if it's a bigger number.

MR. WAGNER: Right.

MEMBER SIEBER: And mispositioning in the pool is sort of a setup for an error when they're --

CHAIR RYAN: Exactly, that's my point.

MEMBER SIEBER: And some of the serial numbers are hard to read, you know, quite hard.

MR. WAGNER: Yes. I think Drew will talk about it. I think some of the recommendations get at that point. So the probability they'll end up with this is 20 casks misloaded out of 1,200 that have been loaded. I think that's as of the end of 2009, which

gives you on the order of 10 to the negative 2 per cask.

MEMBER POWERS: Which is kind of the human error rate for everything known to man, right?

MR. WAGNER: Right. All right. So for the second method, to develop the event tree model, we modeled the process for loading a cask at a high level. So the first major step is choosing what assemblies you're going to load in the cask, and as we've been discussing, that's a complicated step.

There's lots of requirements that go into that, and you have to satisfy all of them. Next, you make a fuel move sheet, which contains the serial numbers, allocations of the assemblies that we move from the spent fuel pool to the cask, and that's the sheet that's actually used by the crane operator to do the moving. The next step is actually transporting --

MEMBER ARMIJO: Brian, before you go forward, of the four examples shown on your Slide 17, the errors are really in that first layer, that first level, choosing the assemblies and creating the fuel move sheets.

MR. WAGNER: Yes.

MEMBER ARMIJO: So that's a misplanning error, rather than a mishandling of the fuel.

MR. WAGNER: Absolutely.

MEMBER ARMIJO: I think you should start thinking there's a problem working there, that's more significant than somebody just basically picking one assembly in error.

Even though the fuel sheet's told him to pick another one, he just -- it just seems to me there's a bigger problem working down there in the planning of these loadings, and verification that you're doing the pool time calculation right or the -- your database is solid.

It just seems to me that that's most of your errors are in that, yeah, that category, and not really simple misloads. It just see misloads as a mechanical thing, picking up the wrong thing and putting it in, even though the planning was good.

But you're showing us that in these, most of these examples, planning before is a problem.

CHAIR RYAN: Is that a fair comment? I mean in your view the planning is really the root cause of this?

MR. WAGNER: There's a lot of requirements, and if you don't satisfy any one of them, then it's misload.

MEMBER ARMIJO: You call it a misload, but

--

MR. WAGNER: Yes.

MEMBER ARMIJO: For example, on Palisades, it was a cooling tent, so it was misloaded because it wasn't using a heat requirement.

MR. WAGNER: Right.

MEMBER ARMIJO: But it wasn't necessarily misloaded from a burnup or a criticality issue?

MR. RAHIMI: That's correct, yes, yeah. And yeah, to go back to again, Dr. Armijo's, you know, argument, that it's true. When you look at it's most of the events happening on the planning phase, for example. Some example, you go into the details what exactly happens, you know, the indexing, for example.

A crew, refueling crew, you know, they work together. They were used to some indexing. They skipped some letters, and the new crew came in and that was the error, you know, the indexing. When you really read about details of these events, then it's mostly in the planning and identifying, you know.

If you give the right information, you know, if the fuel had a leak and most of the time pick up the right --

(Simultaneous speaking.)

CHAIR RYAN: Is there any way of taking a

hard look at -- what are the events that caused the error, they mislaid something. But if it's a planning error, it seems to me that it's something systematic that's gone wrong. If all these errors are in one bin or the bulk of them, who's pulling that string, to see what really is the root cause.

I mean what you describe is not necessarily the root cause. It's the root cause for the next thing down the road, what caused that to begin with. Is there a corrective action that's needed there or some new process or proof process or something of that sort? Have you taken a homework assignment for that target?

MR. BARTO: Well I think, you know, we certainly can pull that string more. But I think in looking at these events and getting the probability analysis from Research, that's pretty much what led us to the misload analysis.

CHAIR RYAN: Yeah.

MR. BARTO: We're basically concluded that they're going to happen, and from a criticality safety perspective, we believe that the better way to deal with it is simply to show that your cask isn't going to have a criticality issue, even if you have one.

CHAIR RYAN: Yeah.

MR. WAGNER: And I should mention there has been some HRA work done on --

MR. BARTO: There has been.

CHAIR RYAN: What work, sorry?

MR. WAGNER: There's been some HRA work.

MEMBER RAY: Does Appendix B apply? Part 50, Appendix B, does it apply to these activities you're talking about?

MR. RAHIMI: Yes. The loading activity, yes, yes, because in the loading --

MEMBER RAY: It wouldn't necessarily apply. I'm just asking does it apply.

MR. RAHIMI: Yeah, it does apply.

MEMBER RAY: Because it ought to then produce the information Mike is asking for.

MR. RAHIMI: Well actually we do have the information. One of the NUREGs that Drew listed that Oak Ridge did a few years ago, went through all the reactor event reports, in terms of a misload even, you know, on the pool sides.

And so we did, you know, look at -- that's where we started, you know, looking on the reactor side in the pool, all the misload events, and but what we wanted for this, for the ISG for casks, we then focused okay, the misload for casks. Let's focus on

that.

CHAIR RYAN: Yeah, but that's kind of separating the dancer from the dance.

MR. RAHIMI: Yeah, but you're actually --

CHAIR RYAN: There is a second problem, and now that your basic situation is, because as you've just explained, there may be errors in that basic setup that you don't recognize.

MR. RAHIMI: The basic setup, yes.

CHAIR RYAN: So you can't separate the dancer and the dance.

MR. RAHIMI: You're right. Yes, we will do that. We will look into it further on the reactor side.

CHAIR RYAN: So we'll think about it too. But there may be a recommendation coming out of that discussion. I think that's something -- that's a productive moment from the work you've done so far.

MEMBER ARMIJO: Yeah. Just for information. On these events, misload events that you've identified here, I'm just presuming that these led to some corrective action programs within the various utilities and they corrected whatever.

Can you verify that that's what would have happened or did happen? Let's say at Palisades or

North Anna and Grand Gulf, that their process for identifying what needs to be -- identifying the things that need to be loaded, that their process is okay, their database is okay? Because if that's wrong, it's amazing that you don't have more.

MEMBER SKILLMAN: It seems that there should have been an information bulletin or something, not at a stunning, attention-getting level, but at a basic level, telling the licensees hey, you know, there appear to be some process problems, in actually loading the cask.

It appears that one in a thousand, that they physically latch onto the wrong assembly. One in a hundred that there is an error in some part of the transportation process, whether it was the calculation of heat load or the symmetry or some other such thing.

But I would have thought there would have been an information bulletin or something like that to our licensees that would communicate a heads-up. Looks like you're picking the right assemblies; you may be using the wrong logic to get those in casks.

CHAIR RYAN: That may be a good homework question to take away for the preparation of the full Committee meeting, and maybe cover a little bit more of these other things we've just talked about it about

wanting to pull the string.

MR. WAGNER: And I think also this problem, obviously, isn't unique to casks. We have been talking with NRR and NRO on this issue. So maybe that can be part of our homework, is to --

CHAIR RYAN: Okay. Great, terrific.

(Simultaneous speaking.)

MEMBER POWERS: When you do your -- in the error analysis, did you talk to the ATHEANA folks about errors of commission and things like that?

MR. WAGNER: Not really. It's done a whole lot of time. Doing the human error analysis, we -- I'll get this in a little more detail. But we use THERP values, THERP values, techniques for human error rate prediction.

MEMBER POWERS: The problem that comes to mind is you have a highly structured, lots of checks and balances, checks in the process setup, lots of opportunities to catch an error, and you end up with an error rate that THERP has for everything that's known to man.

And ATHENA is really set up to look at process, that to identify what's wrong with the process, and you know, that's what it's designed to do.

It's, I mean it's god-awfully difficult, but it's a god-awfully difficult problem, because you've set up a system here that should be making sure that you're coming down to an error rate that's extremely low. But you came down to one that is the same for me adding up numbers in columns.

You know, I will make roughly a big mistake every time I do one of those that I do, with nobody checking me. But if you ask both Armijo and I to do the same thing or him to check my work, you will cut that in half, and you know, it just strikes me on my to-do list, not necessarily for a full Committee meeting, but you know, as long as you're planning to go into BWR plan as well, and to be re-looking at some of this stuff, go chat with those plant people, and see if there's anything they can do for you.

And understand that the magnitude of an effort that it is, that maybe sometimes these efforts are not worth it, not cost-effective. Sometimes they're very effective. I mean I think the fire folks found that a lot, when they take Chadwick for the ATHEANA folks, and came away saying okay.

There are other areas that it's not worthwhile. It's just a different perspective on how to work the process, as opposed to, you know, doing

the THERP thing. THERP is really good for understanding how the operator selects things.

But when you've got a process sort of thing, then you need to go to a ruling, because just to design a process, it's not so obvious as you might think it is. It's probably worthwhile chatting with them a little bit, to see what they know that you don't know.

MR. WAGNER: Right, right, right. So as we've discussed, there's -- after you transfer the assemblies, there's a review, but that only reviews the movement of the assemblies, not what terms you chose in the first place.

Next slide. After a few iterations, this is the event tree we ended up with. It's pretty much the same as the process flow diagram, except that there's an independent review which is sometimes performed.

We're a little behind on time, right? So since we talked about a lot of the takeaways from this, I'll just go over it really quickly, in that you can see the sequences where your errors in making the move sheet or transporting the assembly is, you have several checks that can catch you for the error, and actually choosing assemblies.

Once you've made that error, there is, you

know, we have a fault tree under that top event, which has review on it. But once you've made that error, there's no more checks that can really help.

MEMBER ARMIJO: Shouldn't the independent review be at the point where you're choosing the assemblies? In the event trees of the examples you showed us, 64 out of 64 of the misloads were choosing the wrong assemblies.

MR. WAGNER: Right.

MEMBER ARMIJO: And so wouldn't your independent review, shouldn't it focus on the very beginning of the process, rather than a final end?

MR. WAGNER: That would certainly reduce the probability.

MR. BARTO: And that is a recommendation, as part of our additional administrative loading procedures, that you have an independent review.

MEMBER ARMIJO: Yeah, yeah, because I know in my mind, I just took it for granted that the front was really nice and clean and orderly, and that somewhere out in the plant, somebody just picked up the wrong assembly for one reason or another. But that's not the case at all.

MR. WAGNER: Right.

MEMBER ARMIJO: Yeah.

MR. WAGNER: And the way we modeled this is we modeled errors in choosing assemblies as multiple misloads, and errors in transferring them as single misloads, mostly just based off of what we've seen.

So we use spent fuel pool data for the actual transferring of assembly use. There was a Bechtel report that actually looked at misloads within the spent fuel pool, and actually came up with a probability for that. So we used that probability for the actual transferring of the fuel assemblies, and THERP numbers for everything else.

And as we discussed, the dominant sequence by far is the multiple misload sequence, that comes from errors in the initial planning process. Next slide.

So ultimately the conclusion is that misloads are credible. We see them, they happen, and they happen from errors in the planning process and involve multiple assemblies and casks.

Once you've made that error in the planning process, the errors that were seen at some -- you largely kind of lose control of the whole system, and many of the assemblies that you're loading are all just random at that point. So it does seem that they always cause multiple misloads.

CHAIR RYAN: Well, it's really not random. It's a sequence of misloads that are created by a single first error.

MR. WAGNER: Right.

CHAIR RYAN: It's a consequence of the first error. It's not random.

MR. WAGNER: Well, correct.

(Simultaneous speaking.)

MR. WAGNER: Well, it's not right. Not actually the right term.

MEMBER ARMIJO: The characteristics of the misloaded assemblies are more or less random.

CHAIR RYAN: Right.

MR. WAGNER: Or at least unpredictable.

MEMBER ARMIJO: Right.

MR. WAGNER: That was intended.

CHAIR RYAN: Well theoretically, if you knew the first failure, the assembly that was misloaded, you could predict all the rest, because it would follow. So that's not random.

MR. WAGNER: Right, right, right.

CHAIR RYAN: That's a direct result of the first error.

MR. WAGNER: Right.

CHAIR RYAN: Now that's a big deal. To me,

that's a really big deal, because I mean if you find the first one, we've prevented a lot of them.

MR. WAGNER: Right, that's true.

CHAIR RYAN: So I would find the first error.

MR. RAHIMI: Okay. So I guess eventually the analytical method, 10 to the minus 3. So it came out in the actual data, compared to the theoretical, and it was about, you know, one order of magnitude.

CHAIR RYAN: Okay, John.

DR. WAGNER: My name is John Wagner. I'm a group leader of Design Safety Assimilation and Integration Group at Oak Ridge National Laboratory. I'm going to be talking about burnup credit code validation. I want to make sure that I say for the record that the key personnel.

There's been a number of folks that worked on this project; John Scaglione, Don Mueller and B.J. Marshall on the criticality side; Georgetta Radulescu, Ian Gauld and Germina Ilas on the depletion validation side.

I have the challenge, a little bit, of talking about two rather thick NUREG documents, and try to convey the technical information in them within 20 to 30 minutes. So we'll see how well I do on that.

Rather, before I dive into what's in the depletion validation NUREG and what's in the criticality validation NUREG, I want to say a few words about background and purpose, why these were done. Some of that probably will be obvious, but I'll spend at least a couple of minutes on that.

What I would like to see is what I've seen so far, is that ask you questions where things aren't clear enough. So just quickly on the background, the most significant challenge to expanded burnup credit, what I mean by that is additional credit for burnup, has been the validation of the depletion and criticality calculations, and in particular the availability and use of applicable measured data as was mentioned earlier, and especially for fission products.

Applicants and regulatory reviewers have been constrained about the paucity of the data, as well as the lack of clear technical basis or approach or use of the data. As I think Meraj indicated earlier, there's been a number of applications for burnup credit, and I think it's fairly safe to say that each one took a slightly different approach on how they used the data that they had to work with, and how they made their case for their credit for burnup.

Which leads to, you know, protracted review times for those license applications. So that's hopefully something that will help with this work. Just note that it was mentioned already that the rationale for restricting ISG-8 Revision 2 not only was based on limitations in the available validation data at that time.

So the purpose, the charge that we took on with this work is to try to establish technically sound defensible validation approaches for criticality safety evaluations, based on the best available data methods that are available.

And so -- and then to, after developing those methodologies or approaches, to apply those approaches to representative systems. In particular we're talking here about storage and transportation casks. I'll note that in the NUREGs, we also looked at spent fuel pool application 12.

But the focus on everything I'll say here is PWR storage and transportation casks. But apply those to those representative systems, and to demonstrate their usage and applicability, and to provide the reference results which were referred to earlier, and I'll show some of those.

Then of course obviously we need to document

the approach so that for those folks who are -- or those potential applicants who do not choose to use the reference values, that they understand how the approach is to be implemented and used.

So kind of an overview of the depletion validation approach is in this slide, and so the bias uncertainty in the predictive fuel isotopic compositions is developed based on a comparison to measured isotopic compositions from disruptive radiochemical assays.

For those that are not familiar with what that is, these are where actual spent fuel pieces are dissolved, and the individual nuclide concentrations are measured, okay. So that's a key point, that we are preparing what we predict the concentration of a given nuclide could be with our curves, to what has actually been measured.

Then based on that, that gives us a bias and a bias uncertainty in those compositions. What we'd really like to know is how do those biases and uncertainties affect our prediction of k-effective for our safety analysis models.

So we used something called a Monte Carlo uncertainty sampling method to estimate the bias and uncertainty in k-effective, due to the bias and

uncertainty in the isotopic compositions. So you can think about this as sort of propagating that bias and uncertainty from the compositions to the k-effective values.

We also, there's a few different ways to try to do validation for the depletion calculations, and we also did additional analyses using something called the direct difference method, and sensitivity uncertainty techniques, to provide additional confidence in the results we're getting, to evaluate certain aspects of it and so forth, to basically increase our confidence in what we were doing.

So then using the Monte Carlo sampling method and the radiochemical assay data, we determined bias and uncertainty values for representative storage and transport cask configuration, using SCALE 6.1 and ENDF/B-VII data, which is the latest available nuclear data that we have.

Then the sort of last thing that we did in the approach was because we were thinking that we would like to provide some reference validation values for applicant to potentially use, we need to understand how sensitive is our bias and uncertainty values to the various parameters that might be different from our system to another system.

So we did, we evaluated the sensitivity of the bias and uncertainty parameters or values, to things like enrichment and burnup and boron concentration in absorber panels and things like that.

So we picked about, I want say roughly a dozen different parameter variations, and then evaluated the sensitivity of the bias and uncertainty of those parameter variations.

So I have no intention to go through this table. I provided it really more for information and for me to make a couple of points. On the PWR side, we have 100 fuel samples that we used. This number of fuel samples has expanded in the last, particularly in the last five years or so.

We, Oak Ridge and NRC have been very involved in international programs, and that's where some of the more recent and good data has come from. So we've been very in tune to what's going on in the international community.

The other point I wanted to make on this slide is this gives you some indication of the area of applicability of the data, in terms of the assembly design, enrichment and burnup ranges.

This is another cut at the data, and there's a point I would like to make on this, and that is that

say five to ten years ago, most of the radiochemical assay data that we had provided measured values for the principle actinides.

So you'll see on the left-hand side the actinides list a number of samples. There's quite a significant number of measured sample points for the primary actinides. And again, that's part of the reason why ISG-8 Revision 2, you know, recommended credit for actinides.

In recent, at that time, there were a few samples for some of the relevant fission products. But over the last ten years, there's been a rather significant increase in the number of available samples for the fission products, which now enables us to actually make use of those samples and provide a technically defensible validation approach with those samples.

You'll notice that some of them are still relatively few, and how that gets portrayed into the methodology is then fewer samples affects your tolerance factor that is used. So you pay sort of what I call a penalty, due to the uncertainty, due to the relatively few number of samples.

MEMBER ARMIJO: Yeah. I had a question on the analysis for fission products, for volatile

fission products like cesium. You know, in these chemical analyses, is only the fuel particles dissolved and chemically analyzed, or is there effort to dissolve what's deposited on the ID of the cladding? You know, there's quite a bit of cesium that accumulates there.

DR. WAGNER: So this is one of the challenges with doing radiochemical assays, because for one thing, this table of data represents data that's been collected over the course of more than two decades, and as people have done these measurements, they've learned how to do things better.

And so what we see is that, for example, one of the reasons that the rhodium sample is relatively few is that it's not very soluble, and so it can be difficult to measure. So my point in this is you'll see cesium hasn't been measured very much. It certainly wasn't measured in the early, so-called early days.

It's been measured more recently, where these kinds of issues are more understood. But I don't want to give you the impression that these measurement approaches are perfect. There's a lot of room for uncertainty to creep into these measurements, in terms of capturing all the material. So I don't

know if that gives you a satisfactory.

MEMBER ARMIJO: Yeah. I'm just saying my guess is if all you did -- let's say if cesium was really important, the isotope in cesium, you would under-predict the amount of cesium if you just measured the fuel particles, the fuel tube, because quite a bit of cesium is deposited on the ID of the cladding, particularly for high burnup fuel.

And it may or may not be important, but those volatiles are in there some place, and maybe they affect criticality, even though you're not measuring them. They're either on the cladding or in the fuel there. They're there.

DR. WAGNER: Right. So one of the challenges, I'll reiterate. One of the challenges with this approach is the accuracy of the measured samples. So if the measured samples are not accurate, or they are inaccuracies, then we see a spread in the comparisons between calculating experiment values that is due to that.

You know, whereas normally you would think the experimental measurements are perfect and the calculations are wrong and we're comparing to that. So we have the combination of these effects, and what we find is that it manifests itself in terms of a

rather large uncertainty around the bias values.

That's part of the reason that we've gone, I'm getting ahead of myself a little bit, that's part of the reason we've gone to try and address that uncertainty in a more realistic manner. Hence this Monte Carlo uncertainty sampling approach that we've gone to.

MEMBER POWERS: Now I don't understand quite how you do that.

DR. WAGNER: Pardon me?

MEMBER POWERS: I don't understand quite how you do that.

DR. WAGNER: How we do what?

MEMBER POWERS: The Monte Carlo sampling.

DR. WAGNER: I'll get to that.

MEMBER POWERS: In the face of systematic error.

DR. WAGNER: You have probably systematic error in one sample, in set of measurements. But what we're actually dealing with is a number of sets of measurements.

MEMBER POWERS: Not to what he's talking about. I mean if you take his example, and what's a good example? A good example is a pandemic, systemic error.

Say you have an accumulation of cesium in the ID of the cladding as you will, every analyst will make that error and it will be in the scatter, because the simple matter of that cladding ID material will adhere to the fuel particle, and that will depend on how you extract the fuel from the cladding to give you the dissolution.

But it's systematic error. So how do you pick the distribution that you're going to use for your Monte Carlo sampling?

DR. WAGNER: So one of the things that we looked -- first of all, regarding the dissolution, the cladding is put in the acid as well. So --

MEMBER ARMIJO: Actually, you slice the whole fuel rod and you take that sample of cladding plus fuel, and dissolve it all.

DR. WAGNER: Right, right, and obviously the cladding doesn't dissolve, but the surface things will dissolve.

MEMBER ARMIJO: Right.

DR. WAGNER: Actually, things that typically won't come out of that as well are things where heavy kind of projectile actinides actually, are embedded in the deep end. That's a very small effect, but that's more of the thing that would not get dissolved out,

and it would have to be kind of mechanically taken out of it. So that's kind of that issue.

So the point that you're getting at, I believe, is are these, is this a -- is the distribution of the uncertainty, is it a normal distribution, which would mean it's, you know, it doesn't have a systematic kind of behavior, or is it a systematic uncertainty that we have to deal with?

And so that's part of where we calculate a bias, an uncertainty on that bias. We look at the distribution of that uncertainty around that bias, so that we can determine how to sample that distribution in our Monte Carlo uncertainty approach. Should we use a normal distribution or should we use something else?

MEMBER POWERS: Manifestly it's not normal.

DR. WAGNER: Right, right, and so we do --

MEMBER POWERS: It could not possibly be normal.

DR. WAGNER: Right. So we do have some isotopes that were non-normal, and so that --

MEMBER POWERS: None of them could be normal.

DR. WAGNER: A fair number of them were normal.

MEMBER POWERS: Never, absolutely could not be, because the normal distribution goes out to infinity on both of them, and you never have negative values.

DR. WAGNER: You have normality tests, and for a fair number of isotopes, they passed the standard normality test.

MEMBER POWERS: Doubt it.

(Laughter.)

DR. WAGNER: I don't know what else to say on that. We have the data. I mean I can provide the data. We can run our favorite normality test and look at that.

MR. RAHIMI: But also I do want to add, if the question touches on -- I mean Dr. Armijo had a question on okay, the cesium deposit on the cladding. If you're just going after the fuel, the cesium in the fuel, and your question is that is a systematic error.

If all the time you're looking at the fuel you're missing. But normally these samples, you know, are done at a month elapse, and that's different techniques each lapse uses, and an average is used.

So there's that sort of a combination, one technique, one lab is not used, you know. I don't know specifically, John, that the cesium-137, we've

got only seven samples, if those samples were done in your data or data --

(Simultaneous speaking.)

MEMBER ARMIJO: Yeah, but you know, I think they tend to be conservative. The thing I'm raising, the issue I'm raising is you measure less cesium than is actually there. It's more from the criticality standpoint it is there.

DR. WAGNER: It is there.

MEMBER ARMIJO: And so it's a conservative error. But I just wondered how it, what actually got measured, whether it was the entire slice of the fuel rod at a particular location, including the cladding, all of it's resolved, and then you do the isotopics.

MR. RAHIMI: Yeah, we can look into that. Yes, we can ask. Dionne is at Oak Ridge. He's the --

(Simultaneous speaking.)

DR. WAGNER: I mean actually we already have. We've looked at each laboratory, how the measurements were done, what their stated uncertainties are, what we believe their uncertainties to be based on evaluation of the data. It's all documented.

MR. RAHIMI: But the question is when they measure the cesium, is it just from the fuel, or is it

from dissolved cladding and cesium? I mean that's the question.

(Simultaneous speaking.)

MEMBER ARMIJO: You've got all of it. So you've got --

DR. WAGNER: You can never say whether you've got all of it. I mean you can't say that for sure. But certainly the cladding in the fuel was put in those, and dissolved in the acid together, and then the cladding is --

MEMBER ARMIJO: Then you've got pretty much all of it.

DR. WAGNER: I would think so. We can go back -- one thing we can do is go back and look at the C over E value specifically for cesium, for example.

MEMBER ARMIJO: --for any of the volatiles, cesium, iodine, cadmium, those kind of things.

DR. WAGNER: You see, iodine you don't credit because of, you know, because of its volatility.

MEMBER ARMIJO: Okay.

DR. WAGNER: Okay.

CHAIR RYAN: So we're scheduled for a break, and I wanted to offer the choice of maybe having a 15 minute break now and having you come back, or do you

want to finish first?

DR. WAGNER: I'll leave that to you guys.

CHAIR RYAN: So why don't you cover it in a few minutes? I don't want to, you know, have a break ten minutes before the break, ten minutes before the break comes up.

DR. WAGNER: Okay.

CHAIR RYAN: So my suggestion would be if this is a good stopping point or break point, we'll take a 15 minute relief break and come back promptly at 10:15. Is that fair enough?

MR. RAHIMI: Yes, sure.

(Whereupon, the above-entitled matter went off the record at 9:59 a.m. and resumed at 10:13 a.m.)

CHAIR RYAN: Dr. Wagner, we left off with you, please.

DR. WAGNER: Okay, all right. Shall I proceed? So I was talking about the Monte Carlo uncertainty sampling method. In the interest of time, I will try to go through this pretty quickly. I've been told that you all have copies of the reports, and so --

CHAIR RYAN: We do.

DR. WAGNER: Bear with me as I try to go through relatively quickly. I've already mentioned

that this uncertainty sampling method is used to propagate the isotopic bias uncertainty to k-effective. It provides a realistic safety margin or bias uncertainty, because it allows for compensating positive and negative effects, the distribution of the sample.

It enables depletion code validation directly on the safety analysis model. This is kind of in reference to other methods, where you have to -- they can't properly address things like axial or radial burnup distributions.

You apply this method directly on the safety analysis model that you want to demonstrate a bias uncertainty for. The other nice aspect of this method that was a reason for choosing it was that it's not sensitive to the limited number of nuclides measured in the individual fuel samples.

You say okay, well what does that mean? It means that one lab may have measured 12 of the isotopes that we care about. Another lab, another sample, may have measured 28. So there's inconsistencies in the number of isotopes of the different ones measured.

Now this doesn't affect this method, because from that, we can still get a bias and a bias

uncertainty and use it directly. For other methods, it presents a challenge. So the method is illustrated schematically in this. I understand this is busy, but I think it kind of portrays it in one slide how it all works.

You actually do calculations for the individual radiochemical assay samples to do your best estimate prediction of the isotopic compositions. From that, you get a bias. You get a bias and uncertainty from that for each nuclide.

Then for your safety analysis model, you calculate a nuclide concentration, and you from the bias uncertainty you sample randomly or according to the proper distribution, the -- how far off the mean bias that you are. Then based on this formula, you adjust your calculated nuclide concentration by the bias and a representation from the Monte Carlo sampling of the uncertainty in that bias, which gives you an adjusted nuclide concentration.

You do that for each nuclide within the model, and then you put those adjusted concentrations into a k-effective calculation. You calculate k-effective with those adjusted isotopes, and you repeat that process for a statistically significant number of criticality calculations.

For the calculations that we did, we typically did 500 criticality calculations, to ensure that we had a reliable mean and a reliable standard deviation estimate, from which we can calculate bias and bias uncertainty based on the upper limit for one side of the tolerance interval for 95 percent of the population and 95 percent confidence.

That's how we get the bias and bias uncertainty values. Here's some examples of those bias and bias uncertainty values. Now I should note that in the cases that we looked at, we found that the bias actually was positive, meaning that we were over-predicting k-effective, and you don't take credit for positive bias in criticality safety applications, just as a standard practice.

So these values, although the bias and bias uncertainty values together, the bias is zero. So it really is the bias uncertainty values that result, because the bias is set to zero.

So this shows as a function of burnup the bias or bias uncertainty values for actinide-only cases, and for actinide and fission product cases. You see generally speaking, these numbers vary between one and a half and three percent in delta k over the range of five gigawatt-days to 60 gigawatt-days.

These are numbers directly out of the ISG-8 Revision 3.

So then from this, we had recommendations that came out of -- into the ISG. I mean there's a whole lot of information in these reports, by the way, so I'm just touching it really. But from those numbers then, we worked with the NRC and came up with recommendations on how to use this information.

So you're probably all familiar with this, so I don't need to go to every detail of it. But the general main points are that there are several isotopic depletion validation methodologies that are recommended as applicants can use. That's stated in the ISG.

If they, if an applicant does not wish to go through that process, and you know, there is a fair bit of work in that, then they can use the values that are provided in that NUREG, which are also echoed in the ISG, provided they meet certain criteria.

I think Meraj spoke of this criteria earlier this morning. The main points being same code and cross-section library in similar kind of situations to what was evaluated in the NUREG document. Those values, by the way, correspond to these listed actinides and fission products.

So moving on to the criticality validation side, I'll pause for a minute in case there's questions on the isotopic. I start out with what's the challenge on the criticality side?

On the isotopic side, it was the number of data points in measured data and the issues with the measured data, how to use it. On the criticality side, the primary issue was that the existing or available laboratory-critical experiments.

These are experiments with fuel that was taken critical, gives us a reference k-effective, experimental k-effective value, and then we calculate k-effective on that configuration with our codes, and we look at the difference between the two, and that's how we develop a bias and uncertainty. It's sort of the traditional approach.

A key point in that is that those critical experiments need to be representative of the system that you're trying to show that your codes predict accurately for, so-called applicability of the critical experiments to your safety application.

But the challenge has been that these laboratory-critical experiments that are available do not have the minor actinides and fission products in the proportions that are actually similar to actual

spent nuclear fuel, and hence are not directly used in a traditional way, cannot be directly used. So that's been our challenge.

Now part of that challenge has been mitigated, at least for the primary actinides, through evaluation of the laboratory criticals that do have uranium and plutonium, identifying which ones are applicable and actually neutronically similar to spent fuel, and then the access to the available French HTC critical experiments that was mentioned earlier.

It's a very nice set. There's 156 experiments there, so a very significant number. It was specifically designed for this purpose.

And related to that, there was a question about us coordinating with the French program. I'd just note that primarily what we did there is we purchased access to that data, because they spent quite a bit of money obviously on those experiments. So the U.S. purchased access, and have made that data available to cask licensees.

So that remains then the challenge. The focus has been squarely, and really the focus of this NUREG is focused squarely on how to validate the fission products and the minor actinides, where we simply don't have much in the way of laboratory-

critical experiments.

Now we do have some. There are a few laboratory-critical experiments that include a couple of the fission products in the international handbook of criticality benchmarks, and there are -- the French did a series of fission product criticals as well that we had access to.

So we do have some. So we looked at those. Then, but again, they're not in the right proportions and there are issues, and they're relatively few in the number of samples. So then we kind of stepped back and looked at okay, how do we move forward in this area, given that we are not going to get the perfect fission product experiments?

We've been using sensitivity uncertainty analyses quite a bit at Oak Ridge, and looking at neutronics similarity and understanding biases. And through this process, we've also looked at the fact that a lot of our biases in criticality calculations or the real source of the bias in our criticality calculations is errors in nuclear data.

And so we looked at using nuclear data uncertainties to estimate potential biases for given relevant nuclides, and we have some tools that can do that. I'll talk about that in a minute.

Then the next question is okay, well you can do that, but how do you know if it's right or reliable or directly useful for coming up with a bias for the fission product? How do you defend that, okay?

So what we did there was we tried to verify these estimated biases to comparisons with cases where we actually have laboratory-critical experiment data, and can calculate a bias in a traditional means. So we're comparing that to what we get from this other approach.

We also compared in that process, including cases where we do have some limited fission product criticals. So we compared what we get for a bias from those with what we predict, based on nuclear data uncertainties.

So I talked a little bit about how we estimate the bias based on nuclear data uncertainties. Certainly in a system, k-effective can be propagated from the cross-section uncertainty through a sensitive coefficient.

So what a sensitivity coefficient is, I mean I apologize. You can spend a lot of time talking about this, so I'll try to stay at a high level. Sensitivity coefficient gives us the sensitivity of k-effective to change in a cross-section, okay.

Now if we fold that with the uncertainty in a cross-section, then we get the sensitivity of k to that uncertainty in that cross-section. So that's what we're after here, is what's the uncertainty in the k-effective value due to uncertainties in the nuclear data, and we can get this on a nuclide by nuclide basis.

Actually, we can get it on a reaction by reaction basis, if you want it. We're actually interested in it on a nuclide by nuclide basis.

So the fundamental basis for this approach is, again that I mentioned, is that the biases are caused by the nuclear data errors. The biases are caused by other things. This doesn't, you know, we're not dealing with that. So that's an issue.

Now what we've also seen, though, is that the biases are bounded by the nuclear data uncertainties. The point I'm trying to make there is that the key point in this is are the nuclear data uncertainties accurate. If they're underestimated, then you could underestimate the bias that you're predicting from this.

And what we've observed over the years is that the nuclear data uncertainties are actually too large, all right. So they were overestimating the

bias. But we still wanted to go through and do comparisons to check that assertion.

So that in that last point, uncertainties give us an upper bound for the magnitude of the bias. So here's just a couple of examples. We have lots of them. But here's a couple of them, where we confirm that the computational bias is generally bounded by the cross-section uncertainty.

So let me explain what these charts mean quickly. These black dots are actually C over E ratios for given laboratory-critical experiments, and each one of the charts represents a set of experiments. This is not nomenclature I would expect everybody to know.

But for example, this is high enriched uranium metal fast-spectrum systems, and so on and so forth. So we looked at classes of systems, and again, going back to the plots themselves, these black things are the C over E ratio.

So this is the bias that we see on an individual experiment and our ability to predict it. The green dots are the experimental uncertainties or the listed experimental uncertainties, and you know, as they're estimated, and then the brownish bars represent the uncertainty in the k-effective that we

predict, based just on the nuclear data uncertainties.

So in looking at, you know, literally hundreds of these, where we do have good laboratory-critical experiments, we've shown that by using the cross-section uncertainty, we come up with a larger value and a consistent value with what we get from the critical experiments were we have them.

That's important, because now we're going to say that we're going to use those where we don't have good laboratory-critical experiments, or where we have laboratory-critical experiments that are inadequate in various ways.

Now in this chart, I'm only showing you experiments that don't have fission products in there. We also did do comparisons with cases with like the French fission product criticals, in the few cases where we have fission product criticals, and show that we were getting consistent answers.

It wasn't a perfect comparison, because again that data is so limited in numbers of samples, that it's in proportion of fission products, that there's a fair bit of uncertainty in the use of those.

So to give you some example, then, of results, I just have, you know, very kind of high level numbers here to give you some perspective on

this. So this is for -- these tables are for spent fuel pool high density rack-type application and the 32 assembly PWR cask model.

This table shows you the uncertainty in terms of percent delta k over k, for different kind of classes of nuclides. You'll see that for actinides only, we're showing about a 500 PCM uncertainty due to nuclear data uncertainties.

For those who are familiar with these, that's actually pretty consistent with the bias we see for LEU actinide-only types of applications. To put things in perspective then, now when you look at the fission products, we're seeing, you know, on the order of 20 to 50 kind of PCM.

So rather small numbers as a result of the nuclear data uncertainties in the fission products and the minor actinides, particularly in perspective to the actinides. It goes back to, you know, 75 percent of the credit is due to the actinides.

CHAIR RYAN: The largest difference is in structural materials. Why?

DR. WAGNER: Pardon me?

CHAIR RYAN: A larger difference is in structural materials. Why?

DR. WAGNER: It's actually due, and then

this was actually a good outcome of this work too, in terms of kind of shining the light on that a little bit, and so you picked up on that. But a lot of the structural materials, like the chromiums, the nickels, the irons, they actually have a lot of resonance behavior.

Now most of it's not in the thermal range, but because of that, there's actually a fair bit of nuclear data uncertainties in those cross-sections.

CHAIR RYAN: Okay, thank you.

DR. WAGNER: And that's what we're seeing. So that's another thing. While we focused a lot on validating the fission products, kind of it draws us back to going we've got to make sure we've got our structural materials well-represented in these critical experiments as well.

CHAIR RYAN: So that's really over a range of different kinds of structural materials in different settings. What's why the difference?

DR. WAGNER: Yes.

CHAIR RYAN: Thanks.

DR. WAGNER: So this talk is fairly high level, but the uncertainties due to the nuclear data uncertainties were investigated for special

configurations as a function of burnup and a variety of other relevant parameters.

The point I'm trying to make here is that just like for the isotopic validation, for the criticality validation, we also looked at the sensitivity of our predictions to different ranges of relevant parameters. For like -- for example, the boron content in the cask and things like that.

In all cases, and there's a lot, there's several big tables in the NUREG. We found that the uncertainty due to nuclear data uncertainties was less than 1.5 percent of the reactivity worth of the actinides that we considered.

So this was important, because the question becomes how do we use this information? How does an applicant use this information? We say that the nuclear data uncertainty in our configuration is .1 percent. But how does somebody else take that number and use that in a manner that's more general?

And so what we did is we characterized this in terms of the reactivity worth to the isotopes that they're trying to credit, and show that in all cases again, 1.5 percent of that worth was bounding what we saw.

So that enabled us to -- well, it will be in

a minute, to have a general type of recommendation for others to use. So the recommendations that are in the ISG-8 are that you validate -- basically, you validate everything you can, to the extent you can, with the critical experiments that you have.

If we have perfect critical experiments, I would recommend that you use those, even for the fission products. But we don't. We do have them for the actinides. The actinides are the principle component that we need to be careful, that we need to be concerned about.

So we recommend that you validate the principle actinides with laboratory-critical experiment data, and we specifically focus on the HTC experiments. Our NUREGs have identified a number of MOX criticals in the international handbook that are applicable.

Then for the nuclides that you can't validate through laboratory-critical experiments, we suggest a conservative estimate of the combined bias and bias uncertainty associated with those nuclides to be 1.5 percent of their worth, of their work.

This estimate is appropriate provided the applicant, you know, just like the actinides, or just like the isotopic validation, provided they do similar

things to make that number consistent with what they're doing.

We've since been looking at how can we expand that out to other codes, and particular other codes. So it's right in the version of the ISG-8. Drew will talk in a minute about some of the comments back on the ISG-8 and NRC's responses to those.

What's in the ISG-8 right now is that you have to use the SCALE code, with either five, six or seven cross-section libraries. If you use a different code, you have to use a higher bounding. Again, I'll let Drew talk about what the thinking is on maybe changing that.

Then similar initial assumptions, similar cask models. And we capped the ability to use this at ten percent k-effective, and for those of you who don't do this kind of stuff every day, the fission products and minor actinide credit increases as a function of burnup, and just kind of a realistic rule of thumb, be about three percent for low burnup, up to about ten percent for high burnups in the 50 to 60 gigawatt-day kind of range.

So that's, you know, kind of where you would expect the fission products and actinide worth to be about, in maybe a 60 gigawatt-day kind of range. And

that's my summary of those two NUREGs in a very quick period of time.

I'd like to acknowledge the NRC support for all this work. It's been quite rewarding, in the sense that this is the major technical issue I think we've been wrestling with for a long period of time, and I think we've got adequate resolution on this with these NUREGs.

CHAIR RYAN: Thank you, John. I appreciate that very much. Steve, where are we? Ah, let's see. Drew, we're back to you.

MR. BARTO: Yes. Unless there's any other questions or comments on John's presentations. Thank you. I wanted to talk at a reasonably high level about the public comments we received so far on the draft ISG.

We got three sets of comments, one from NEI, another from Stefan Anton out of Holtec International, and another from Dale Lancaster from Nuclear Consultants dot com.

I want to discuss the major comments in our proposed resolutions. I don't intend to go through every comment we got, but just sort of the --

CHAIR RYAN: That's good.

MR. BARTO: Just sort of what we perceive to

be the major ones, and any of the commenters that are here, if you don't see it on this list, then that means we probably agreed with it and are going to incorporate it into the ISG.

The one major comment that we got from, I believe it was only from NEI, and it was a recommendation, probably a question more than anything about why isn't this material in a Reg Guide as opposed to an ISG, and our intention is really that we believe this guidance needs to be consolidated into one place for staff, as opposed to having some of it in an SRP and then referencing a Reg Guide, which would further reference these NUREGs.

So for the time being, we intend to have it as an ISG, and at the next revision of the SRP, to incorporate in the whole, into the SRP. The thinking is that this ISG format allows more flexibility to modify it in the future, and although this does represent a significant step in burnup credit, we do anticipate changes coming down the line.

I think EPRI is going to discuss how they have a different validation methodology that they're proposing and that we may, in some form, incorporate into our guidance in the future. The other big piece that's coming down the road is BWR burnup credit.

We're eventually going to have recommendations for that that we'd like to incorporate into this.

So we leave open the option to consolidate this guidance into a Reg Guide at a later time. But for the time being, we're going to leave it as an ISG that will be incorporated into the SRPs.

I already mentioned the EPRI methodology that Albert is going to discuss. There was a recommendation in the comments that we explicitly point out in the ISG that there are alternative methodologies.

We will likely include text in the ISG that reinforces that, but the whole concept of the ISG is that this is one methodology that the staff has reviewed and found to be acceptable, and it's one path that we're identifying as sort of the path of least resistance to get an approval.

It by no means excludes all other methodologies, and any other methodology that would come in would be reviewed on a case by case basis. So we will reinforce that in the ISG.

We talked already at some length about misloads, and as I stated, we previously only had a recommendation to do a confirmatory burnup measurement to prevent misload, and we've since revised the ISG to

include an alternative to that, which is performing a misload analysis, accompanied with additional administrative loading procedures.

We've got -- we received one comment that suggested that we simply remove the burnup measurement option. We are going to leave that in the ISG as an option, you know. We believe most applicants will probably choose the path of doing the misload analysis and incorporating these additional procedures.

But there is always the potential that even for some systems with the misload analysis, you may not be able to demonstrate adequate subcriticality, in which case measurement might be a better option.

And also, thinking into the future, you know right now, I believe most applicants and utilities would find the measurement techniques that are available to be burdensome, out of core, in the pool with additional equipment.

But there's always the possibility in the future that there's an easier measurement technique that may make that option more appealing than doing the misload analysis. So the bottom line is we're going to leave it in there as an option.

We got a good deal of comments on our recommended loading procedures, that are intended to

accompany a misload analysis, and there was -- at least in one set of comments, there was a -- they proposed taking out the ones that we had recommended and putting in a completely different set.

We looked at those that had been proposed, and our view of what had been proposed is that those procedures were things that should already be done for cask loading procedures, even if you're not considering a burnup credit. There's a couple of examples listed on this slide, you know, verify the identity of the fuel assembly prior to loading, you know, reverify them prior to closing the cask, and you know, all of the lists.

I've got a backup slide that has the complete list on it, but it's all sort of things like this that you would expect to be done already, and that we routinely see incorporated into cask operating procedures for casks that don't include burnup credit.

So our view of the administrative loading procedures is that they should be additional procedures for a burnup credit cask, targeted at reducing the likelihood or consequences of high reactivity misload.

Again, these are just a couple of examples of things that we've recommended, such as assuring

that you have no fresh fuel in the pool during system loading, since we've not considered fresh fuel as a misload possibility.

Verification of the location of high reactivity fuel, both prior to and after loading, and then independent third party verification of the loading process, as we've already discussed a little bit. And again, this is just a sampling of I believe six or seven recommended procedures that are in the direct ISG --

CHAIR RYAN: Drew, just to kind of ask a question about that.

MR. BARTO: Sure.

CHAIR RYAN: Severely underburned fuel, what handbook do I look up what that is? I'm trying to think. Are you looking at a burnup number or is there a better way to express that, or does everybody understand severely underburned fuel?

MR. BARTO: I don't think there's any official definition of it anywhere, but you know, as we've -- if you go back to what I was discussing earlier about the misload analysis criteria that we've recommended, we came up with somewhat of an ad hoc definition of a severely underburned assembly, and it's an assembly that would bound 95 percent of the

underburned fuel population, with 95 percent confidence in terms of reactivity.

CHAIR RYAN: Ahh. So in terms of reactivity is the metric?

MR. BARTO: Right. So there's no burnup value, but it would be -- you know, for a specific cask system, it would be a burnup value that would vary with initial enrichment.

CHAIR RYAN: I think maybe it's just me, but I mean telling a little bit more about what it is you mean by that when you define this, or you should define it, would be helpful. Because you know, whenever there's a qualitative statement like that, you're going to get a lot of opinions about what it means sooner or later.

Even if you trained everybody on the first day, they'll go no, no, let's do it this way, because it is severely underburned, and I think this is a better way to express it. So I just caution that, you know, a little bit more analytical view of it might be helpful. Those are just my thoughts.

MR. RAHIMI: Yeah, yeah. Well, we'll do it. Actually, maybe in the parenthetical statement, we should have said "i.e., 95 percent of the underburned."

CHAIR RYAN: Or see this report, which tells you more about it, or however you want to do it.

MR. RAHIMI: That's right.

CHAIR RYAN: But just to say "severely underburned fuel" and not point me to the right understanding of that would be a missed opportunity.

MR. RAHIMI: Yeah.

CHAIR RYAN: Okay.

MEMBER SKILLMAN: It seems back on Slide 53, the several bullets that you've identified are the ones that are the classical way people think about loading casks. But the events that you pointed to are not those types of events.

MR. BARTO: Right.

MEMBER SKILLMAN: The events that you pointed to are failures to understand either load or some other parameter. So as I look at your Slide 54, it doesn't appear as though you've pinpointed that systematic error or that process error as part of your recommendation.

Is it your intention to make sure that the recommendation that points to either the process or systemic error, or the events from Palisades and McGuire and Grand Gulf?

MR. BARTO: I think the third sub-bullet

there under the first bullet is intended to get at that, you know, independent verification of the loading process to include selection of fuels at least.

MR. RAHIMI: Yeah, and I think that is a point well-taken, that it goes back to earlier discussion, that we've got to define that verification going all the way back to the beginning of the process.

(Simultaneous speaking.)

CHAIR RYAN: --stuff on the diagram, that's where it all -- that's where the action is.

MR. RAHIMI: Yes.

CHAIR RYAN: But nothing -- I mean I'm not sure it does, but that's where maybe the root problem is. So that to me is kind of the takeaway message.

MR. BARTO: And again, I think, you know, we want to include these procedures, because we believe that, you know, particularly this one that I'm talking about, will reduce the probability. However, there's a lot of room to make them not credible. We're down in the 10 to the minus 2 range.

So I think regardless of what you do, you're looking at a situation where there are going to be misloads. So our view has been analyze for them or

measure each fuel assembly to show it's the one that you think it is and demonstrate that you're not going to have a criticality issue, even if you have one.

CHAIR RYAN: Even if you have a misload.

MR. BARTO: Right.

CHAIR RYAN: Yeah.

MR. BARTO: So we can certainly, you know, as you say, pull the string on it a little further and see what we can do to further reduce that --

CHAIR RYAN: Yeah. Maybe I'm thinking ahead and shouldn't be speculating at all, but it seems to me that this kind of a topic is ripe for an appendix, to walk people through it, as we can go wrong at this step and all the way through and kind of road map a little bit, so you give people the insights as to what they have to be looking for. It was just a thought.

MR. BARTO: Okay, that's good. A little takeaway. So I guess the bottom line with the administrative loading procedures is we're going to not adopt the list that we got as a comment. However, there were significant comments on each of the individual, recommended individual procedures, and we will end up revising that, and I'll talk about that right now.

We, in our list of recommended procedures,

we had two that got at fresh fuel. So we had one that said ensure that there's no fresh fuel in the pool at time of loading, and then an additional one that required qualitative verification of burnup, this --

It's not stated explicitly in the ISG, but this basically means look at them, you know, because they are so visually different.

A point was made in the comments that these are essentially redundant, and although fresh fuel is not routinely in the pool during loading, we have had some discussions with licensees and vendors that have stated that there are situations where fresh fuel could be in the pool.

So we intend to revise that to be a single recommended administrative procedure that would give a cask user the option, you know, either of demonstrating you don't have it or incorporate a step in your loading procedure, where you verify that you're not putting fresh fuel in the cask.

And you know, this has to somewhat recognize that there's a tremendous financial incentive to not putting fresh fuel in a -- sealing it inside a storage casket. So it's, I think the staff does not think that this is going to come up basically.

And then that's what steered us towards our,

you know, rather than looking at fresh fuel, looking at severely underburned fuel in the misload analysis.

There was another recommendation for a full pool audit within one year of loading, as an administrative procedure. One of the commenters pointed out that this is -- this overlaps in a 10 C.F.R. 74 MC&A requirements, and we might be sort of adding additional requirements on top of that.

And it was somewhat duplicative of another administrative procedure that we recommended, where you verify the location of high reactivity assemblies, both prior to and after loading. So I think --

CHAIR RYAN: Well how would you verify the already-loaded canisters --

MR. BARTO: This is for -- I mean you put a cask -- before you put the cask in the pool, you make sure you know where your high reactivity fuel is.

CHAIR RYAN: Yeah. Oh, okay, all right. Sorry.

MR. BARTO: And then, you know, you go through the whole loading procedure. Before you seal it up and take it out of the pool, you make sure they're still there.

CHAIR RYAN: Got it.

MR. BARTO: And the idea is that it should

be a small number of assemblies.

CHAIR RYAN: Got you.

MR. BARTO: So how we intend to revise this particular recommendation is to turn it into a QA audit of already-loaded canisters. As we've stated before, there's upwards of 1,500 systems already sealed and loaded, sitting on pads that most of which are going to require burnup credit.

So the question becomes how do you verify that you don't have misloads in those? So prior to transport, there would have to a rule recommending that there was a QA audit to determine that those have been properly loaded.

And there were a number of other minor comments on, or what we consider to be minor comments on the administrative procedures that we recommended. So we're going to do our best to clarify the intent of those. I'm not going to discuss them specifically now.

CHAIR RYAN: Okay.

MR. BARTO: We got a number of comments on our misload analysis recommendations. We had some language about the administrative margin for the misload analysis. It was similar to language that we found in other regulated areas for criticality safety,

where you use a reduced administrative margin, where the .02 is basically the minimum administrative margin.

But any margin less than .05 would have to be justified. So the question is was -- what do we consider to be properly justified, and we are still sort of internally iterating on that.

We want to go back and look at the other areas where we found this language, and discuss what it is that's being sought for justification for administrative margin. So we still have some work to do on that.

There was a couple of recommendations that we got in the comments for different misload analysis criteria, and the theme of these comments is that this, what's being proposed would be simpler, and we agree. It would be simpler, but it might be overly-restrictive.

For instance, for the single high reactivity misload, the recommendation was make it a single fresh fuel assembly. If a vendor were to come in and with that sort of criticality analysis for the misload, that would obviously be acceptable.

But we know from our previous misload consequence work that a single five weight fresh fuel

assembly can easily overwhelm your administrative margin. You can get as much as a five and a half percent in k-effective increase from a single fresh fuel assembly in a burnup cask.

So it's probably not going to be doable for most vendors, and you know, we've already discussed that we've got procedures that would, that we believe would mostly eliminate any possibility of loading a fresh fuel assembly, and that's why we concentrated on the burned, yet still relatively reactive assemblies that we know exist in some spent fuel pools.

MR. RAHIMI: But however, you know, if the vendor wants to do a fresh fuel misload analysis, they're more than welcome, you know. That's the comment, you know, why not do a fresh fuel misload? By all means. Yeah, that is acceptable. That's a lot more conservative.

CHAIR RYAN: Okay.

MR. BARTO: And then for the multiple assembly misload, the recommendation that we received is that, you know, rather than determining this 90 percent threshold for the total discharged fuel population, why don't we do something that's simpler, which is 25 percent, assume the burnup is 25 percent below what is required for half, half the assemblies

in the cask.

And again, we agree that this is simpler and in some cases is more restrictive. However, it really depends on loading curve. If you've got a loading curve that only permits say 20 or 25 percent of the fuel population to be loaded, going to a 25 percent underburn criteria is not as restrictive as the 90 percent criteria that is in the draft ISG.

So I think, you know, we're intending on leaving this recommendation for multiple assemblies, burned to a level that bounds 90 percent of the total inventory in place, although we will, you know, we will obviously consider other, you know, alternative criteria as they're submitted to us.

And then one thing that the 90 percent criteria allows for is that it doesn't -- that 90 percent, the line doesn't move depending on the cask design. It's dependent entirely on the discharged fuel population.

So it's conceivable that you could design a cask where the loading curve already encompasses 90 percent of the fuel, in which case you might not have to perform that analysis.

CHAIR RYAN: Right.

MR. BARTO: There were a number of other

comments that I'll try to run through reasonably quickly. There was a request for some language about how to credit additional isotopes beyond the 28 that we recommend. So we're intending on modifying the ISG to state that you may credit those, provided that there's quantification of the bias and bias uncertainty associated with those nuclides.

There was a recommendation that we, or comment that we should somehow address BWR burnup credit in the ISG. Again, as we've already discussed, we've got a research project upcoming on BWR burnup credit, and we'll have, likely have recommendations on that in the future.

Right now, though, we don't have -- we haven't settled on explicit recommendations for BWR burnup credit. But we will revise the ISG to state that we'll look at those on a case-by-case basis.

We did not, ISG Rev 2 has an applicability section that states that it's only applicable intact fuel, which was our recommendation at the time. We didn't change that applicability section moving forward with this draft, but we have already looked at burnup credit analyses that consider damaged fuel and other variations that are less than damaged.

So we're going to revise this section to

basically incorporate what we've done so far, which is basically -- I mean basically that you may do it, but you need to consider fuel reconfiguration and any other uncertainties associated with it.

MEMBER ARMIJO: As long as you have maintained the geometry of the assembly. For example, a pinhole failure in the cladding would be, is that considered damage?

MR. BARTO: Pinhole leaks and hairline cracks are not considered damaged. We have another --

MEMBER ARMIJO: Fractures of the fuel rod or missing sections of the fuel or something like that. It has to be pretty gross then.

MR. BARTO: Well, I think the way we're going to revise the ISG is that as long as you consider fuel reconfiguration in a bounding fashion, then you can use burnup credit even for damaged fuel, or severely damaged fuel.

MR. RAHIMI: Because the basis of all these isotopes that we are recommending, they had to pass three criterias. They have to be non-gaseous, non-volatile, stable. So at this point, we don't see anything that make applying burnup credit to reconfigured fuel or damaged fuel invalid.

So unless, you know, we can think of

something. But at this point, we can't think of anything that would make it invalid for damaged fuel.

CHAIR RYAN: Okay.

MR. BARTO: We got a comment about how we apply bias and bias uncertainty terms, both for the depletion code and the criticality code. The way we've reported bias and bias uncertainty in the ISG currently is to combine them, and as John stated, for the ENDF/B-VII data, it's not so much of an issue, since the bias term is zero.

But we've sort of reported it as a combined bias and bias uncertainty that you would add directly to your calculating k-effective, and our commenters have, we believe, correctly pointed out that that's not entirely, an entirely accurate way to do, and that your uncertainty terms are typically statistically combined with each other.

So we're going to revise the ISG to state that this, at least for the depletion code, the uncertainty that's -- the bias uncertainty that's the reference value that's reported there may be statistically combined with the other calculation uncertainties.

However, the criticality code uncertainty, this delta k sub-x term, is -- we're going to treat

slightly different. It is an uncertainty in k-effective due to uncertainty in the minor actinide and the fission product cross-section data.

However, we have no information on for minor actinides and fission products to determine a bias, how you would routinely do that in criticality safety space. So what we're saying is that this uncertainty that's calculated is basically bounds the bias.

So we don't know what the bias is, but it may be as big as this term. So we are conservatively treating it as a bias that will be added directly to the calculated k-effective, and we will clarify that in the ISG.

CHAIR RYAN: Good.

MR. BARTO: The last comment that I'm going to discuss is the reference values in the NUREGs are reported for the SCALE code system with ENDF/B-V, -VI or -VII data for criticality, and the recommendation is that that delta k sub-x term should be one and a half percent of the minor actinide and fission product worth for the SCALE code system.

Currently, the recommendation is that you would double that number for other code systems. We had a lot of discussion with our colleagues at Oak Ridge about what's appropriate to do for other codes,

keeping in mind that when we say "other codes" that use ENDF/B-V, -VI or -VII data, we're talking almost exclusively about MCNP.

So the thought was that with a code that's as well-qualified as MCNP, there shouldn't really be a significant difference in the bias, as long as it's using that same data set. However, we had some questions about how each code handles the cross-section data, and the thought was there might be --

CHAIR RYAN: Wasn't there some calculational assessment of that, whether they should be the same or not?

MR. BARTO: That's what we're going to recommend. Currently, we just say double it and don't do anything. I think it's been suggested in a couple of the comment sets, that we provide an option to do a worse comparison.

So you calculate the minor actinide and fission product worth with the SCALE code system, calculate it again with the MCNP system and compare the results.

If they're using data in the same way, you should get very similar answers. So as long as the worths are comparable, then you may use the same one and a half percent number for delta k sub-x.

CHAIR RYAN: That's right. Okay.

MR. BARTO: I guess in conclusion, we're extending the ethical basis for burnup credit for fission products and minor actinides, providing an alternative to the confirmatory burnup measurement that was in ISG Rev 2.

The ISG has been generally well-received by industry, with some comments as I've just discussed. So we will, our next step is to resolve the comments into a final ISG, and present this to the full ACRS in September.

CHAIR RYAN: That's great.

MR. BARTO: This is just a backup slide.

CHAIR RYAN: Well, I thank you for being so thorough in such a relatively short period of time. We're really getting a lot of information this morning. We appreciate it. I guess we're scheduled now for two briefings, one from EPRI and another one from NEI.

So if we can do a quick change around the table and gather up the speakers, and again, thank you all very much for a very informative presentation.

MR. BARTO: Thank you.

(Off record comments.)

MR. MACHIELS: Good morning.

CHAIR RYAN: Your name, sir.

MR. MACHIELS: My name is Albert Machiels and I'm with EPRI, and thank you for the opportunity to present some material which I hope you will find relevant to the discussion that you've had today. My intent is to come back to discuss or to present information on a couple of items.

The first one is the probability of criticality run during transportations, which build up on the misload analysis that has been already discussed to some extent. This is work that we did a number of years ago, four, five, six years ago, and it had to do within the context of transporting high burnup fuels. So the high burnup picture will also be mentioned a number of times.

The second topic is I'm going to talk about burnup credit validation. This is recently completed work, and it has been motivated really by a different application, criticality analysis in spent fuel pool. But the work that we did under that context, I believe fits very well also in the context of transportation.

Now what I'm going to talk about is not to detract of the information that we have received today. I think it was properly characterized of the ISG having been very well-received and being a

significant progress over Rev 2.

But what I want to do is at least provide some additional information that might be of interest to the topic that we're talking about. So with regard to brief introducing criticality safety and burnup, which we got to brief the criticality safety, we'll just mention that the standards and the methodologies were originally developed for the front end of the fuel cycles, with fairly pure materials.

So at that time, we were talking about critical analyses of simple mixture like enriched uranium or plutonium. Those involved particular species like enriched uranium with some relative isotopic content, or plutonium maybe, which is a handful of species.

Now spent fuel is a challenge, because what makes a lot of sense for a simple situation is very complicated, that in spent fuel obviously you have a large number of nuclides involved. Clearly, a lot of those will disappear very quickly because of their half life.

But still when we talk about composition of spent fuel after some time, we still have to deal with about 400 nuclides of interest. Clearly, a methodology which is allowed to look at nuclide per

nuclide become less practical when you have to talk about a very large number of nuclides.

So spent fuel is an issue when we talk about criticality analysis, in terms of relating how these safety methods have evolved over time. So one way to alleviate this complication about the fresh fuel assumption; as you know, that introduces a significant amount of conservatism, especially as the fuel designs have evolved to higher enrichment.

If you follow that, you end up with a low capacity system for storage and transportations, which means more systems, more operation, increased cost and overall, this approach typically, especially in transportation, would not result in the optimal safety.

The reason being is that the non-radiological risks dominate the radiological risks. If you take transportation, for example, by truck, the classic accident rate just of normal transportation overwhelms the risk coming from the radiological side.

So from that point of view, from a point of view of minimizing risk, the idea is to minimize the number of shipments. That means to maximize the loading for a given -- a capacity for a given volume.

And so burnup credit has already been

mentioned. It's simply giving credit for the reduced reactivity of spent fuel compared to fresh fuel, and that has been the evolution that the NRC has below for transportation for the first version of the ISG back in '99.

As already discussed, the burnup credit comes in different flavors. You have actinide only, then you have actinide-only plus a subset of a fission products, which have been extensively discussed, with including some minor actinides.

Then there's the full burnup credit, which basically you don't leave anything on the table, but you're going to try to take advantage of everything which is in the spent fuel.

The technical challenge with those as has been indicated is to be able to calculate the uncertainty of the biases that may come with some of those different subsets. From that point of view, in my case, what I'm going to talk is about the full burnup credit, the last one.

That means that we are looking at spent fuel as an entity, without neglecting any part of it. We could arrive to the same system by basically looking at all the nuclides in the spent fuel. But it would entail basically a large amount of work, and also are

collectively then a lot of fission products which have very little impact on the criticality analysis.

Together, a large number of them may have an impact. But if you look at them individually, individually they have little impact and it would come with fairly large uncertainties, that you basically would get to negative feedback if you want to include more things with larger uncertainties.

So basically what we're going to look at eventually is the full burnup credit approach. But before I get into that full burnup credit approach, I'm going to the dark side for a moment, and say that with regard to regulations, we have a fairly simple requirement with regard to subcriticality. It has to remain basically subcriticality, subcritical and in all conditions, normal and accident.

Normal means that you have control of things. That means you're doing things the way you expect to do them. So your cask is designed for that and this is an issue. For accidents, it's a difference. It's a loss -- accident is a loss of control. There are things that are happening obviously, that you don't have to happen typically.

And with regard to the NRC positions, this is where the high burnup picture gets into the

picture, is that when the burnup of the fuel is less than 45 gigawatt-days per metric ton, the NRC assumed that in that case, the impact loadings resulting from accidents results in the fact that there's not much change in the geometry of the assembly within its normal configuration.

That means that all the analyses that you made for normal configurations will hold for accident conditions as well. However, when we talk about high burnup, there are definitely some questions about how the cladding would behave, and in that case, it's not straightforward to assume that it would be the case.

That means now you basically have options right now to say well, even if there's damage, there won't be any water because of there's all these reasons, or you can make an analytical simulation which says those are the worst conditions for reconfigurations, and calculate the impact on criticality.

The observation is that there are sort of contradictions in this to the extent that clearly, high burnup means that you have achieved somewhat something equal or larger than your designed burnup, and that means that the reactivity, the leftover reactivity is very low, okay.

If for reason that you suspect that now you're dealing with something which has a very low reactivity compared to what it means, that means also that the cladding has a half life basically that led to a low burnup. So you could assume really that if you have significant underburning, whatever it means, is that the normal configuration would be assumed.

Which means that with burnup, if you assume that the life of the reactor will lead to deterioration of the cladding, such that it will eventually this property will degrade over time, as the cladding properties degrade, that means that the burnup has been fairly high.

From a criticality point of view, the more burnup you have, the better it is obviously, because you have the reactivity very low.

Now there has been some scenario which have been considered, and published in an Oak Ridge report, and some scenario though really beyond critical conditions, as I will mention, and according to the report, they represent the theoretical limits of the effect of severe accident conditions.

Certainly, we agree that they go beyond critical conditions, but I think have some question about their represented theoretical limits.

An example which is easy to visualize is that you would have an assembly, and as a result of an impact, basically all the cladding would basically be removed, and what you would have basically is a column of fuel, which would maintain the identical geometry as in the assembly, but without any cladding supporting them. This is obviously beyond critical conditions.

Now even in those type of scenario, the calculations in terms of increasing k-effective, which basically is fairly small, to the extent that they are less than five percent, .05 compared to 1 for a case like this, which means that even if you would assume that Mother Nature would not behave the way it's supposed to, is that you would go from a normally designed cask, which has a k-effective of equal or less than .95, to something which is less than 1, even assuming those more drastic scenarios.

And this obviously is something which indicates that it's not possible to get the reconfiguration in such a way that you would get criticality. We did some similar work, and we came to the conclusion that actually k-effective is much more likely to decrease rather than increase as a result of reconfiguration, and the very simply reasons for that

is that PWR assemblies tend to be fairly optimized to start with.

That's the way they are designed. The environment in a PWR is fairly uniform, and deviations from the normal conditions tend to over-moderate or under-moderate, while the ideal is actually the normal conditions. From a reactivity point of view, the most reactive configuration is actually the actual geometry of the assemblies, at least for PWR.

Now shifting a little bit still in the accident, in the risk domain, let's look at misload. We looked at that some close to ten years ago, and you see a curve here where we assume that we have a cask which is loaded with a fuel burned to 45 gigawatt-days per metric ton with the initial enrichment of being five percent.

And you can see on the X axis that the k-effective calculated is somewhere around between .85 and .9. Now each point now indicates on -- going along the curve indicates a misload, which means that here, this is no misload and here, I have a misload where I put something instead of being 45, I put a 25, and I put in the center of a cask, which is the more favorable position from a criticality point of view.

They introduced a second one, a second

misload next to it, in order to maximize the effect, a third one, a fourth one and so on. So basically increase the misloading, replacing each time a 45 by a 25. You can see basically how the k-effective increased.

Now the red curve is basically when you load a fresh fuel assembly, a five percent fresh fuel assembly. It will jump by about six percent by one misload, by another six to five percent or roughly by a second misload, if you put it again at the center and next to it.

So what it shows is that in this case, it would take about two to three misloads of a five percent to go beyond criticality conditions, and with regard to misloading with 25, it basically takes, doesn't get there.

Now this is basically a calculation starting from this point. The NRC will license and assume that the licensing conditions start at .95 now. Basically you have the limiting conditions, which are .95 and assuming this is true physical value, not including all the conservatism which are built in to make it .95.

Then you would start basically those curve, starting roughly at .95. You will see that to get at

1, it would require several misloads, three to five misloads of underburned assembly to get the critical conditions, but it would take a single fresh five percent assembly to go from .95 to over 1. So you can see the impact of misloading in a situation like this one.

Now we typically have disregard loading fresh fuel assemblies as being a potential for the following reasons. Some have been mentioned already, is that typically, the loading cask and refueling a reactor are done at different times of the year, and an example is TVA. Of 33 cask loadings, 33 cask loadings, two of them have fresh fuel and 31 of them didn't have fresh fuel in the pool.

Second thing is that the physical appearance of a fresh assembly compared to a once-burned assembly is different, as you can see on the picture.

So at least this is a visual check, and the last one is that the economic value of a fuel assembly, a fresh fuel assembly is over $1 million, and clearly, you don't lose track easy of something which costs $1 million, is that there will be obviously something that will prevent that misload of cask to be sent on the road with a fresh fuel in it.

So from that point of view, misloading of

fresh fuel, from our point of view at least, is not really a credible event for the number I mentioned, and with regard to multiple misloading, it requires significant underburnup. It would take typically several assemblies to be misloaded.

But as we have seen, it's not an unreasonable assumption, given as we can see, how the errors can introduce themselves.

So what we did is that we did basically this type of work about five years ago, and what we did is that we started from the very beginning, excuse me, where we actually, the utility received fresh fuel to be loaded in the reactor.

Then we follow the fresh fuel upon reloads until it's discharged, and then the preparation of the sheets for loading, unloading and so on, and that we used as reference the plant procedure of a plant and our contractor was located in southern California, in Irvine. There is a plant which is located nearby basically.

We basically interviewed the plant and applied the procedures, and we also introduced some recommendations, in terms of how to manage the database, which is very foreign, and also we made an additional recommendation that when you get at the end

of life here, and you have your cask sitting, is to go back and look at the records, and go back to the original reactor records.

That means you make the relationship again between the assembly and the records from the reactors, such that some of the discussion that we had that if an error happens, basically that you don't, you are unable to correct it, is that actually you're able to do that.

The reason is basically based on the discussion that we had earlier, is that the key issue is basically introduction here in the database here as you go along, and there will be plenty of opportunity to basically correct picking the wrong assembly.

To our knowledge, that has not ever happened that somebody picked the wrong assembly, at least having been documented and put it into a cask. But at least there are records of paper issues and configuration management.

So that's what we did, and then we looked, since this is a 32 PWR assembly, this is railroad transportation, the Federal Railroad Administration database. Plenty of data will give you plenty of information about statistic about accidents, hazmat transportations, the velocity of the train, what did

happen and so on.

Then finally we got what is the probability of if you have an accident. An accident is large enough, is severe enough that there is a defect in the cask. Water is present, and basically, you know that you have motivation for criticality.

When you put those together, you can see that -- you can see this is a very small number to start with, okay, and although it's a small number, it's likely to be even smaller now because this work was done many years ago, and the updates, which have been done in that, indicate what the methods available, that the frequency or probability of this is probably several orders of magnitude compared to what we used.

The next one is this one here, and that's assuming that all the recommendations that we made were implemented, because we get credit obviously for recovery, and then finally the last one that contributes to the risk is this one.

The bottom line is that when you look at all those factors, the likelihood of a potential criticality event during a 2,000 mile railroad shipment of casks designed for 32 PWR assemblies is basically extremely low, or about 10 to the minus 16.

Now you can see, this is a point estimate, and you can change those numbers. For example, if you don't implement the recommendation that we made in these studies, this can increase by a factor of three, 10 to the minus 13. On the other hand, if you put some restriction on velocity of the train, for example, not going over a certain velocity, then you reduce the probability of a severe accident, and you can lower this number by another couple of orders of magnitude.

So there is obviously this is not an accurate number, but depending basically how you handle the numbers here, you get something which is in our case extremely small, but with a pretty broad range of values, depending on what you do or what you don't do.

Okay. So away from accident conditions now, and going back to the main thrust, which is a conservative estimate of the loss of significant reactivity as a function of burnup, range up to 60, and with obviously a requirement of coming within an uncertainty of an estimate.

We basically adopted a different approach, which came from the spent fuel pool environment, because in that case, they always have so far used a

full burnup critical approach, and the estimate was based on a genuine judgment, which was based on some understanding and the accuracy of the code to predict a number of things.

And so what we did is we went back, because you have -- basically in a reactor environment, you have fuel at the beginning of cycle, from fresh fuel to something which has been burned through roughly one cycle. Then at the end of the cycle, you have something, a range of burnup from something which has been burned through one cycle, to something which is ready to discharge.

So you have a true representation and sampling of burnup from zero to basically the point of discharge here. We do that by doing, not taking advantage of the measurements which have been done in the reactor, flux map essentially, which are required, are part of operating the reactor.

And so we entered into a cooperative effort involving Duke Energy, Studsvik Scandpower, and a lot of that was inspired by Dale Lancaster, Dr. Dale Lancaster. The principal investigator was Professor Kord Smith, who's now an endowed professor at MIT.

And what we did is basically collected data from four PWR loop reactors, over 600 flux maps, and

over a million data points basically. And this represents basically the core here, it's maybe hard to see. But without looking at any number in particular, where you see a square where there are numbers, those obviously are the instruments, the location of instrumentation.

So once a month, there is a requirement that you introduce a fission chamber and measure basically the reaction rate into those. So you get a map of the power distribution in the reactor, and you have to be able to predict that before you start the reactor, what's going to happen.

This is high-precision measurement, and what the advantage or what we can do with that is that you can basically extract the value of the burnup in all position in this reactor. Because if you do not do that correctly, you will not be able to reproduce the power distribution.

So the power distribution in the reactor is very dependent of the burnup of the fuel, of the composition of the fuel in the different locations. So by taking advantage of that, and doing a lot of perturbation on sub-batches basically, we can extract information which I will briefly describe.

The information that we have, we have

translated that information into 11 reactivity decrement benchmarks, assuming a 17 by 17 PWR reactor design.

So you have here basically 11 problems or 11 that we're giving you here, and for each of those, each of those levels, for example, this one, we give you an exact geometry of what the result we're going to give you.

You for -- also then for each burnup, we are going to give you the difference in reactivity between zero burnup and 10, 20 and 60 and so on. With your code, whatever code you want to use to use in your storage or transportation cask, you can compare, because this is actually experimental value, which will tell you how well your code is going to do in comparison of those values.

So those are essentially measurements of criticality which are obtained on actual spent fuel, which basically contains a complement of all the nuclides in the spent fuel. Dale did a little calculation here, and what I didn't mention is that in those benchmarks, we also come up with uncertainty value, which I will not get into the discussion.

But the fact is that the accuracy coming from the reactor is actually extraordinary. It's

very, very good. Our main uncertainty is that when we extrapolate the reactor conditions, typically a reactor will run around 300 degrees Centigrade and the fuel inside, the fuel at some higher temperature.

When we extrapolate the number from reactor conditions to cooled condition, this is where most of the accuracy is actually coming from.

But the reactor itself, when you look at the million, over a million data points, and when you're looking at over 600 flux maps coming from four different reactors, the accuracy sticks out extraordinary and it takes into account isotopic content, cross-section value and so on. So it takes a full, all the elements which are typical of a criticality analyses.

Dale did the quick calculations, which shows basically the bias and uncertainty coming from the methodology, which is recommended in the ISG, and you can see that basically you go between a value of about 1.5 to about 3 percent between those two values. Ours is essentially constant at less than one percent.

This is using the same tool. The difference obviously is first of all, we involve all the elements here, rather than a limited number. But the fact is it's coming from the accuracy of the measurements that

we have here, compared to the accuracy of the data that this method has to rely to.

As mentioned before, basically it's not straightforward to take a small piece out of a fuel rod. We basically gather representative sample and then be able to calculate what should be the conditions that prevails, so that you can make a comparison of what you calculate compared what you find there, and on top of that, you have to insert what is coming from the chemical analyses.

That's what is very easy to see. All you have to do is sell the same sample to three different labs, and you can be assured that you will have a variation in the number. That has been known as the Yankee Rowe data from a number of years ago, and there was a difference of 20 percent between the result of the lab coming from the same sample. So this is basically -- basically this is a given when you deal with chemical analyses.

So in summary, I just wanted to let you know that there are definitely a proposed alternative approach, which is relying on full burnup credit, not looking at individuals at the time, and having to assume content plus uncertainty, then cross-section and uncertainty.

We certainly consider that the reactor data are experimental benchmark. Also, the reactor operators don't want to be called basically an experimental as to the process. This is the methodology which also has the advantage of being applicable to storage, transportation, disposal, wet or dry storage.

It basically has the beauty that it has a continuity from the reactor operator going down the chain, going into the pool, going to the cask for storage and eventually going for transportation.

So coming back now to transportation specifically, burnup credit is definitely a high priority topic. As mentioned, increased cask capacity for 24 we can make it to 32. We mentioned that a great percentage of the spent fuel population can be loaded.

Our assessment shows that there's an extremely low probability for the potential of a critical event in transportation, due to a number of factors that I've mentioned. And I will then typically, recognizing Cecil in the audience there, again that's always been our basically message, is that the conservatism is good, but at sometimes it can hurt you because you don't have only the radiological

risk to consider, but also you have the non-radiological risks.

Often, we are worried about the damage to the cask, but in fact what we should be worried is the damage caused to the cask by something else. Thank you.

CHAIR RYAN: Thank you.

MR. MACHIELS: I have behind a number of reference, including --

CHAIR RYAN: We can see those in the package. Thank you very much.

MEMBER ARMIJO: Albert, is the industry or anyone proposing for the staff to apply this alternative approach?

MR. MACHIELS: We have submitted all the reports which are mentioned in the slides to the NRC, and we are waiting for -- we've basically submitted them about five, six weeks ago, and we are waiting for a response on them, in terms of whether they would be willing to review them, as well as whether they will ask for review fees, for example. But we have submitted them.

MEMBER ARMIJO: And this approach would be equally applicable to BWR core data as well as PWR --

MR. MACHIELS: The PWR, the principle is

somewhat similar in way that we use the reactor. But the BWR has the advantage of cold criticals. So we would collect the cold criticals from BWR, and basically go through a similar process, and it would be actually a little bit easier for BWRs.

(Simultaneous speaking.)

MR. MACHIELS: The fuel design is a little bit more complicated, but --

MEMBER ARMIJO: Yes. Okay, thank you.

CHAIR RYAN: Welcome.

MR. NICHOL: Well, good morning. My name is Marc Nichol from the Nuclear Energy Institute. I appreciate the opportunity to come and speak with you today. I recognize we're behind schedule, so if I could ask, how much time do I have?

CHAIR RYAN: You're probably allotted 30 minutes. We've got a meeting that starts just right at 12:00.

MR. NICHOL: Okay. I'll try to speed it up, to help us get out on time. So I would like to present today an industry perspective on burnup credit for spent nuclear fuel storage casks and transport packages, specifically on the NRC's proposed draft ISG-8 Revision 3. Also in here I'll discuss some of the comments that we had made and submitted to the

NRC.

In general, a high level view of industry's feedback. We generally believe that Revision 3 is a significant improvement from Revision 2. I'll explain that in a little bit, although we do think that there's some opportunity for further improvements, especially through flexibility and some risk insights.

If you go back and look at the history of burnup credit for cask storage and transportation, it's been a little bit contentious between NRC and industry over the years, in terms of differing perspectives.

I think this revision of the ISG puts industry and NRC closer together in our perspectives, although I would note that if you look historically, and I've heard some of the data today, that there's only been four transport packages that have applied for burnup credit, and none of them met the previous revisions of the ISG. Burnup credit hasn't been used for cask storage.

I think the insight to come out of that is that the ISG, the guidance, needs to be developed with the perspective of can it be used and will it be widely implemented by industry. So from that perspective, our major comments were designed to

improve the guidance, such that it results in a set of guidance that is easily adopted by industry and could widely be adopted for the future.

That, in my opinion, ultimately increases the value of burnup credit. Also some further background. Burnup credit is desirable by the industry. Certainly, calculating the burnup of fuel has been done from the beginning of designing the cores. So it's very well known how to do that. There's huge benefits in terms of cask storage and transportation for using burnup credit, and it assures, has reasonable assurance for adequate safety and protection.

And I would also note that if you look at spent fuel pool criticality, there's been decades when there's been use of burnup credit. So it's nothing new; it's just it hasn't been widely used in cask storage and transportation, and that's really where we focused, to get to that end goal, where it could be widely adopted.

We had five major areas of recommendations. The last four on my slide, I think the NRC has done a pretty good job of explaining how they're planning on addressing those. I don't think that there's too much contentious there.

I would like to focus on burnup verification, because I think that's really where if we could see improvements, we would also likely see an improved likelihood of adoption by industry.

So just to highlight, the NRC has done a good job explaining the improvements, but certainly greater benefit to utilizing the burnup credit, and this is because of the inclusion of fission products and extending the range.

They also have -- NRC has also used more risk insights and improved the flexibility. So we commend them for doing that. In the area of burnup verification, and this is certainly industry's greatest interest for providing our comments and hoping that we can influence some improvements to the guidance, we believe that the approach for burnup verification should fundamentally start at looking at the basic concept of how do misloads happen.

The misload is identified as loading an assembly with a higher burnup than what was qualified for that location. So industry went back and looked at all of -- well, we looked at some of the different types of misloads that could occur, and tried to put them into buckets.

We identified three buckets. The first is

loading the wrong fuel assembly. This is the one that's traditionally thought of picking up Assembly 1 when you should have been picking up Assembly 2. So that would be the first category.

The second category is calculating a burnup value higher than actual. So my calculations say that this assembly is 45 gigawatt-day per MTU, when in reality it's 44. That would be the second category.

The third category is assigning the wrong burnup value to a fuel assembly, and I think from what I've heard today from the NRC's presentation, in identifying that a lot of the errors occur in the procedural process at the beginning, this is the category that those would fall within.

And so even when you identify -- after identifying these three categories, our position is that one, you should, through risk information, identify which one has the most, highest probability or consequences. Then start to look at how do you advocate for mitigative actions, such that you could prevent or preclude or mitigate those types of consequences.

So we believe that the guidance should take that type of a focus.

MEMBER SKILLMAN: Marcus, before you

proceed, if I go back to the example, which is Slide 17 of the first presentation, in the Palisades event, it was the cooling time. In the North Anna and Surry event, it was asymmetrical decay heat limits and in Grand Gulf, it was database issues. It seems that there's a number four that needs to be on your list, and that is understanding the relationship of the characteristics of the fuel assemblies to the design requirements of the cask.

So I think it's more than just the wrong assembly or the burnup, or the burnup failure to the assembly, but it's understanding how that fuel assembly, whatever its burnup, presuming it's accurate, fits into the package.

MR. NICHOL: I agree. We intended three to cover that category, so perhaps we could go back in and rephrase number three. Number three was intended to capture those types of events, where you loaded an assembly with the wrong characteristics, or you did not identify those.

CHAIR RYAN: Basically, you've got a mismatch between the assembly and burnup.

MR. NICHOL: Mismatch, right, right. So you should have been loading an assembly with these characteristics, but somehow you loaded an assembly

with different characteristics. So our intent was to include that in three.

MEMBER SKILLMAN: Thank you.

CHAIR RYAN: Okay.

MR. NICHOL: So when we looked at those three, we also tried to evaluate the activities or the elements to prevent a misload, that the NRC has identified and that we could think of. Largely, we identified the very same ones that the NRC did, although we had a different perspective on their roles, functions and what they actually accomplish.

Those three elements that I'll discuss are burnup measurement, misload analyses and admin procedures. In the draft ISG-8 used -- burnup measurement was a primary method of verification. The alternative to that was misload analysis and the defense-in-depth misload analysis was some admin procedures.

When we considered the function and purpose of these different things, we actually saw it a little bit differently. So we saw the admin procedures being the primary method of verification. The admin procedures are the ones that are intending to prevent misloads.

Now we recognize that in the current state

of the procedures, that a misload is still credible. So because of that, we wanted to use misload analyses as a defense-in-depth.

Therefore, say using them to verify that if you did have a misload, even though you're trying to prevent the misload, if you did have a misload, that you would still remain subcritical. So that's the defense-in-depth approach we had proposed.

We did propose to eliminate burnup measurements. I believe burnup measurement is one of the major reasons why burnup credit in cask storage and transportation hasn't been more widely adopted, and so that we, industry does not want to do burnup measurements. That's why we recommended eliminating that.

I will say, going back to an earlier conversation you had on the front-loaded processes for selecting the assemblies, those are performed under QA programs and QA control. So just to clarify that point.

In essence, not only does industry not want to do burnup measurements because they're problematic to implement, but they're also inaccurate and we believe that they're not very effective at addressing those three types of categories.

So here, we don't believe that burnup measurements address Categories 1 and 3, 3 being the one that's been identified today as perhaps the most significant. It's, in our opinion, less effective than admin procedures for addressing the second category.

It is widely recognized that reactor records are very accurate, typically within two percent, and that reactor records are also calibrated with in-core measurements. So in effect, that's a type of measurement. We think that provides a high level of confidence that the records that are -- the calculated burnups are very close, and we also recommend taking an uncertainty penalty on the burnup, to further do that.

So in our proposed administrative procedures, these were designed to address those three types of categories, and in our formal comments to the NRC, we tied them a little bit more closely. I didn't do it on this slide. Certainly verifying the identity of the fuel prior to and after closing the cask would address Category 1, loading the wrong fuel assembly.

Verifying burnup value from a QA record would also be, I think that would be looking at Category 2, as well as reducing the reactor record by

the associated uncertainty. Verifying the fuel meets the loading criteria, number five here, that's really addressing misload Category No. 3, and that certainly is an important part.

As well as number six here, performing and developing all those processes and procedures, according to the QA program. We think that that's going to go a long way in preventing these, and certainly the NRC has identified some misload events.

I can't comment specifically on whether, on what processes and procedures they used, whether they've improved over time since those events or not. But certainly we can look into that.

MEMBER POWERS: But can you comment on the reliability of administrative procedures in general?

MR. NICHOL: I'm sorry. What was the question?

MEMBER POWERS: Can you comment on the reliability of administrative procedures in general?

MR. NICHOL: In general, they're reliable. Certainly, we agree that there are areas that could be --

MEMBER POWERS: I guess I'm looking for something other than an opinion. I'm looking for something, you know, that I can grab ahold of and say

it's reliable to this extent.

MR. NICHOL: I think that we would have to do some more quantitative analysis on that, to give some type of response to that.

MEMBER POWERS: Well, I mean it seems like that's absolutely essential.

MR. NICHOL: And I would agree, that that would be very important to do, to incorporate into this guidance, to further risk-inform, to make sure that the procedures and processes that are being proposed in the guidance are the ones that are --

MEMBER POWERS: It seems to me your problem is that the Commission has specifically in fact asked the staff not to rely exclusively or even predominantly on administrative procedures, and here you're saying this is the key to the thing.

MR. NICHOL: Well, this is --

MEMBER POWERS: You're asking them to do something that I would approach with a certain amount of trepidation, going up and telling the Commission that --

MR. NICHOL: Well not necessarily, because if you go to this slide, it's the primary means of verifying, but it's not the only. We also add in the defense-in-depth with the misload analyses. So what

we're saying is don't rely only on the administrative procedures.

Rely on them to a great extent to reduce as much as possible the potential, but also perform the misload analyses, such that if that event does occur, you're assured that it will remain subcritical.

MEMBER POWERS: If I do, if I make a mistake despite my procedures, or a misloaded things, what are the chances my misload analyses are going to be any use?

MR. NICHOL: Your misload analyses will have limited -- they will have already limited the burnup curves of the fuel, such that -- and they will have been informed by the potential for misloads. Such that that misload event would have been within an analyzed condition, and an analyzed condition that would have concluded that it's subcritical.

MEMBER POWERS: Okay.

MR. NICHOL: There are some procedures that we do not recommend. The first, verify location of high reactive fuel in the pool. This is less effective than what we've recommended, and it also requires additional resources. We don't believe that it's necessary. It could be some best practice that industry implements, but we don't believe it should be

in the guidance.

Qualitative visual is similar, that it's not absolutely necessary. We recognize the NRC has proposed an alternate to this, to do one or the other. That could be acceptable, and since qualitative is not very resource-intensive.

The condition for no fresh fuel in the pool would duplicate the qualitative one. It would mitigate the consequences, but we're proposing to mitigate the consequences through the misload analysis, to have that condition analyzed and bounded.

A requirement for not having fresh fuel in the pool could be problematic, if such a situation arose, where the risk fresh fuel and the utility does need to load a cask. There would be a very difficult position.

The pool inventory audit mentioned that this duplicates some other regulations, and should not be necessary. Now on the topic of independent third party verification, the question is not whether this should be performed or not. We agree this should be performed.

The question is whether it should be in NRC guidance, or whether it should be an industry-initiated best practice, which is currently performed

now. So all of the -- the utilities that I've talked to do perform independent third party verification as part of their best practices. So --

MEMBER ARMIJO: At what point in their process? In the selection of the fuel to be --

MR. NICHOL: My understanding, it's the entire process, from the very first selection of the fuel, all the way through the end. Now --

MEMBER ARMIJO: But somehow at least in the three examples that we were presented this morning, those processes are independent, but they don't really help.

MR. NICHOL: Yeah, and that's why I can't comment specifically on those, because I haven't researched them. But it's possible that this third party verification came about afterwards. Possible. I wouldn't know, but we could certainly investigate those to find out why they occurred and how they could have been prevented.

MEMBER ARMIJO: Yeah. I want to make sure I understood what you were saying about the misload analysis. Is it the NEI position that the misload analysis would be done with fresh fuel, since you're not going to verify that there's no fresh fuel in the pool?

MR. NICHOL: Yeah. Let me get to that. Yeah, that would certainly be one. So the misload analysis for the single misload would be a fresh fuel assembly, because you would have a condition that would allow fresh fuel in the pool.

We recognize, and also we -- well, we recognize for both of our proposed assumptions for the misloads that they're extremely conservative, and the NRC pointed that out. We didn't have sufficient time to come up with something that is both simple and a little bit less restrictive.

Certainly, the industry desires something that's easy to implement. Some of the proposals where you have to verify 90 percent, you have to verify your spent fuel pool inventory according to some assumptions, could be rather intensive and difficult to implement, as well as in the general cask design, it has to encompass all of the potential sites.

If you do that, that's rather restrictive. If you start to allow it to be site-specific, you get into an area where you may be -- it may result in multiple amendments to that cask --

(Simultaneous speaking.)

CHAIR RYAN: Another thing, you've got a variety of plants that exist in the United States,

more on the way, and over a 30 year or 60 year lifetime. They're going to be different.

MR. NICHOL: They're going to be different, and --

CHAIR RYAN: So saying that they can all be kind of cookie cutter doesn't seem to be smart.

MEMBER ARMIJO: Oh, it's a penalty that the industry would pay if they do it this way.

MR. NICHOL: Right, right. Yeah. It would be a penalty. Also, there's going to be future fuel that's discharged, and you're going to have to verify that the future fuel is encompassed in the old assumptions. Not that it could not be a valuable option. I believe it's conservative. I think it could be viable, but industry would desire something that's a little bit more simple to implement.

CHAIR RYAN: But it's not a simple problem you're trying to solve. There's a lot of variation within the problem, based on specifics we're given, right?

MR. NICHOL: No. There are a lot of variations in the specifics of the plant. But the cask, when it's approved, should be all-encompassing of the sites. If it is and it's overly-restrictive, then certain sites are going to request the vendor to

have a special amendment just for their fuel.

You get into a condition where there are so many amendments or potential for exemptions. So what industry would like is a streamlined process that's very simple to implement, but is not too restrictive.

Now I agree, that the answer may be in between what NRC and industry is proposing. Perhaps this is something that needs more effort to come up with a reasonable solution.

CHAIR RYAN: That's the sense I'm getting. It's a pretty wide playing field. I don't know how you get there with one answer, without having a range of possibilities within that answer.

MR. NICHOL: Yeah. Certainly from our perspective, we have not been able to put a lot of thought into how you could streamline this and not be overly-restrictive. But there could be a solution out there, and perhaps that's needed. More time is needed to find that.

MEMBER SKILLMAN: I guess I'm alarmed from the perspective that a number of years ago, there was an airplane accident, and what had happened is they refueled a gas turbine jet airplane with piston engine octane 100 fuel. Then the plane crashed and killed a bunch of people.

There has never been another incident like that. Somehow the airline industry realized that segregation of fuel types of critical. So while I appreciate what might be considered to be a making easier a requirement by those procedures not recommended on page seven.

We ought to be doing both. We ought to be making sure that the administrative procedures are so robust, and maybe more importantly, that the people that are handling the fuel really get it. So the procedures, so the process and procedures are restrictive is not the word I would choose, but are accountable enough that the likelihood of a misload is down in the grass.

Then on top of that, do the misload analysis to further reduce the likelihood of an event.

CHAIR RYAN: Doing a misload analysis really gives you, you know, what's the consequence, not what the likelihood is.

MEMBER SKILLMAN: Well, I'm going for prevent it, prevent it right in the front end of the dialogue, and it seems to me that industry ought to be saying you know what? This is one that's worth investing in, because it protects everybody.

MR. NICHOL: Which one was that?

MEMBER SKILLMAN: Some combination of the ones on page seven.

MR. NICHOL: On page seven?

MEMBER SKILLMAN: And clearly, as you have well pointed out, industry has great interest in ensuring that new fuel doesn't find its way into a cask, because that is a huge investment, and they would like to have as little SWUs going out the doors as economically appropriate.

But it seems that there ought to be kind of a mentality that says a couple of these, you have visual and others, might be worth their investment.

MR. NICHOL: We agree that some of these are very worthwhile, and as I mentioned, independent third party is already ongoing. The question is whether it should be in the NRC guidance, or whether it should be industry best practice. I think that could be, you know, discussed in great detail.

CHAIR RYAN: But sadly we don't have time for it this morning.

MR. NICHOL: Right.

CHAIR RYAN: So press on.

MR. NICHOL: Okay. In terms of depletion validation flexibility, I won't go into too great of a detail here, but certainly Albert from EPRI

presented on an alternative to the Oak Ridge method. We recognize the Oak Ridge method is overly conservative, and it is due to the measurement uncertainties. I think you heard a little bit about that today.

So there could be other methods out there that aren't as restrictive. So certainly flexibility in the guidance to accommodate potential future methods would be efficient from that standpoint. I think the NRC did a good job of addressing these comments in their presentation.

So here are my conclusions. So of course Revision 3 is a significant improvement. However, we believe that further improvements to the guidance could result in greater efficiency and effectiveness. Certainly industry has a great interest in having guidance that we would desire to widely adopt, and from that perspective, we hope that there's due consideration of our perspectives. Thank you.

CHAIR RYAN: Thank you very much, Marcus. I appreciate you being here. Any comments, Jack?

MEMBER SIEBER: No, I don't think so. I think the presentations were pretty well. I do favor some of the industry comments, like in-pool measurements as not being as helpful as it might be

because of uncertainty. Otherwise, everyone did well.

CHAIR RYAN: Thank you very much. Dana?

MEMBER POWERS: Yeah. I think the presentations were superb, and I especially appreciate Mr. Machiels?

MR. NICHOL: Machiels.

MEMBER POWERS: Machiels' comments about making it operationally simpler, where the guidance is fairly complicated. But I think I agree with Dick. Let's do a belt and suspenders approach here, because this is a mistake you just don't want to have, from all our perspectives.

Being able to do it so that it remains operational and flexible is really an excellent goal. But reliance on administrative limits is a problem for the staff, just because the guidance the Commission has given in connection with defense-in-depth regulatory philosophy.

You cannot do that. I mean I wouldn't want to be the staff going to the Commission to focus on administrative limits, in the face of that kind of guidance, without a lot of body armor. I mean it's just a problem for them, because five great Americans have told them don't rely exclusively or even primarily on administrative limits to the extent

possible.

So it's just a problem. So you've got to give them armor to go make that pitch, wherever, wheresoever they choose to make it.

CHAIR RYAN: Thank you. Sam?

MEMBER ARMIJO: Yeah. I agree with the prior comments. I think the Revision 3 is a huge improvement. I compliment the staff, Oak Ridge and everyone who worked on it. I think burnup credit, full burnup credit, you probably will never be able to get it unless you take an approach similar to what Machiels has proposed.

Whether you really get much by having the full burnup credit versus this actinides plus a certain set of fission products, whether there's really much benefit there, I don't know. But you know, I can see that as a next step in this type of analysis.

I think this has been a lot of progress and good work. Thank you.

CHAIR RYAN: Thank you, Sam. Dick.

MEMBER SKILLMAN: I do. I thank the staff and EPRI for the presentations. They were thorough, they were crisp, they were clear, well-done. I would like to recognize Mr. Machiels' comment in balancing

the risk between burnup credit, what is gained, and the transportation risks.

I think that's a very important point that needs to be clear in everyone's lenses. I'm going to settle on this idea that the process has to carry the day.

After all the data's completed, after everyone's, after the engineers and the physicists have said by golly, we know the k-effective is less than .95 or whatever the number is, there still needs to be a robust process that prevents an inadvertent misload, that results in an event.

It seems to me that we found a way to do that in so many other places in the industry, and so many other places in operating these plants. There's no reason why we can't insist on that same level of accountability and integrity on this piece of our processes.

CHAIR RYAN: Harold.

MEMBER SKILLMAN: Dr. Ryan, thank you.

CHAIR RYAN: Thank you.

MEMBER RAY: Yeah. I just think there's maybe too much emphasis on efficiency without the kind of rigor that I think everybody else has expressed here, in terms of ensuring that an event is avoided,

because the delta efficiency being achieved by not doing some things that can be done is very small.

CHAIR RYAN: I agree. Let me first thank the presenters for that. I think it's been a very, very good meeting and it, I think, summarizes a lot of work by Meraj, you and your staff and your consultants and contractors.

I think as all the members have expressed, we've gotten an awful lot out of today's briefing. So thank you all very much for your hard work, and coming today to present it to us.

On the technical points, I think I agree with the way that several colleagues have expressed it, and Dick in particular, that you know, it really should not be something because it's a little bit faster and the risks are low.

That's how we get in trouble, and I think we have to say things that, you know, we have to be rigorous, whether we think there's a chance that something could happen or not.

So I share those kind of sentiments, that we really ought to maintain the rigor of how we approach these things, even though they may be off. I was taken by the slide that showed that the misidentification chart led to a lot of problems or a

number of problems, more than any other route down that event tree.

So that stuck with me as sort of the watch word there. While we have to balance, we sure have to recognize that we have to err on the side of safety and conservatism about accepting by faith or some other method that things won't happen just because we're doing a better job.

So I think with that, if there are no other comments, we shall -- sir?

DR. LANCASTER: Are you asking for comments?

CHAIR RYAN: Yes. We have just a very short time, because we're ten minutes over.

DR. LANCASTER: Right. I just want to be very quick about this.

CHAIR RYAN: Just for the record, identify yourself?

DR. LANCASTER: Oh, I'm Dale Lancaster, Nuclearconsultants.com. I just want to make it clear, because some of the comments here, I think, may have missed something. A misload does not create an event, because if you miss -- we've been talking about PWR burnup credit.

The spent fuel pools have significant

dissolved boron, soluble boron that would prevent a criticality in any misload event in the pool. So that's the Case 1. So misload doesn't cause a problem there.

These casks are then dried, again in order -- if there was a misload, then you would have to flood these casks in order to get an event. So again, I think the primary answer is we are not relying on administrative procedures to prevent an event.

We're relying on administrative procedures to prevent one possible way of getting to an event.

CHAIR RYAN: That's the way I took it. It was one possible route. So I appreciate your thoughts. Thank you for that clarification. Raj, any last words or we're good?

MR. RAHIMI: I cannot go without rebuttal.

CHAIR RYAN: Well, don't rebut.

MR. RAHIMI: Okay. Meraj Rahimi, NRC. Again, these casks, we certify these casks that could be loaded in fresh water. No boron credit for these transportation casks. Once we certify these casks, they can be loaded, unloaded, even at the non-reactor site, non-borated pool. So that's what we use these casks for.

CHAIR RYAN: Thank you, Raj. With that and

hearing all the comments from members of staff, we'll --

MEMBER POWERS: And from the public.

CHAIR RYAN: Any members of the public? Is the bridge line open?

(No response.)

MR. BROWN: I never heard a beep, so I don't think anyone has called in.

CHAIR RYAN: No, I don't think we had anybody on the bridge line. Any other comments from members of the public, participants in the audience?

MR. BROWN: Mike, we'll get back to Drew on how to prepare for the full Committee on what --

CHAIR RYAN: Yeah, that will be fine. Yeah, we'll talk after the briefing about that. Anything else?

(No response.)

CHAIR RYAN: Hearing nothing else, the meeting's adjourned.

(Whereupon, at 12:10 p.m., the meeting was adjourned.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission

Proceeding: ACRS RPNM Subcommittee

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken and thereafter reduced to typewriting under my direction and that said transcript is a true and accurate record of the proceedings.

Neal R Gross

Official Reporter
Neal R. Gross & Co., Inc.

Interim Staff Guidance 8, Revision 3 – *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transportation and Storage Casks*

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee on Radiation Protection and Nuclear Materials

July 10, 2012

Agenda

Item	Topic	Presenter(s)	Time
1	Opening Remarks and Objectives	Dr. Michael Ryan, ACRS	8:30 – 8:35 a.m.
2	Staff Opening Remarks	Meraj Rahimi, NMSS	8:35 – 8:45 a.m.
3	Changes to ISG-8	Drew Barto, NMSS	8:45 – 9:00 a.m.
4	Cask Misload Probability	Brian Wagner, RES	9:00 – 9:30 a.m.
5	Burnup Credit Code Validation	Dr. John Wagner, ORNL	9:30 – 10:00 a.m.
6	Break		10:00 – 10:15 a.m.
7	Public Comments and Proposed Resolution	Drew Barto, NMSS	10:15 – 10:45 a.m.
8	Industry Efforts on Burnup Credit	Dr. Albert Machiels, EPRI	10:45 – 11:15 a.m.
9	Industry Perspective on ISG-8	Marcus Nichol, NEI	11:15 – 11:45 a.m.
10	Committee Discussion	Dr. Ryan, ACRS	11:45 a.m. – 12:00 p.m.
11	Adjourn		12:00 p.m.

Background

- Radiation and heat were the primary design drivers for older generation of transportation packages which were designed for short cooling times
- Sub-criticality became one of the primary design drivers for new generation of high-capacity casks for longer cooled fuel

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Background (cont.)

- To achieve high-capacity, cask designers eliminated flux traps (i.e. spacing between fuel) which are needed for the Fresh Fuel assumption, and relied on Burnup Credit instead
- Burnup Credit is credit for reduction in reactivity that occurs with fuel burnup due to the net reduction of fissile nuclides and the production of actinide and fission-product neutron absorbers

Background (cont.)

- Based on available data in 2002, staff issued guidance on taking credit for the major actinide isotopes.
- In 2007, SRM SECY-07-0815 stated:

  "… staff should focus its effort on using burnup credit as a means to insert more realism into spent fuel transportation cask criticality analyses."

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Background (cont.)

- In 2008, letter from ACNWM to Chairman Klein stated:
  - "… recommends that the staff take a risk-informed approach to evaluating Burnup Credit, including consideration of realistic and credible scenarios, probabilities, and consequences."
- In May 2012, staff issued draft ISG 8, Rev.3, for public comment. This ISG provides guidance for taking credit for actinides and fission products

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Major changes to ISG-8

- Credit for minor actinides and fission products
- Extend credit up to 60 GWd/MTU assembly-average
- Provide option for misload analysis with additional administrative loading procedures in lieu of burnup measurement

Expanding Technical Basis for Burnup Credit

- NUREG/CR-6979, *Evaluation of the French Haut Taux de Combustion (HTC) Critical Experiment Data* (2008)
- NUREG/CR-7012, *Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel* (2011)
- NUREG/CR-7108, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Isotopic Composition Predictions* (2012)
- NUREG/CR-7109, *An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses – Criticality (k_{eff}) Predictions* (2012)

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ISG-8 Revision 3 – Code Validation

- Availability of French HTC actinide data gives greater degree of confidence in actinide criticality validation than existed at the time ISG-8, Rev. 2 was published
- New chemical assay data expands the available database for fission product depletion validation and extends the range of applicability to higher burnups
- All available data used in NUREG/CR-7108 and -7109 to develop alternative isotopic depletion and criticality code validation methodologies.
- ISG-8, Revision 3 recommends crediting both actinides and fission products for up to 60 GWd/MTU

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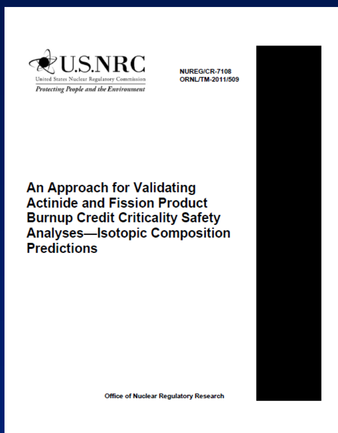
ORNL NUREG/CRs



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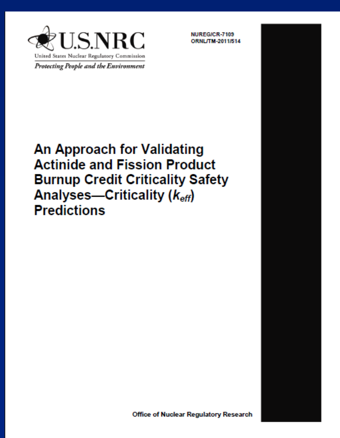
NUREG/CR-7108
ORNL/TM-2011/509

An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions

Office of Nuclear Regulatory Research



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ORNL/TM-2011/514

An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions

Office of Nuclear Regulatory Research

- Work performed under joint contract (SFST/NRR/NRO) through RES
- New isotopic depletion code validation methodologies and reference bias and bias uncertainty values
- New minor actinide and fission product criticality code validation methodology and reference bias value
- Provides recommendations regarding the use of the reference values, and the use of methodologies developed in the NUREG/CRs

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ORNL NUREG/CRs

- Applicant may use the reference bias and bias uncertainty numbers developed by ORNL in lieu of an explicit depletion or minor actinide and fission product criticality validation, provided:
  - the same code and cross section data are used in the applicant's analysis
  - the applicant's storage or transportation system is demonstrated to be similar to that evaluated in the NUREG/CRs
- Applicant should perform traditional criticality code validation for major actinides using MOX and HTC data

Code Validation – ISG-8, Revision 3

	Major Actinides	Minor Actinides and Fission Products
Criticality Analysis	Applicant can perform analysis with Fresh UO_2, MOX, & HTC experiments	Use ORNL-supplied bias number
Isotopic Depletion Analysis	Use ORNL-supplied number, or use ORNL-developed validation methodologies	

ISG-8 Revision 3 – Burnup Measurements

- NUREG/CR-6955, "Criticality Analysis of Assembly Misload in a PWR Burnup Credit Cask" (2008)
- NUREG/CR-6988, "Review of Information for Spent Nuclear Fuel Burnup Confirmation" (2009)
- RES report: *Estimating the Probability of Misload in a Spent Fuel Cask* (2011)
- ISG-8 modified to allow misload analysis combined with additional administrative procedures in lieu of direct measurement

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Misload Analyses

- Single *severely* underburned misload, chosen such that reactivity bounds 95% of the underburned fuel population with 95% confidence
- Multiple *moderately* underburned misloads, chosen such that half the cask is filled with a fuel assembly that bounds the reactivity of 90% of the total discharged fuel population
- Reduced administrative margin ($\Delta k_m \geq 0.02$)
- Additional administrative procedures, such as ensuring no fresh fuel in pool at time of loading, or independent, third party reviews of cask loading

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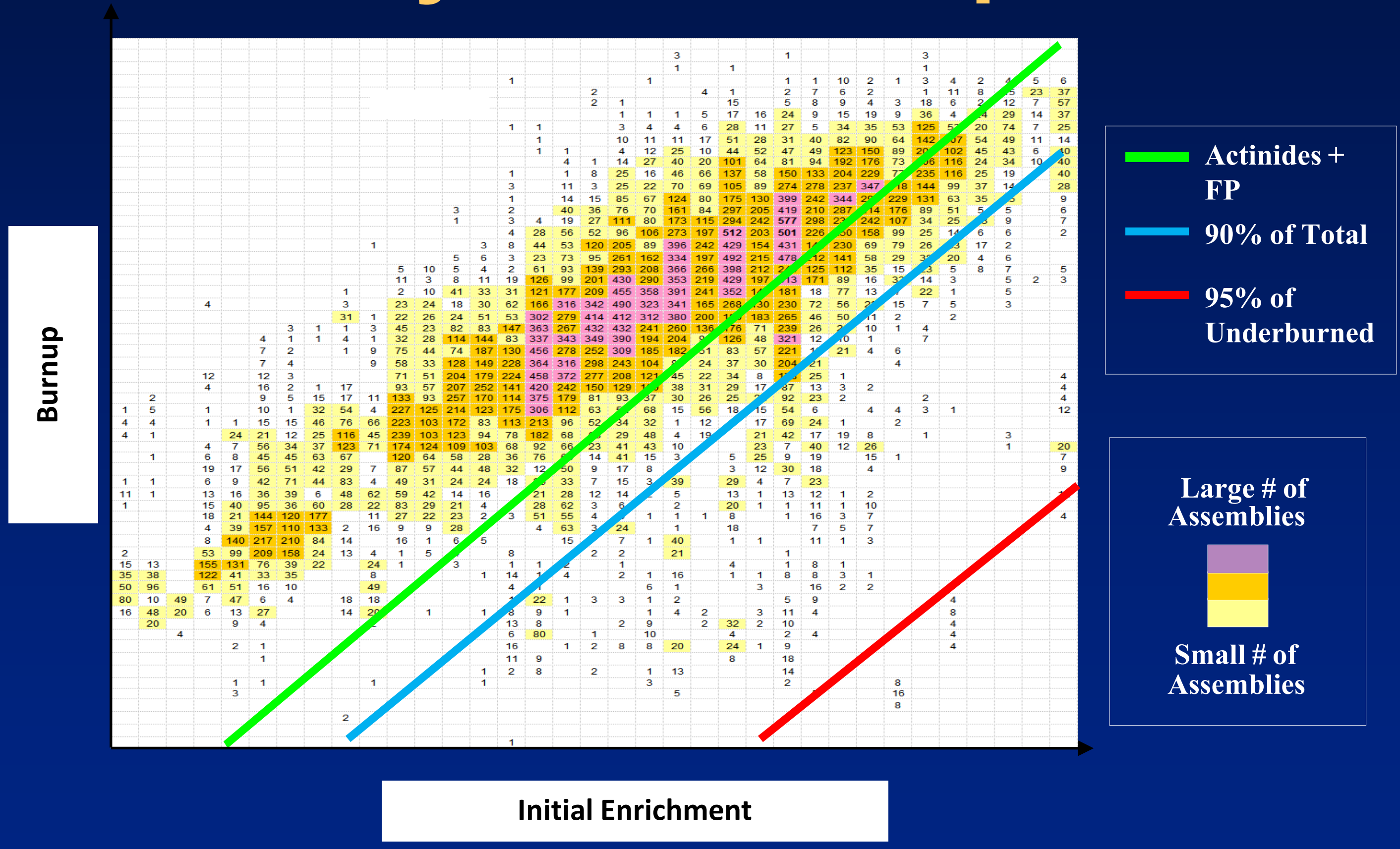
Misload Analysis Fuel Population



From the 2002 EIA RW-859 Fuel Database

Misload Report

- Reviewed cask misload events to determine underlying causes and to identify common failure modes
- Calculated the probability of single or multiple cask misloads using two separate methods
  - Empirically from actual misload data
  - Using an event tree model
- Considered impact of burnup on misload probability

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Misload Events

- **Palisades**: Calculation for cooling time was based on planned loading date, was not updated when loading date changed. 5 casks, 11 assemblies misloaded.
- **North Anna & Surry**: Cask design allowed for asymmetrical decay heat limits. Written procedures did not adequately explain this requirement leading to repeated errors. 11 casks, ~19 assemblies misloaded.
- **Grand Gulf**: Improper use of database containing incomplete information led to loading of assemblies exceeding allowed decay heat. 4 casks, 34 assemblies misloaded
- **McGuire (near misload)**: Crane picked up incorrect assembly adjacent to the correct assembly. Error caught while assembly was being lowered

Total of 20 casks misloaded out of 1200 → ~10^{-2} per cask

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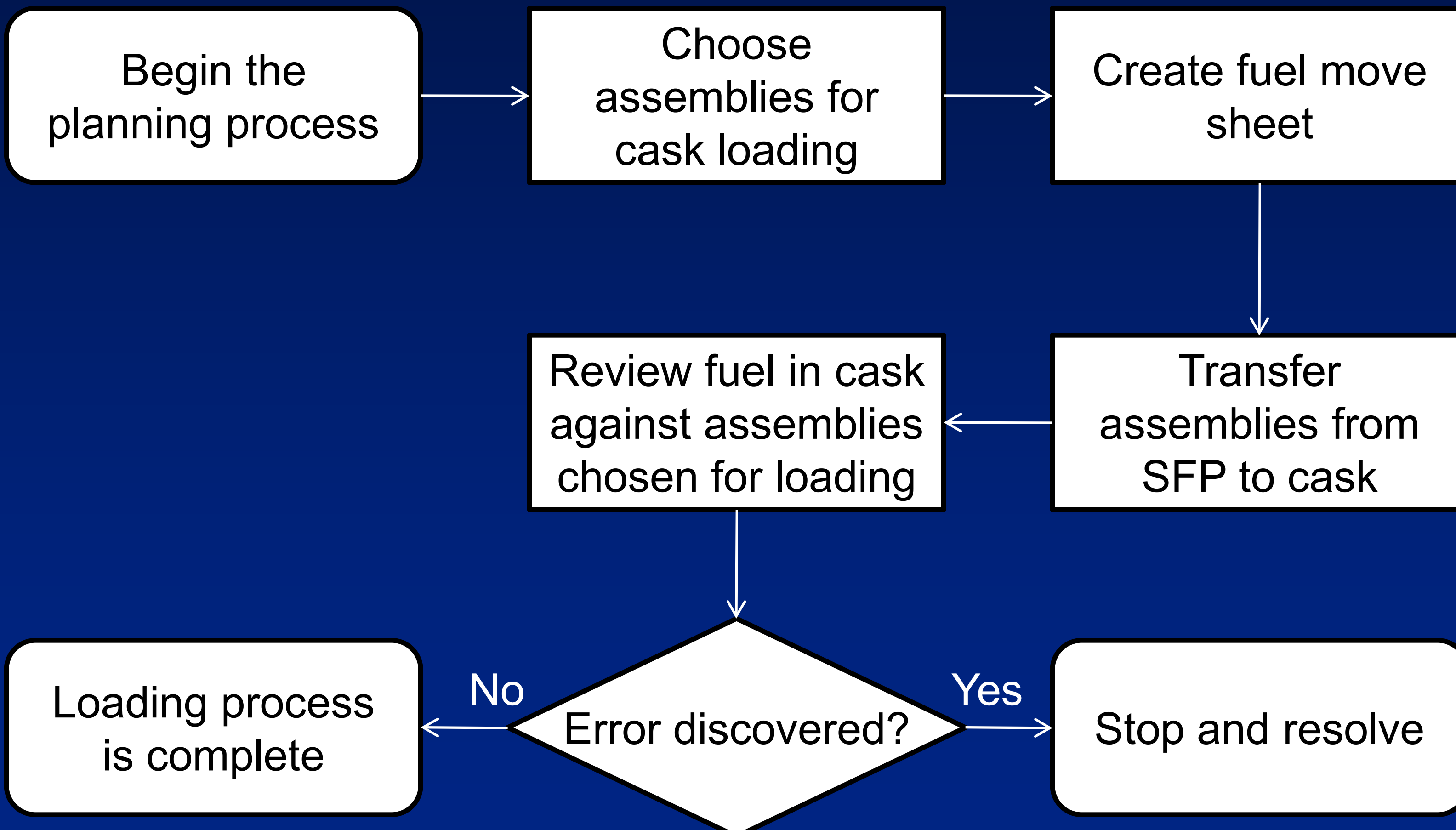
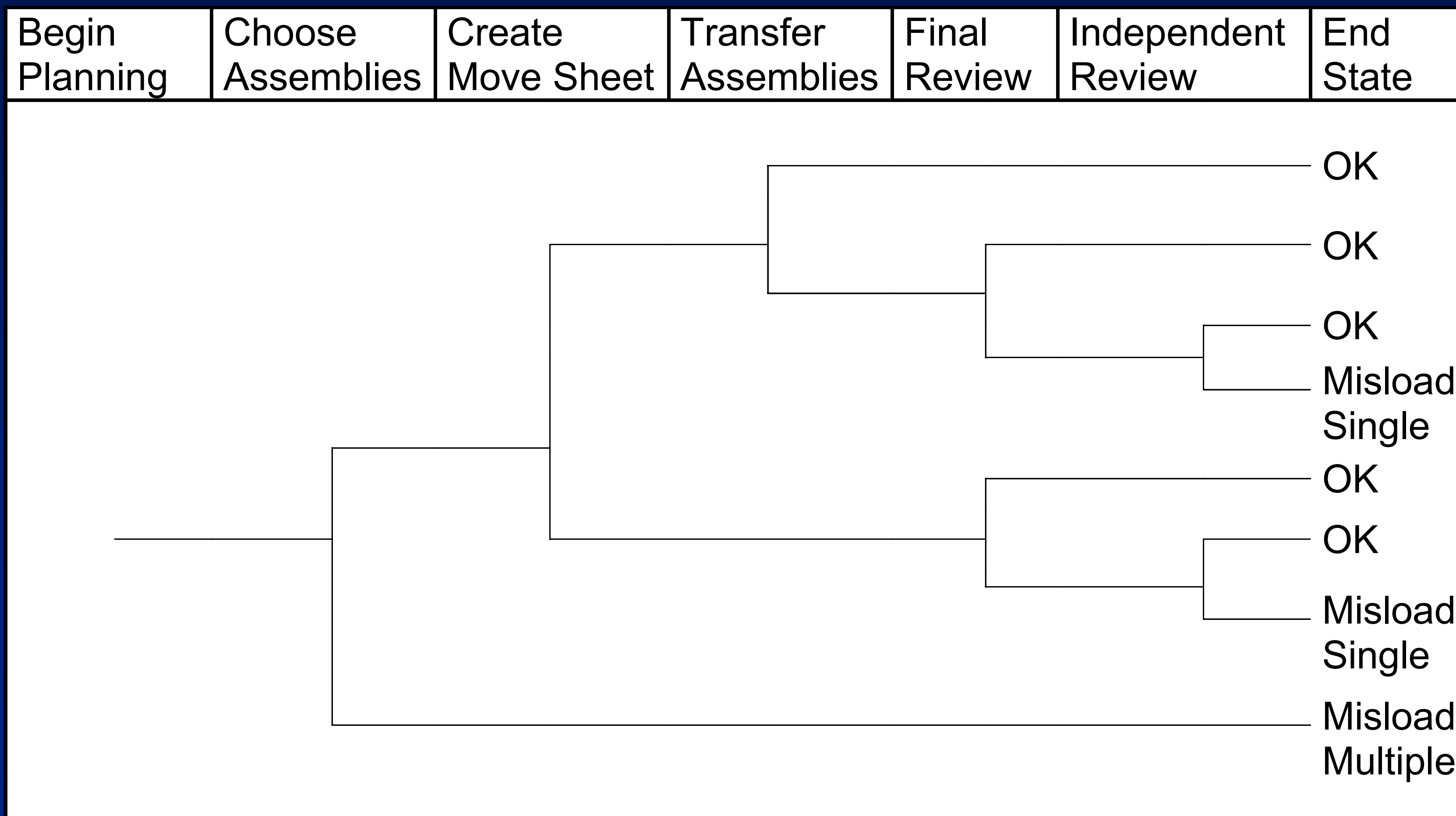
Cask Loading Process



Event Tree


Begin Planning
Choose Assemblies
Create Move Sheet
Transfer Assemblies
Final Review
Independent Review
End State
OK
OK
OK
Misload Single
OK
OK
Misload Single
Misload Multiple

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Event Tree Model Details and Insights

- Empirical SFP data used for fuel transfer errors, human error probabilities from THERP used for other values
- Multiple misload event is dominant sequence
  - Multiple reviews limits the potential for single misload events
  - Multiple misload events reflect fundamental misunderstanding of cask requirements or error in procedures, database, etc and are therefore harder to catch with a review

Misload Conclusions

- Misload events are credible
  - Empirical probability: 20 misloads / 1200 casks loaded ≈ 10^{-2} per cask
  - Event Tree Model probability ≈ 10^{-3} per cask
- Based on event tree model and empirical data, misloads are most likely caused by errors in the planning process
- Event is likely to involve multiple assemblies and casks

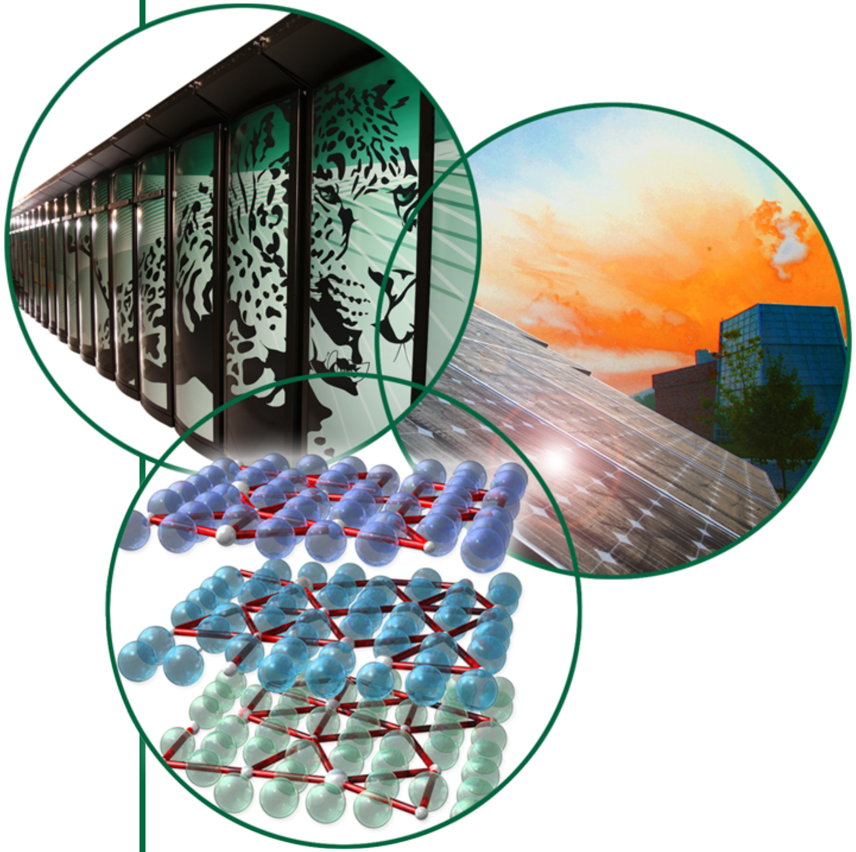
Burnup Credit Code Validation

Presenter: John C. Wagner

Key Personnel: J.M. Scaglione, D. E. Mueller, B.J. Marshall (criticality); G. Radulescu, I. Gauld, G. Ilas (depletion)

Presentation to the Advisory Committee on Reactor Safeguards Subcommittee on Radiation Protection and Nuclear Materials

10 July 2012



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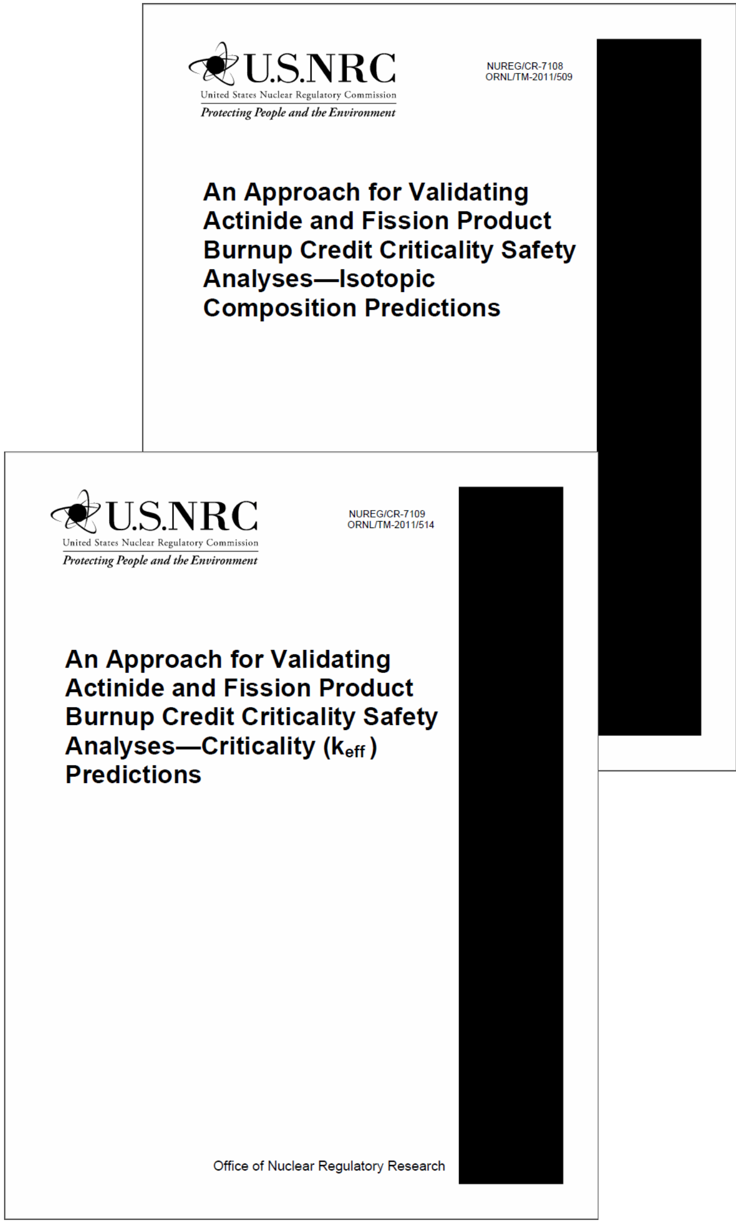
Outline

- Background and Purpose
- Depletion Validation NUREG/CR-7108
  - Approach
  - Reference numbers
  - Recommendations
- Criticality Validation NUREG/CR-7109
  - Approach
  - Reference numbers
  - Recommendations
- Closure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NUREG/CR-7108
ORNL/TM-2011/509

An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Isotopic Composition Predictions

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NUREG/CR-7109
ORNL/TM-2011/514

An Approach for Validating Actinide and Fission Product Burnup Credit Criticality Safety Analyses—Criticality (k_{eff}) Predictions

Office of Nuclear Regulatory Research

Background

- Most significant challenge to expanded burnup credit has been the validation of depletion and criticality calculations
  - In particular, the availability and use of applicable measured data, especially for fission products

- Applicants and regulatory reviewers have been constrained by both a paucity of data and a lack of clear technical basis or approach for use of the data

- Rationale for restricting ISG-8, Rev. 2, to actinide-only was based on limitations in available validation data at the time

Purpose

- Establish technically sound validation approaches for spent nuclear fuel (SNF) criticality safety evaluations based on best-available data and methods
- Apply the approaches to representative SNF storage and transportation configurations/conditions to demonstrate their usage and applicability and provide reference results
- Document the validation approach, its technical basis, and its application

Depletion Validation Technical Approach

- Bias and uncertainty in predicted fuel isotopic compositions based on comparison to measured isotopic compositions from destructive radiochemical assays (RCA)
- Monte Carlo (MC) uncertainty sampling method used to estimate the bias and uncertainty in k_{eff} due to the bias and uncertainty in the predicted isotopic compositions
  - Additional analyses using the direct-difference method and S/U techniques
- Bias and uncertainty in k_{eff} determined for representative PWR transportation/dry storage cask with SCALE 6.1 and ENDF/B-VII data
- Sensitivity of bias and uncertainty in k_{eff} to relevant parameters was evaluated

RCA Data (100 PWR fuel samples)

Reactor	Measurement Laboratory	Experimental Program	Assembly Design	No. of Samples/ Fuel Rods	Enrichment (wt % ^{235}U)	Burnup (GWd/MTU)
Trino Vercellese	Ispra, Karlsruhe	JRC	15 × 15	15/5	2.72, 3.13, 3.897	7.2–17.5
	Ispra, Karlsruhe	JRC	15 × 15	16/5	3.13	12.9–25.3
Obrigheim	Ispra, Karlsruhe	JRC	14 × 14	10/6	3.00	17.1–37.5
	ITU, IRCh, WAK, IAEA	ICE	14 × 14	5/5	3.13	27.0–29.4
H. B. Robinson-2	PNNL	ATM-101	15 × 15	4/1	2.561	16.0–31.7
Turkey Point-3	Battelle-Columbus	NWTS	15 × 15	5/1	2.556	30.5–31.6
Calvert Cliffs-1	PNNL, KRI	ATM-104	14 × 14	3/1	3.038	27.4–44.3
	PNNL	ATM-103	14 × 14	3/1	2.72	18.7–33.2
	PNNL, KRI	ATM-106	14 × 14	3/1	2.453	31.4–46.5
Takahama-3	JAERI	JAERI	17 × 17	13/3	2.63, 4.11	17.4–46.2
TMI-1	ANL	DOE YMP	15 × 15	11/1	4.013	44.8–55.7
	GE-VNC	DOE YMP	15 × 15	8/3	4.657	22.8–29.9
Gösgen	SCK•CEN, ITU	ARIANE	15 × 15	3/2	3.5, 4.1	29.1–59.7
GKN II	SCK•CEN	REBUS	18 × 18	1/1	3.8	54.1

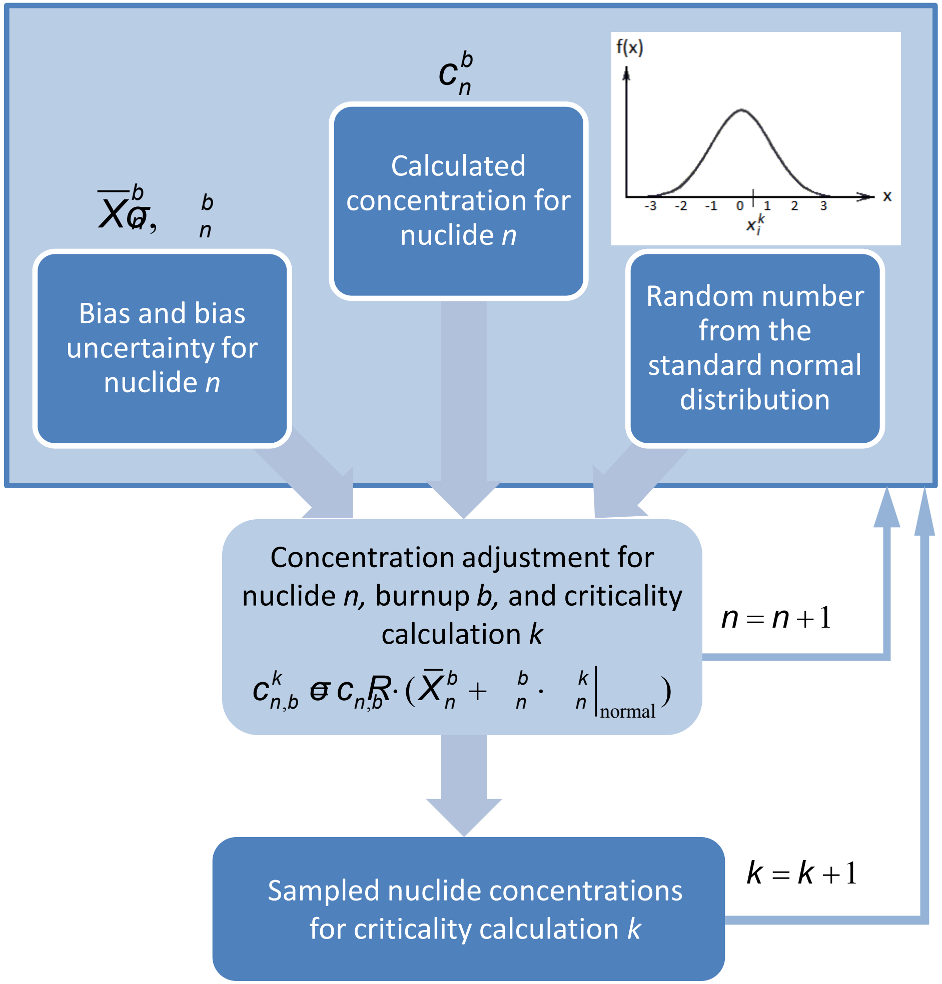
PWR RCA Data for Burnup Credit Nuclides

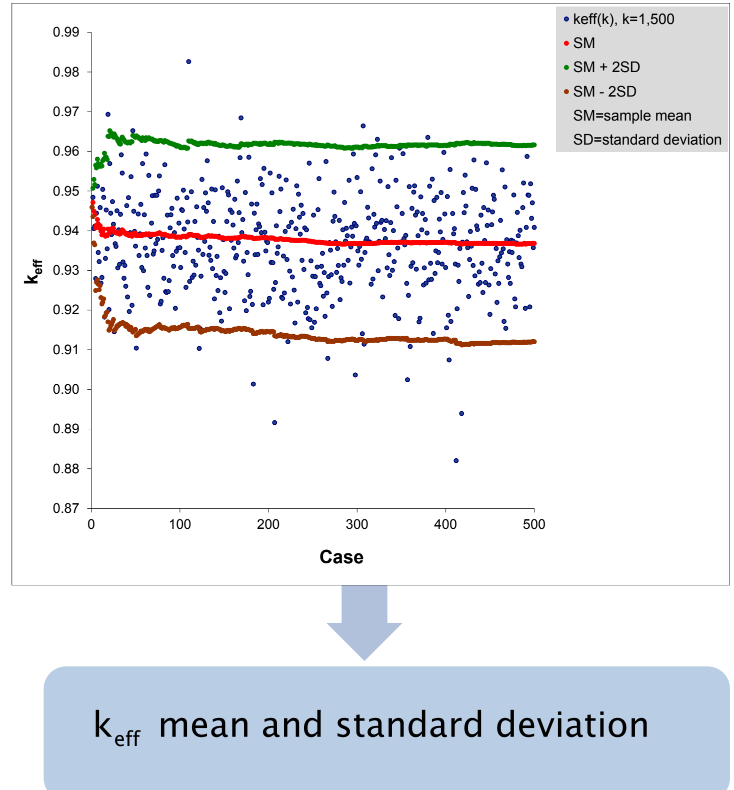
Isotope	No. of samples	Enrichment range (wt% ^{235}U)	Burnup range (GWd/MTU)	Isotope	No. of samples	Enrichment range (wt% ^{235}U)	Burnup range (GWd/MTU)
^{234}U	63	2.453–4.657	7.2–59.7	^{101}Ru	15	3.5–4.1	31.1–59.7
^{235}U	100	2.453–4.657	7.2–59.7	^{103}Rh	16	2.453–4.1	31.1–59.7
^{236}U	85	2.453–4.657	12.9–59.7	^{109}Ag	14	3.5–4.1	44.8–59.7
^{238}U	100	2.453–4.657	7.2–59.7	^{133}Cs	7	3.038 – 4.1	27.4–59.7
^{237}Np	44	2.453–4.657	16.0–59.7	^{143}Nd	44	2.453–4.657	16.0–59.7
^{238}Pu	85	2.453–4.657	12.9–59.7	^{145}Nd	44	2.453–4.657	16.0–59.7
^{239}Pu	100	2.453–4.657	7.2–59.7	^{147}Sm	32	2.453–4.657	23.7–59.7
^{240}Pu	100	2.453–4.657	7.2–59.7	^{149}Sm	28	3.5–4.657	23.7–59.7
^{241}Pu	100	2.453–4.657	7.2–59.7	^{150}Sm	32	2.453–4.657	23.7–59.7
^{242}Pu	99	2.453–4.657	7.2–59.7	^{151}Sm	32	2.453–4.657	23.7–59.7
^{241}Am	47	2.453–4.657	17.1–59.7	^{152}Sm	32	2.453–4.657	23.7–59.7
^{243}Am	48	2.63–4.657	17.1–59.7	^{151}Eu	21	3.5–4.657	23.7–59.7
^{95}Mo	15	3.5–4.1	31.1–59.7	^{153}Eu	27	2.453–4.657	23.7–59.7
^{99}Tc	25	2.453–4.1	16.0–59.7	^{155}Gd	27	2.453–4.657	23.7–59.7

MC Uncertainty Sampling Method

- Used to propagate isotopic bias and uncertainty to k_{eff}
- Provides an estimate of the bias and uncertainty in k_{eff} due to bias and uncertainty associated in the predicted fuel compositions
- Provides a realistic safety margin by allowing compensating positive and negative effects of isotopic bias on reactivity
- Enables depletion code validation directly with safety analysis models
- Not sensitive to the limited number of nuclides measured in individual fuel samples

MC Uncertainty Sampling Method Schematic





B&BU in k_{eff} based on the upper limit of the one-sided tolerance interval for 95% of the population and 95% confidence

Reference Numbers

- **Combined isotopic k_{eff} bias and bias uncertainty for the representative PWR SNF system model using ENDF/B-VII data**

	Actinides Only	Actinides and Fission Products
Burnup (GWd/MTU)	Δk_i	Δk_i
0-5	0.0145	0.0150
5-10	0.0143	0.0148
10-18	0.0150	0.0157
18-25	0.0150	0.0154
25-30	0.0154	0.0161
30-40	0.0170	0.0163
40-45	0.0192	0.0205
45-50	0.0192	0.0219
50-60	0.0260	0.0300

Recommendations

- ISG-8, Rev. 3, provides several methodologies that are considered acceptable for isotopic depletion validation
- In lieu of an explicit validation, the applicant may use the combined bias and bias uncertainty (Δk_i) values estimated in NUREG/CR-7108. These values may be used directly, provided that:
  - the applicant uses the same code and cross-section library as was used in NUREG/CR-7108,
  - the applicant uses the same or similar initial assumptions and code modeling options as was used in NUREG/CR-7108,
  - the applicant can justify that their design is similar to the cask system used as the basis for the NUREG/CR-7108 depletion validation, and
  - credit is limited to the specific nuclides listed in ISG-8, Rev. 3.

^{234}U	^{235}U	^{236}U	^{238}U	^{237}Np	^{238}Pu
^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh	^{109}Ag	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd	–	–

Criticality Validation Technical Approach

- Challenge:
  - Existing/available laboratory critical experiments (LCEs) do not have minor actinides & fission products (FPs) in proportions similar to actual SNF, and hence are not directly usable for validation

Criticality Validation Technical Approach

- B&BU based on comparison of calculated k_{eff} values and measured data from LCEs to the extent possible
  - Validation of principal actinides – utilize available LCE data

- Validation of FPs and minor actinides
  - Utilize available LCE data to estimate bias
  - Use nuclear data uncertainties and calculated application sensitivities to estimate potential biases for all relevant nuclides
  - Verify estimated biases through comparisons with calculated biases for cases where LCE data are available, including for the limited available FP LCE data

Estimating Bias Based on Nuclear Data Uncertainties

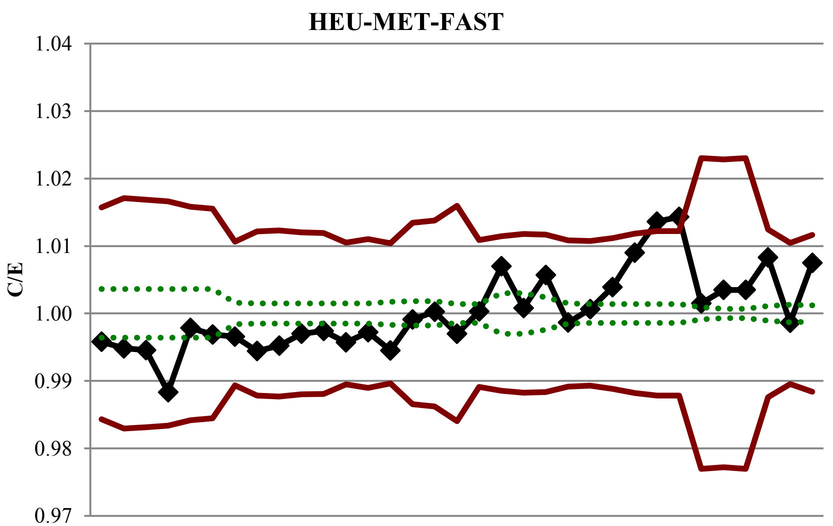
- Uncertainty in the system k_{eff} is propagated from the cross section uncertainty using the sensitivity coefficient:

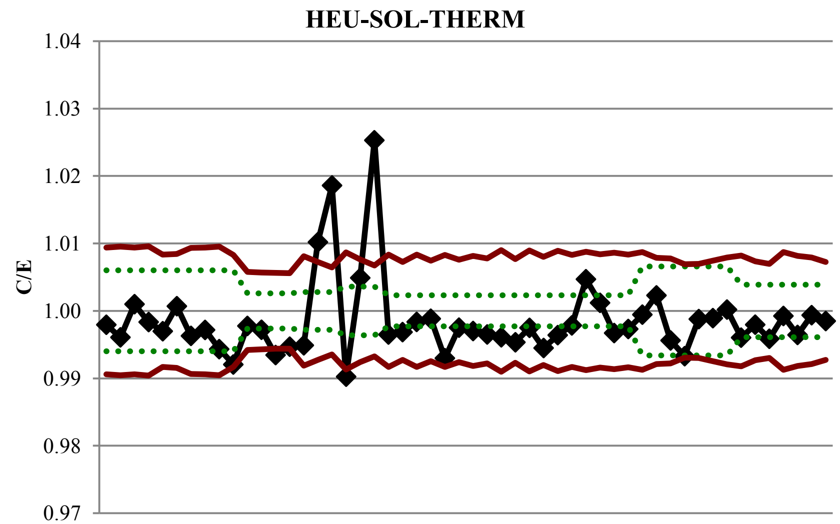
$$\sigma_{k_{eff}}\left(\frac{\%\Delta k}{k}\right)=\sigma_{\sigma}\left(\frac{\Delta\sigma}{\sigma}\right)\times S\left(\frac{\Delta k/k}{\Delta\sigma/\sigma}\right)*100\%$$

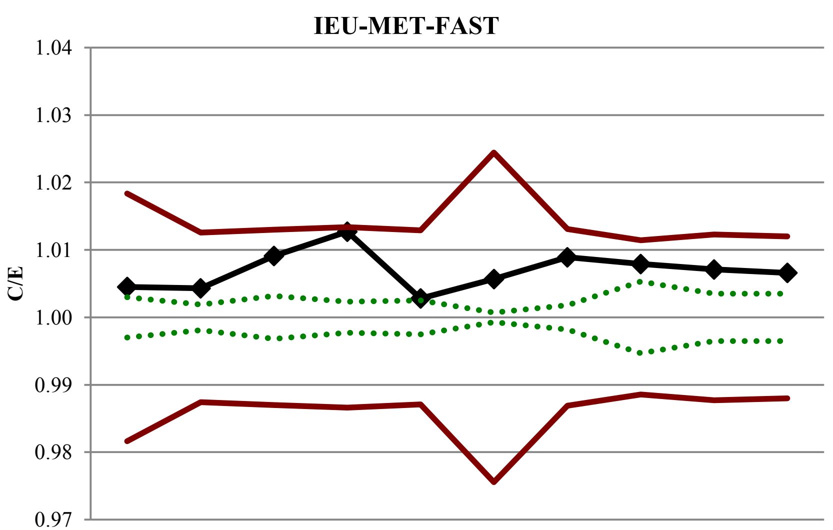
- Fundamental basis for this approach is that biases caused by nuclear data errors are bounded by the nuclear data uncertainties
- Uncertainty therefore gives an upper bound for the magnitude of the bias

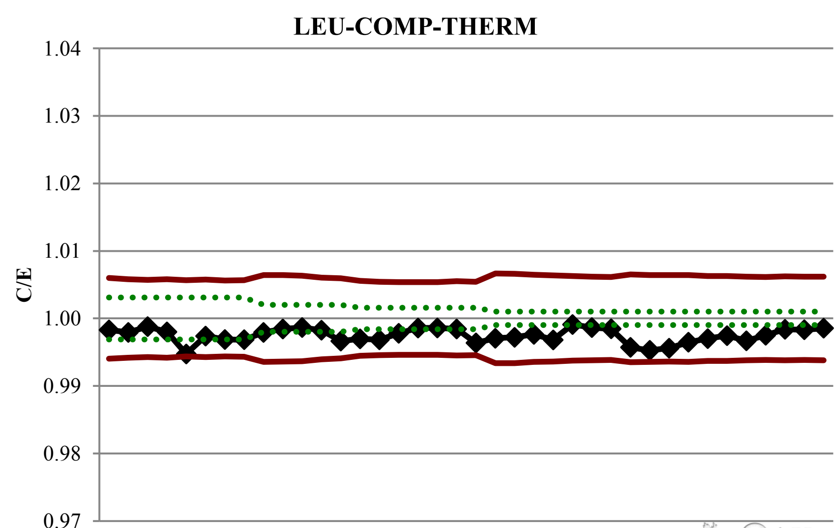
Examples confirming that computational bias is generally bounded by cross-section uncertainty

Computational Bias
Experimental Uncertainty
Cross-section Uncertainty









Example Results

Spent fuel pool and GBC-32 cask, fuel burned to 40 GWd/MTU

Uncertainty in k_{eff} due to uncertainty in nuclear data

	Uncertainty (%Δk/k)	
Uncertainty Source	**GBC-32**	**SFP**
All nuclides	0.512	0.491
Actinides-only	0.496	0.480
Structural Materials	0.111	0.073
Primary 6 FP	0.049	0.047
Next 10 FP	0.024	0.023
All Other FP & Actinides	0.037	0.044

Results

- Uncertainty due to nuclear data uncertainties investigated for SNF configurations as a function of burnup and a variety of other relevant parameters

- Uncertainty determined to be < 1.5% of the reactivity worth of the minor actinides and FPs in all cases considered

Recommendations (1/2)

- Validate principal actinides with available LCE data
  - ^{234}U, ^{235}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am
  - HTCs experiments and several in the International Handbook are useful for burned fuel

Recommendations (2/2)

- A conservative estimate for the combined bias and bias uncertainty associated with minor actinide and FP nuclides of 1.5% of their worth may be used. This estimate is appropriate provided the applicant:
  - uses the SCALE code system with the ENDF/B-V, ENDF/B-VI, or ENDF/B-VII cross section libraries,
  - uses the same or similar initial assumptions and code modeling options as were used in NUREG/CR-7109,
  - can justify that their design is similar to the cask system used as the basis for the NUREG/CR-7109 criticality validation, and
  - demonstrates that the combined minor actinide and fission product worth is no greater than 0.1 in k_{eff}

Closure

- Questions / Discussion

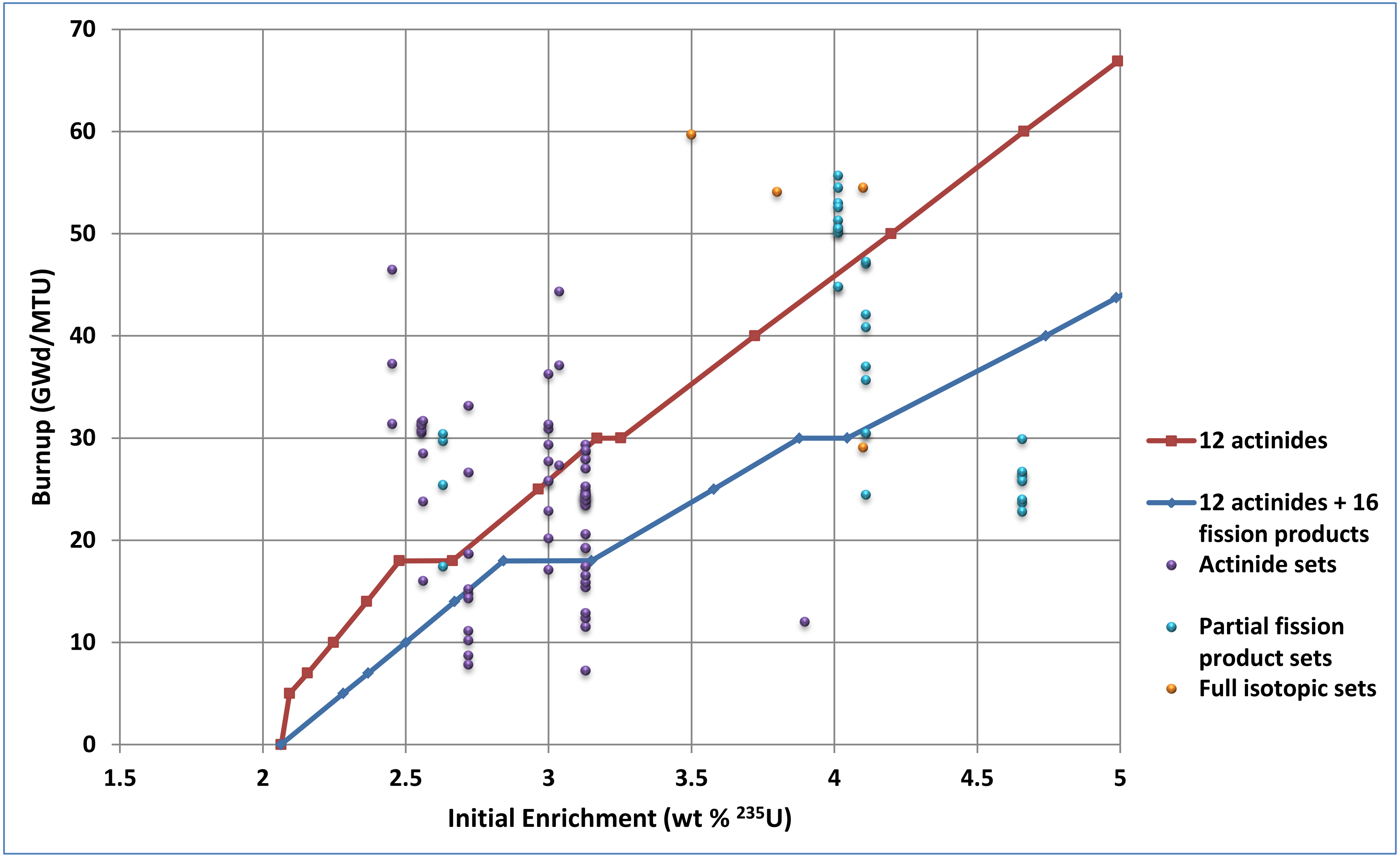
- Acknowledgment
  - This work was supported by the U.S. Nuclear Regulatory Commission (RES, NMSS, NRR, and NRO)

Nuclides Important to Burnup Credit

^{234}U	^{235}U	^{236}U	^{238}U	^{237}Np	^{238}Pu
^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{243}Am
^{95}Mo	^{99}Tc	^{101}Ru	^{103}Rh	^{109}Ag	^{133}Cs
^{143}Nd	^{145}Nd	^{147}Sm	^{149}Sm	^{150}Sm	^{151}Sm
^{152}Sm	^{151}Eu	^{153}Eu	^{155}Gd	–	–

- Credited nuclides based on their importance to SNF reactivity and on availability of RCA data
- Nuclides important to SNF reactivity previously evaluated in NUREG/CR-6665 & ORNL/TM-12973
- 12 major and minor actinide nuclides (all very long lived except for ^{241}Pu [$T_{1/2}$=14.4 y]; ^{238}Pu [$T_{1/2}$=87.7 y]; and ^{241}Am [$T_{1/2}$=432.7 y])
- 16 fission product (FP) nuclides (all stable or very long lived except for ^{151}Sm [$T_{1/2}$=90 y])

PWR RCA Data for Burnup Credit Nuclides



MC Uncertainty Sampling Method Steps

- Sample from the distribution models established for the bias associated with predicted isotopic concentrations
- Adjust the predicted isotopic concentrations using the sampled random values
- Use the adjusted isotopic concentrations in the criticality model and calculate k_{eff}
- Repeat the procedure for a statistically significant number of criticality calculations
- Determine the mean and standard deviation of the k_{eff} values
- Establish the B&BU in k_{eff} based on the upper limit of the one-sided tolerance interval for 95% of the population and 95% confidence

Validation Results for the GBC-32 Cask Model

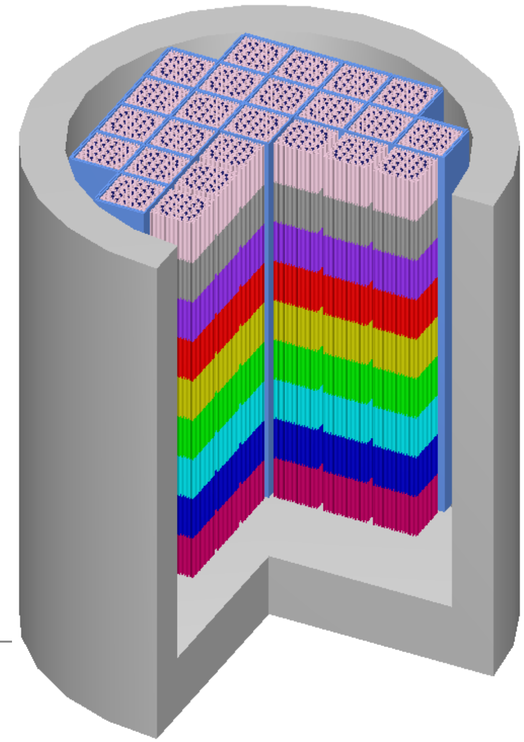
	Actinides only		Actinides and fission products	
Burnup[a] (GWd/MTU)	k_{eff} bias [c]	k_{eff} bias uncertainty [d]	k_{eff} bias [c]	k_{eff} bias uncertainty [d]
5	0.0042	0.0145	0.0040	0.0150
10	0.0040	0.0143	0.0039	0.0148
18	0.0036	0.0150	0.0037	0.0157
25	0.0047	0.0150	0.0023	0.0154
30	0.0052	0.0154	0.0031	0.0161
40	0.0059	0.0170	0.0012	0.0163
45[b]	----	----	0.0050	0.0205
50	0.0073	0.0192	----	----
60	0.0107	0.0260	----	----

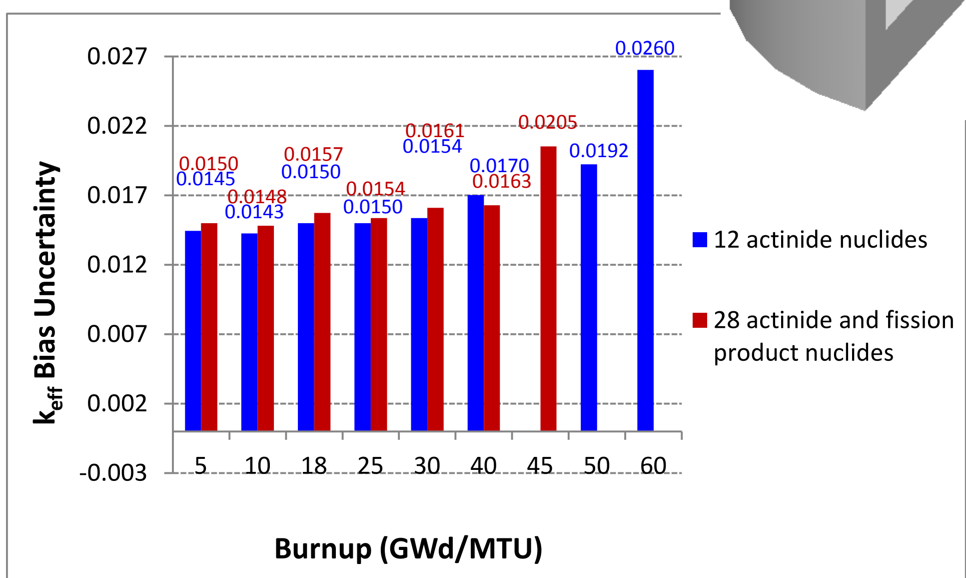
[a]Fuel initial enrichment values are such that the k_{eff} value based on the predicted nuclide compositions is 0.94.

[b]Initial enrichment is 5 wt% ^{235}U.

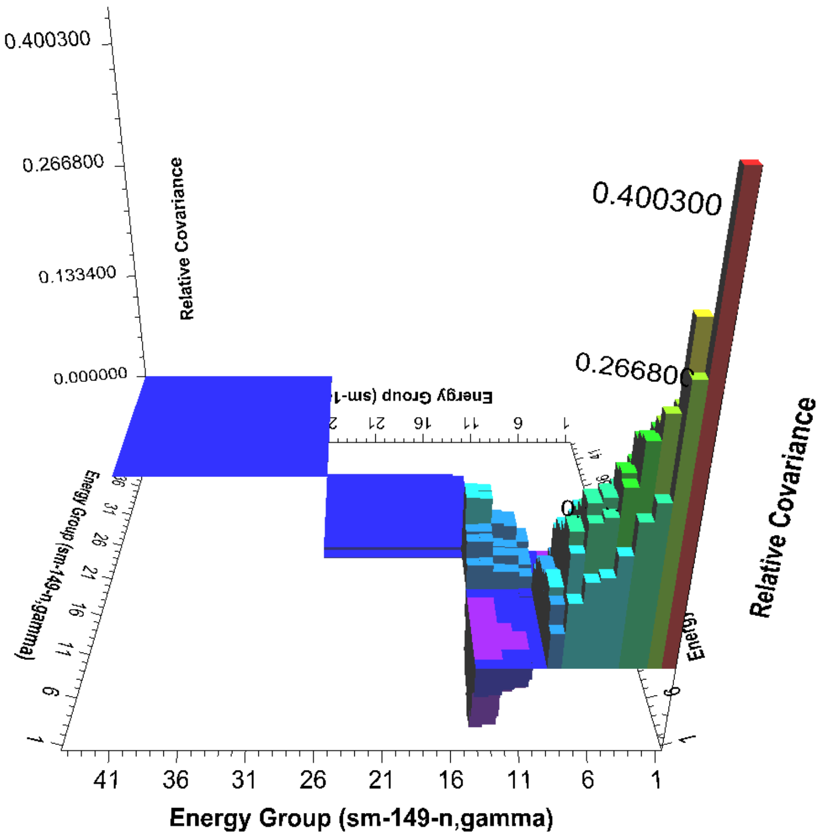
[c]Positive bias is typically not credited in criticality safety analyses.

[d]Based on the 95%/95% tolerance interval.

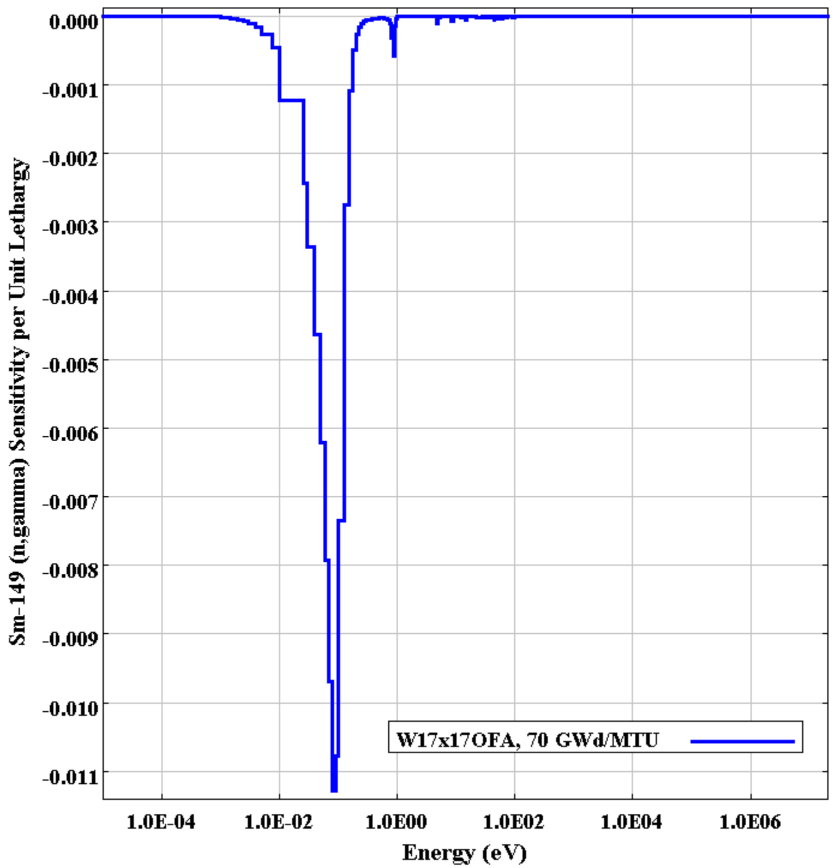




Uncertainty Analysis



Covariance Data $[(\Delta\sigma/\sigma)^2]$

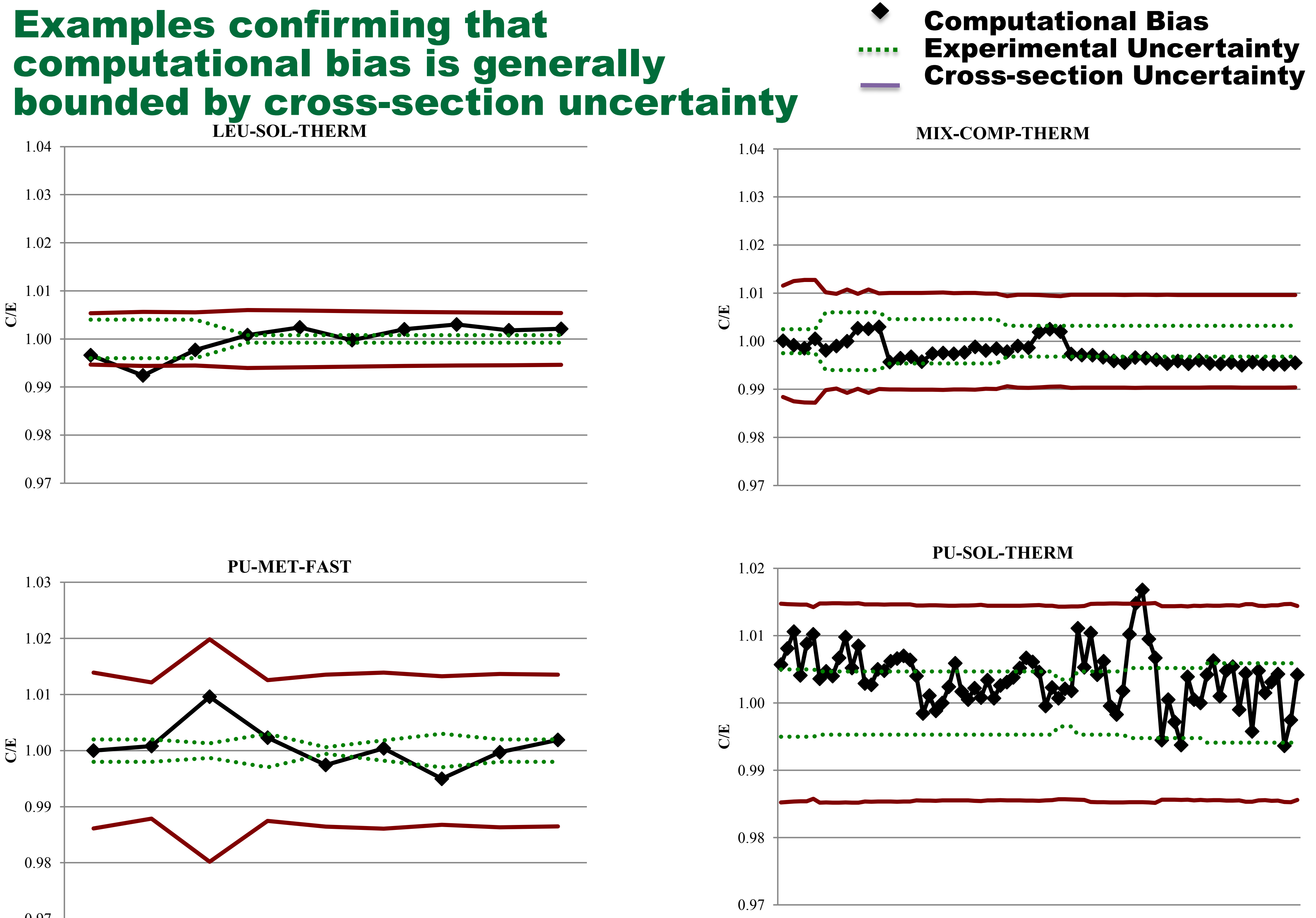


Sensitivity Data $\{(\Delta k/k) / (\Delta\sigma/\sigma)\}$

&

combined using appropriate matrix algebra to yield uncertainty in k_{eff} due to nuclear data uncertainties

Examples confirming that computational bias is generally bounded by cross-section uncertainty



[Break]

Public Comments on Draft ISG-8

- Received comments from:
  - Nuclear Energy Institute
  - Holtec International
  - Nuclear Consultants.com
- Major comments and proposed resolutions

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Turn ISG into a Reg Guide

- ISG consolidates review guidance in one place (as opposed to SRP referencing RG)
- ISG format allows more flexibility to modify guidance to reflect new methodologies (EPRI, BWR burnup credit)
- ISG and Appendix text incorporated directly into criticality chapters of SRPs
- May consolidate guidance into RG at a later date

Flexibility for alternative validation methodologies

- Validation methodology recommended by ISG-8 represents one method that has been reviewed in detail by the staff and found to be acceptable
- ISG does not exclude alternative methodologies
- Will revise ISG text to state that alternative methodologies should be considered on a case-by-case basis

Remove burnup measurement

- Measurement recommendation maintained in ISG as an alternative to misload analysis/admin procedures
  – Allows flexibility to applicants if the misload analysis criteria is too restrictive for their specific design
  – Future measurement techniques may make measurement option more appealing

Administrative loading procedures

- Industry proposed procedures should already be incorporated into cask and site loading procedures; not specific to burnup credit; e.g.:
  - Verify the identity of the fuel assembly prior to loading it into the cask
  - Verify the identity of the fuel assemblies loaded into the cask prior to closing the cask
  - Verify the burn-up values of each fuel assembly to be loaded into the cask from a source QA record prior to loading the first assembly

Administrative loading procedures (cont'd)

- ISG procedures are intended to be *additional* procedures for burnup credit cask loading, targeted at reducing likelihood or consequences of high-reactivity misload, e.g.:
  - Assurance that there is no fresh fuel in the pool during system loading
  - Verification of the location of high reactivity fuel (i.e., severely underburned fuel) in the spent fuel pool both prior to and after loading
  - Independent, third-party verification of the loading process
- *Recommended* procedures; list not intended to be all-inclusive

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Individual administrative procedures

- Fresh fuel procedures redundant – assure no fresh fuel in the pool *or* qualitative verification of burnup
- Pool audit within one year of loading burdensome and duplicates some 10 CFR 74 MC&A requirements – Verification of high reactivity assemblies in pool prior to and after loading should be sufficient
- Revise pool audit for loading to QA audit of already loaded canisters prior to shipment
- Clarify intent of other recommended procedures

Misload analysis recommendations

- 0.02 Δk_m standard for misload analyses
- Single fresh fuel assembly is acceptable, however:
  - procedures should prevent fresh fuel misloads
  - ISG recommends “reasonably bounding” single misload (95/95 level)
- Multiple assemblies 25% underburned is more simple, however:
  - Depends on loading curve (could be less restrictive than proposed in ISG)
  - ISG recommendation (bounds 90% of total inventory) allows this analysis to be omitted if the loading curve already encompasses 90% of fuel

Other Comments

- Credit for additional isotopes:
  - Will modify to state that additional isotopes may be credited, provided the bias and bias uncertainty is quantified
- BWR burnup credit:
  - Upcoming RES user need for BWR burnup credit
  - Will revise ISG to state that BWR burnup credit analyses to be reviewed on case-by-case basis
- Applicability to non-intact fuel
  - Will revise this section to include undamaged and damaged fuel (per ISG-1), provided fuel reconfiguration and any additional uncertainties are considered

Other Comments (cont'd)

- Separate bias and bias uncertainty terms:
  - β_i = bias in k_{eff} due to depletion code; added to calculated k_{eff}
  - Δk_i = uncertainty in β_i; statistically combined with other calculation uncertainties
  - Δk_x = uncertainty in k_{eff} due to uncertainty in minor actinide and fission product cross-section data; treated as bias added to calculated k_{eff}
- k_{eff} bias for other criticality codes:
  - Δk_x = 1.5% of minor actinide and fission product worth for SCALE code system with ENDF/B-V, -VI, or –VII data
  - Can use same number for other qualified codes with same data, provided minor actinide and fission product worth is comparable to that calculated with SCALE

Conclusions and Next Steps

- ISG-8, Revision 3 extends the technical basis for burnup credit to fission products and minor actinides
- Provides alternative to confirmatory burnup measurement
- Generally well-received by industry, with some comments
- Will resolve comments as discussed, revise the ISG, and present to full ACRS in September

Administrative loading procedures

ISG:

- Assurance that there is no fresh fuel in the pool during system loading,
- Verification of the location of high reactivity fuel (i.e., severely underburned fuel) in the spent fuel pool both prior to and after loading,
- Qualitative verification that the assembly to be loaded is burned (visual or gross measurement),
- Confirmation that an audit of the pool inventory has been performed no more than one year prior to the time of loading,
- Quantitative measurement of any fuel assemblies without visible identification numbers,
- Independent, third-party verification of the loading process, and
- Minimum required soluble boron concentration in pool water during loading and unloading.

Industry:

- Verify the identity of the fuel assembly prior to loading it into the cask
- Verify the identity of the fuel assemblies loaded into the cask prior to closing the cask
- Verify the burn-up values of each fuel assembly to be loaded into the cask from a source QA record prior to loading the first assembly
- Reduce the verified reactor record burn-up value by uncertainty in the record value, this is the burn-up value to be used for loading acceptance
- Verify that each fuel assembly to be loaded into the cask satisfies the loading requirements prior to loading the first assembly
- Develop and perform procedures/processes in accordance with the QA program
- Verify that the soluble boron concentration in the pool and cask is greater than the minimum required prior to cask loading

[EPRI Presentation from Albert Machiels]

NEI Presentation from Marcus Nichol]

[Committee Discussion]

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EPRI Work Relevant to Burnup Credit

Albert Machiels
Senior Technical Executive

ACRS Subcommittee on Radiation Protection and Nuclear Materials
July 10, 2012 Meeting

Contents

- *Introduction*
- *Transportation of Spent High Burnup Fuel*
- *Probability of Criticality Event During Transportation*
- *Burnup Credit Validation*
- *Summary*
- *References*

Introduction – Criticality Safety and Burnup Credit

- Criticality Safety
  - Standards & methodologies were originally developed for the front end of the fuel cycle with pure materials
  - Spent fuel is a challenge
    - ORIGEN follows >2000 nuclides
- "Fresh fuel assumption"
  - Significant conservatism
  - Low-capacity storage and transport systems (more systems, more operations, increased $)
    - May result in less overall safety (radiological ←→ non-radiological)
- Burnup credit (BUC): Getting credit for the reduced reactivity of spent fuel compared to fresh fuel
  - "Actinide-only", "Actinide + subset of fission products", "Full BUC"

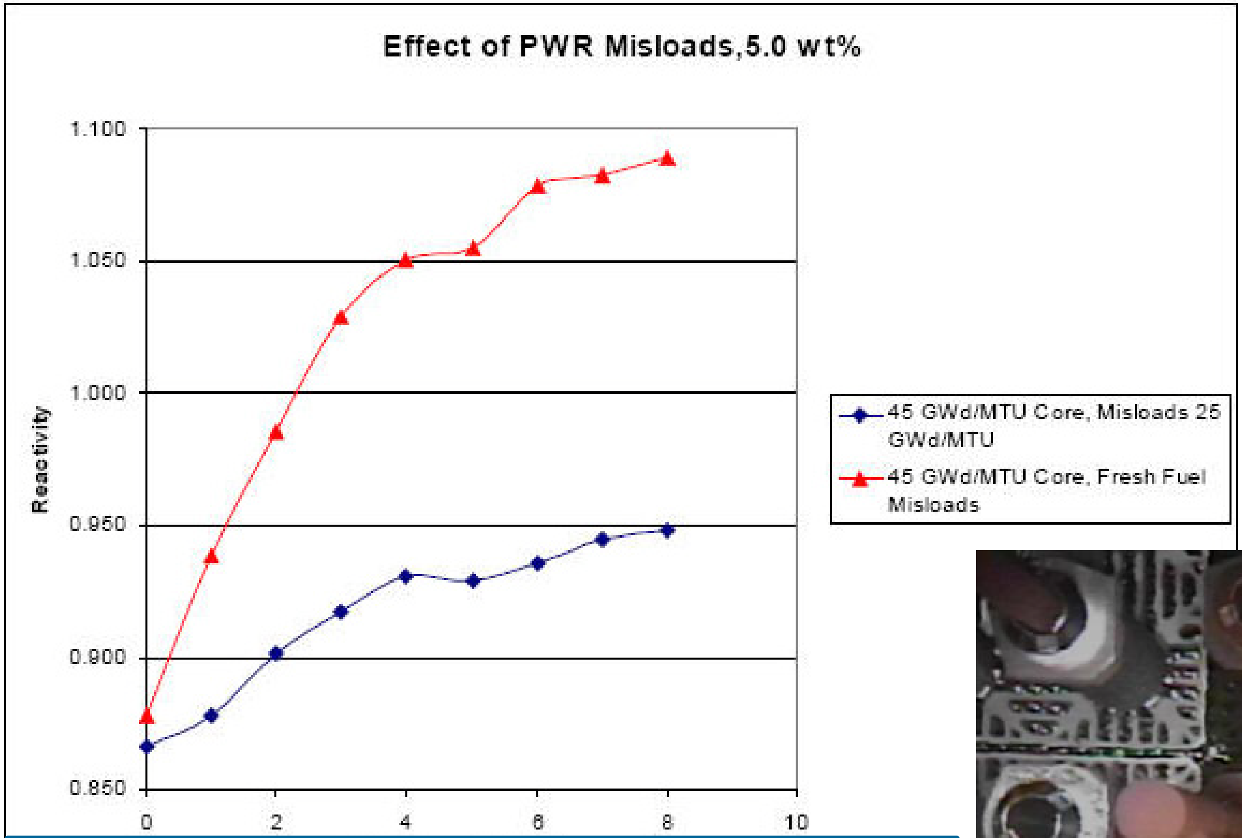
Transportation of Spent High Burnup Fuel

- Key regulatory issue: maintaining sub-criticality under *accident* conditions
- NRC positions:
  - Burnup <45 GWD/MTU: normal assembly configuration
  - Burnup >45 GWD/MTU: fuel reconfiguration cannot be ruled out ➔ “moderator exclusion” or “analytical simulation” option
- Observations
  - High-burnup fuel burned to “design burnup” has low residual nuclear reactivity
  - Should significant reactivity remains (“under-burned”), normal configuration could be assumed
  - With burnup, as cladding properties ➘,nuclear reactivity ➘

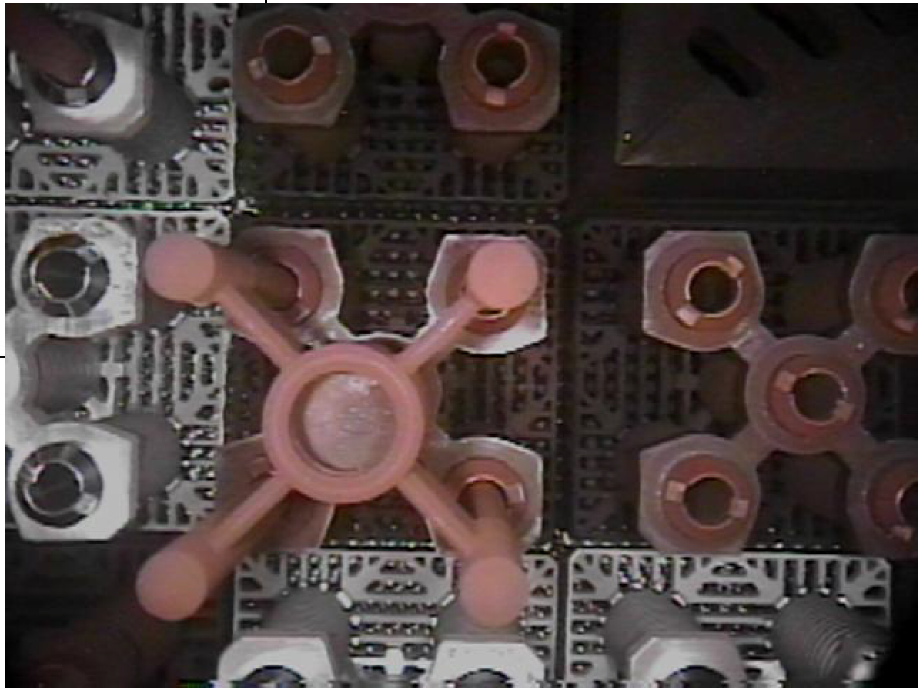
Impact of Spent PWR Fuel Assembly Reconfiguration

- NUREG/CR-6835 (2003)
  - "*Although the scenarios considered go beyond credible conditions, they represent a theoretical limit on the effects of severe accident conditions*" (Ref.: page 1)
  - Modest increases (<0.05) in k_{eff} in "GBC-32" (Ref.: middle column of Table 6)
- EPRI 1015050 (2007)
  - **k_{eff} is much more likely to decrease than increase**
    - Changes from normal PWR assembly configuration tend to decrease nuclear reactivity

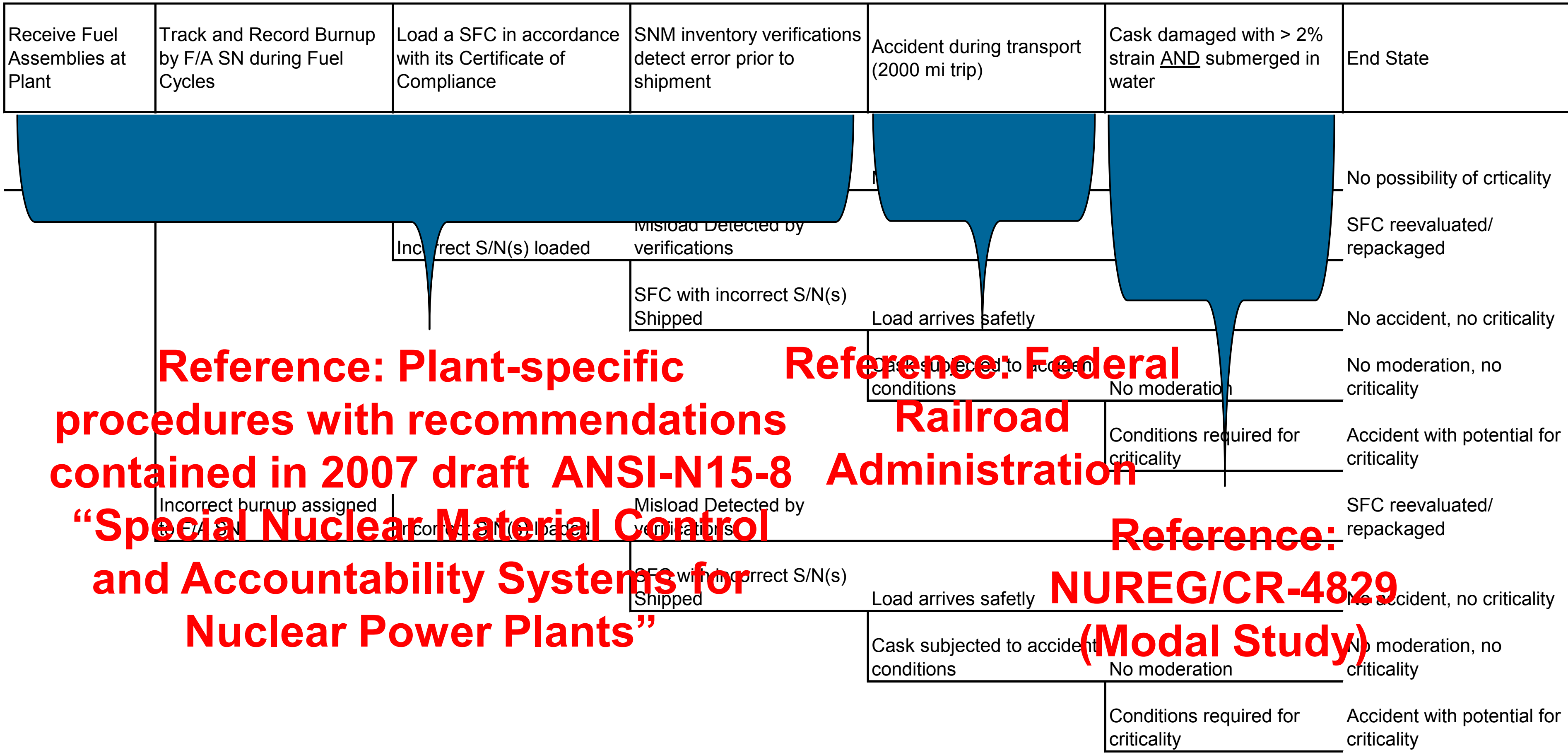
Impact of Misloads
EPRI Report 1003418 (December 2003)



- **Misloading of fresh fuel is not credible**
- Multiple misloading of significantly underburned fuel is required to introduce enough nuclear reactivity



Probability of Critical Event During Transportation EPRI Report 1016635 (December 2008)



Likelihood of a potential criticality event during a 2000-mile railroad shipment of a cask designed for 32 PWR assemblies: ~1x10^{-16}/shipment

Burnup Credit Validation

- Main thrust: conservatively estimate loss of nuclear reactivity as a function of burnup (range: 0 to 60 GWD/MTU)
  - Including uncertainty of the estimate
- Alternate approach
  - Based on in-reactor measurements (flux maps)
    - Required as part of routine monitoring of power plant operations
  - Cooperative effort involving Duke-Energy, Studsvik Scandpower, and Dr. Dale Lancaster
    - Principal Investigator: Prof. Kord Smith (MIT)

Flux Maps: Individual Assembly Reaction Rates

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							0.499 0.491 0.008			0.483 0.483 0.000					
2			0.369 0.389 -0.020			0.998 1.008 -0.006		0.996 0.991 0.004							
3								1.195 1.175 0.020		1.081 1.069 0.012		0.998 0.985 0.012		0.370 0.382 -0.012	
4		0.610 0.627 -0.017	0.997 0.994 0.003					1.315 1.323 -0.008							
5					1.309 1.298 0.010				1.330 1.331 -0.001		1.309 1.304 0.005		1.177 1.170 0.007		
6	0.484 0.479 0.004		1.082 1.071 0.011			1.331 1.332 -0.001									
7				1.191 1.179 0.011			1.276 1.287 -0.011			1.236 1.224 0.011			1.270 1.274 -0.004		
8	0.475 0.478 -0.003		1.196 1.183 0.013		1.232 1.224 0.008		1.209 1.206 0.003					1.317 1.319 -0.002	1.195 1.178 0.017	0.996 0.995 0.001	
9		0.958 0.951 0.007							1.278 1.272 0.006		1.332 1.317 0.014				0.497 0.501 -0.004
10												1.296 1.298 -0.002			
11	0.355 0.372 -0.018				1.306 1.323 -0.017			1.233 1.212 0.020			1.308 1.305 0.004				0.354 0.370 -0.015
12						1.294 1.305 -0.011			1.192 1.182 0.009			1.170 1.173 -0.003			
13			0.876 0.876 0.000		1.170 1.188 -0.018			1.196 1.183 0.013						0.369 0.383 -0.014	
14			0.369 0.395 -0.026				0.956 0.938 0.018			1.000 0.996 0.003		0.611 0.642 -0.032			
15					0.354 0.366 -0.012			0.475 0.476 -0.001						**Reaction Rate** **SIMULATE-3** **MEASURED** **S3-MEAS**	

- Miniature fission chambers are inserted in the central instrument tubes of selected assemblies
- This is a high precision (<1% statistical error) measurement of the core-wide distribution of fission rates
- BOC calculations required by NRC to be within a prescribed tolerance of measurement - to assure core loading
- Required every 30 days by NRC to guarantee that the core is operating within design margins

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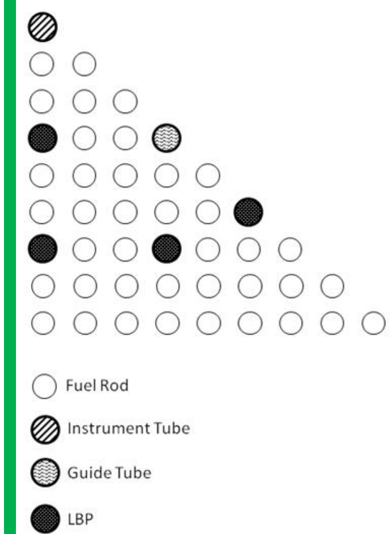
11 Reactivity Decrement Benchmarks for 17 x 17 PWR Fuel Designs

	Table 13.1 Benchmark Lattice Cases
1	3.25% Enrichment
2	5.00% Enrichment
3	4.25% Enrichment
4	off nominal pin diameter depletion
5	20 LBP depletion
6	104 IFBA depletion
7	104 IFBA plus 20 LBP depletion
8	high boron depletion=1500 ppm
9	branch to hot rack (150F coolant/fuel)=338.7K
10	branch to high rack boron = 1500 ppm
11	high power depletion*(power, coolant/fuel temp)

Measured Reactivity Decrement						
	Burnup (GWd/T)					
Case	10	20	30	40	50	60
1	-0.1329	-0.2339	-0.3211	-0.3956	-0.4554	-0.5002
2	-0.1146	-0.2021	-0.2806	-0.3545	-0.4238	-0.4867
3	-0.1223	-0.2157	-0.2990	-0.3758	-0.4445	-0.5029
4	-0.1207	-0.2176	-0.3075	-0.3931	-0.4715	-0.5385
5	-0.2045	-0.2335	-0.2998	-0.3717	-0.4372	-0.4932
6	-0.1736	-0.2215	-0.2968	-0.3726	-0.4418	-0.5009
7	-0.2524	-0.2418	-0.2981	-0.3686	-0.4343	-0.4910
8	-0.1216	-0.2129	-0.2932	-0.3662	-0.4310	-0.4860
9	-0.1237	-0.2171	-0.2998	-0.3756	-0.4432	-0.5005
10	-0.0967	-0.1784	-0.2530	-0.3217	-0.3826	-0.4335
11	-0.1235	-0.2149	-0.2945	-0.3664	-0.4299	-0.4838

Case 5: 20 LBP depletion



Fuel Material Description

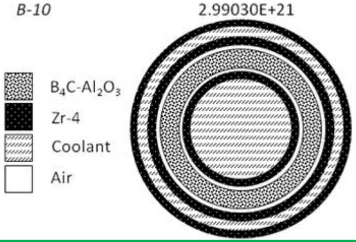
Material Density	10.340 (g/cm3)
Fuel Temperature	900 K
Nuclide	**Number Density**
U-235	9.92536E+20
U-234	7.97571E+18
U-238	2.20709E+22
O	4.61429E+22

Lumped Burnable Poison (WABA)

Annular clad IR	0.2860 cm
Annular clad OR	0.3390 cm
Active region IR	0.3530 cm
Active region OR	0.4040 cm
Inner clad IR	0.4180 cm
Inner clad OR	0.4840 cm
Outer clad IR	0.5610 cm
Outer clad OR	0.6120 cm

Active Region

Material Density	3.65 (g/cm3)
Boron Loading	6.03 mg/cm B-10
Nuclide	**Number Density**
C	1.40923E+21
O	6.23784E+22
Al	4.15904E+22
B-10	2.99030E+21



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Comparison between “Draft ISG-8, Rev 3” and “Depletion Benchmarks”

Bias Plus Uncertainty in k		
Burnup	Draft ISG-8, Rev 3	Depletion Benchmarks
10	0.015	0.008
20	0.016	0.008
30	0.016	0.008
40	0.022	0.008
50	0.030	0.008
60	0.030	0.008

- Both results are for SCALE and ENDF/B-VII
- **“Depletion Benchmarks” uncertainty includes all nuclides** rather than the more limited number of nuclides allowed by Draft ISG-8, Rev 3
- “Depletion Benchmarks’ value is dominated by measurement uncertainties, therefore not burnup dependent. Draft ISG-8, Rev 3 values dominated by chemical assay uncertainties

Summary

- Alternative Approach Relying on Full Burnup Credit
  - Experimental depletion benchmarks
  - Applicable to storage (wet and dry), transportation, disposal
- Spent High-burnup Fuel Transportation
  - Burnup credit is a high priority topic
    - Increased cask capacity (32 vs. 24 assemblies)
    - Loading a greater percentage of spent fuel population
  - Extremely low probability for the potential of a critical event during transportation of commercial spent high-burnup fuel

> *"Removal of extreme conservatism can result in an overall improvement in safety by balancing criticality risks with other operational risks"* [C. Parks *(ORNL), Closing Review Session of 2011 International Conference on Nuclear Criticality (ICNC2011)]*

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Back-up Slides

Fuel Reconfiguration – Beyond Credible Scenarios NUREG/CR-6835 (September 2003)

Table 6: Maximum increase in k_{eff} for each fuel failure scenario*

Scenario	MPC-24 *(fresh fuel)*	GBC-32 (45 GWd/MTU)	MPC-68 *(fresh fuel)*
Single missing rod	0.0015	<0.0010	0.0036
Multiple missing rod	0.0140	0.0130	0.0120
Cladding removed from all fuel rods	0.0468	0.0349	0.0441
Fuel rubble (no cladding)	0.0563	0.0233	0.1149
Assembly slips 20 cm above or below neutron poison panels	0.0021	0.0435	0.0362
Variation in pitch (without cladding)	0.0703	Not calculated	0.1225

***** *"Although the scenarios considered go beyond credible conditions, they represent a theoretical limit on the effects of severe accident conditions" (NUREG/CR-6835, p. 1)***

Quantification of Human Failure Events Leading to a Misloaded Dry Spent Fuel Cask

FMS	Refueling Engineer (RE)	FMS Supervisor	Refueling Engineer and Crew	Refueling Engineer and Rep. from Nuclear Oversight	Third Party	Third Party	
Select F/As for DSC in compliance with CoC	Prepare FMSDS from DSC Fuel Loading Pattern Form	Verify FMSDS S/Ns and DSC locations against DSC Fuel Loading Pattern Form	Individually Transfer 32 F/As from SFP to DSC	Verify F/A S/N against DSC Fuel Loading Pattern Form using 3-Way communication	Independent Verification via Review of Video against DSC Fuel Loading Pattern Form	Perform independent SNM inventories and/or audits prior to shipment	Scenario Likelihood
HASEL1	HAFMS1	HRFMS1	HATRN1	HRDSC1	HRDSC2	HRSEL1	
9.998E-01	9.9E-01	NA	9.9E-01	NA	NA	NA	OK
			8.7E-03	9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	1.9E-08
	1.3E-02	9.3E-01	9.9E-01	NA	NA	NA	OK
			8.7E-03	9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	2.3E-10
		6.6E-02		9.97E-01	NA	NA	OK
				2.8E-03	9.3E-01	NA	OK
					7.4E-02	9.9E-01	OK
						1.04E-02	1.8E-09
2.50E-04						9.9E-01	OK
						1.04E-02	2.6E-06

From EPRI Report 1016635

Total likelihood of a spent fuel cask shipment with one or more misloaded F/As = 2.6E-06

Importance of a Centralized Accounting System

- ANSI 15-8, *Special Nuclear Material Control and Accounting Systems for Nuclear Power Plants,* provides reasonable guidelines to record, track, and verify F/A burnup in a centralized accounting system
- Core follow software provides accurate information of the burnup of fuel assemblies. Each spent F/A can be directly associated with its burnup history over multiple fuel cycles
- At any time before a spent fuel cask is shipped
  - F/A burnup and SNM content can be verified against in-core detector measurements and core follow calculations for reactor controls by F/A serial number.
  - Video of F/A serial numbers during cask loading provides ability to independently verify proper loading

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Train Accident Initiating Events

Case Study Number	Case Study Initiating Event Description From EPRI Report 1016635	Point Estimate Frequency (Events/ Train-Mile)
1	All Train Accidents per Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	4.33E-06
2	Freight Train Accidents per Freight Train-Mile (All Accidents, All Speeds, All Track Classes), 2000 - May 2006.	2.67E-06
3	Freight Train Accidents per Freight Train-Mile (Accidents with Primary or Secondary Derailments, All Speeds, All Track Classes), 2000 - May 2006.	2.25E-06
4	Freight Train Accidents per Track Class 3+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 30 MPH, 2000 - May 2006.	6.51E-07
5	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, All Speeds, All Track Classes), 2000 - May 2006.	3.06E-07
6	Freight Train Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 30 MPH, Track Class 3+), 2000 - May 2006.	8.45E-08
7	HAZMAT Freight Train Primary and Secondary Derailment Accidents per Track Class 4+ Freight Train-Mile (using Table 2-4 of Ref. 8) with Speed ≥ 60 MPH, 2000 - May 2006.	1.05E-08
8	Freight Train Primary and Secondary Derailment Accidents per Freight Train-Mile (Accidents with HAZMAT Car Damage, ≥ 60 MPH, Track Class 4+), 2000 - May 2006.	8.01E-09

Burn-up Credit for Spent Nuclear Fuel Storage Casks and Transport Packages

Industry Perspective

Marc Nichol

Nuclear Energy Institute

ACRS Sub-Committee Meeting

July 10, 2012

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Industry feedback on ISG-8 Revision 3

- Generally a large improvement from revision 2
- Opportunity for further improvements through flexibility and risk insights
- Five major industry recommendations
  – Burn-up verification
  – Depletion validation – alternative methods
  – Depletion validation – additional isotopes
  – Burn-up credit – applicability to BWR
  – Dual uses of guidance

Improvements in revision 3

- Greater benefit to utilizing burn-up credit
  - Now includes limited set of fission products
  - Range extended from 50 GWd/MTU to 60 GWd/MTU
- Use of risk insights
  - Smaller administrative margin for misload analyses
- Improved flexibility
  - Alternative to in-pool measurements

Burn-up verification

- Verification method should most effectively and efficiently address the situations that could lead to a misload
  1. Loading the wrong fuel assembly
  2. Calculating a burn-up value higher than actual
  3. Assigning the wrong burn-up value to a fuel assembly

Element	Draft ISG-8 use	Industry recommended use
Burn-up Measurement	Primary method of verification	Do not use
Misload analyses	Alternative to measurements	Defense-in-depth to Admin procedures
Admin procedures	Defense-in-depth to Misload analyses	Primary method of verification

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In-pool burn-up measurements

- Should be eliminated from guidance
- Do not address
  1. Loading the wrong fuel assembly[1]
  3. Assigning the wrong burn-up value to a fuel assembly
- Less effective than admin procedures for
  2. Calculating a burn-up value higher than actual
- Reactor records are very accurate (within ~2%)
- Reactor records calibrated with in-core measurements
- In-pool measurements use reactor records to calibrate
- In-pool measurements are inaccurate, and problematic to implement

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1) Numbers represent potential misload categories from previous slide.

Administrative procedures

- **Most effective means to address potential misloads**
- **Industry recommends comprehensive set of procedures that adequately address potential types of misload**
  1. ***Verify identity of fuel prior to loading***
  2. ***Verify identity of fuel prior to closing cask***
  3. ***Verify burn-up value from source QA record***
  4. ***Reduce reactor record by associated uncertainty***
  5. ***Verify fuel meets loading criteria***
  6. ***Develop/perform procedure/process according to QA program***
  7. ***Verify soluble boron greater than minimum prior to loading***

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Administrative procedures (cont'd)

- Some procedures are not recommended
  - *Verify location of high reactive fuel in pool* – less effective and more resources than industry recommended procedure
  - *Qualitative (e.g. visual)* – mitigates consequence only
  - *No fresh fuel in pool* – mitigates consequence only, duplicative
  - *Pool inventory audit* – duplicate/supersede other regulations
  - *Independent 3rd party* – inconsistent with NRC's QA requirements
  - *Quantitative measurement if no ID* – not anticipated, handle as exceptional case, not as something expected to be routine

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Misload analyses

- Do not reduce chance of misload, only manages consequences
- Should be simple and straightforward for
  - CoC holders to analyze
  - Licensees to verify
- Industry recommended assumptions[1]
  - Single misload – most reactive fresh fuel
  - Multiple misload – 50% with 25% under-burned
- Guidance should allow other approaches, if justified
  - E.g. if "no fresh fuel in pool" is elected, then single misload could assume a slightly burned assembly

1) There was insufficient time in the review of draft ISG-8 revision 3 to propose optimized assumptions for misload analyses. Therefore, these assumptions are overly conservative. With additional time, it may be possible to develop generic, less penalizing assumptions.

Depletion validation – flexibility

- Currently endorses ORNL developed methodology[1]
  - Overly conservative, due to measurement uncertainties
  - Limited isotopes available for credit
  - Complex
  - Requires significant effort to implement
- Guidance could be more efficient if flexibility added
  - Method to credit additional isotopes if data available
  - Considerations for proposing alternative methods[2]
- Potential application of burn-up credit for BWR

1) Documented in NUREG/CR-7108 and NUREG/CR-7109
2) For example, EPRI reports 1022909 and 1025203 detail an alternative approach for burn-up validation and were submitted to NRC/NRR on June 4, 2012 for review and endorsement.

Conclusions

- **Revision 3 is a significant improvement over revision 2**
- **Industry recommends improvements to burn-up verification for greater effectiveness and efficiency**
- **Industry recommends improvements to burn-up validation for greater flexibility**