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UNITED STATES NUCLEAR REGULATORY COMMISSION'S
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUELS, MATERIALS AND STRUCTURES SUBCOMMITTEE

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THURSDAY

MAY 18, 2023

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The Subcommittee met via hybrid in-person and Video Teleconference, at 8:30 a.m. EDT, Ron Ballinger, Chairman, presiding.

COMMITTEE MEMBERS:

RONALD G. BALLINGER, Chair

VICKI BIER, Member

VESNA DIMITRIJEVIC, Member

GREGORY HALNON, Member

WALT KIRCHNER, Member

JOSE MARCH-LEUBA, Member

DAVID PETTI, Member

JOY L. REMPE, Member

MATTHEW SUNSERI, Member

ACRS CONSULTANT:
DENNIS BLEY

DESIGNATED FEDERAL OFFICIAL:
CHRISTOPHER BROWN

ALSO PRESENT:
HATICE AKKURT, EPRI
MARKUS BURKARDT, Dominion
NATHAN GLUNT, EPRI
ROBERT HALL, EPRI
CRAIG HARRINGTON, EPRI
AYLIN KUCUK, EPRI
SCOTT MOORE, ACRS
KURSHAD MUFTUOGLU, EPRI
DAVID PERKINS, EPRI
AL SANTOS, NEI
FRED SMITH, EPRI
DAN WELLS, EPRI
ERICH WIMMER, EPRI
SURESH YAGNIK, EPRI

C-O-N-T-E-N-T-S

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

8:30 a.m.

CHAIR BALLINGER: Okay, the meeting will now come to order. This is a meeting of the Fuels Materials and Structures Subcommittee of the Advisory Committee on Reactor Safeguards. I'm Ron Ballinger, chairman of today's subcommittee meeting. ACRS members present are Jose March-Leuba, Matt Sunseri, Dave Petti, Joy Rempe, Vicki Bier, and Gregory Halnon. And remotely we have Walt Kirchner, and Vesna Dimitrijevic.

And we may have Dennis Bley, one of our consultants, I don't know for sure.

MEMBER SUNSERI: He's there.

CHAIR BALLINGER: He's there, okay. And I'm probably missing somebody, and I'll be chastised for that, but nonetheless, here we go. Chris Brown of the ACRS staff is the designated federal official for this meeting. During today's meeting, the subcommittee will receive a fuels information slash update from EPRI. The subcommittee will hear presentations by, and hold discussions with EPRI, and other interested persons regarding this matter.

This meeting is open to the public. The rules for participation in all ACRS meetings were announced in the Federal Register on June the 13th, 2019. U.S. NRC public website provides the ACRS charter, bylaws, agendas, letter reports, and full transcripts of all full, and subcommittee meetings including the slides. The agenda for this meeting was posted there, along with the MS Teams link.

We have received no written statements, or requests to make oral statements from the public. Subcommittee will gather information, analyze relevant issues, and facts, informing proposed positions, and actions as appropriate for deliberation by the full committee. A transcript of the meeting is being kept, and will be made available. Today's meeting is being held in person, and over Microsoft Teams.

There is also a telephone bridge line, and an MS Teams link allowing participation of the public. When addressing the subcommittee the participants should first identify themselves, and speak with sufficient clarity, and volume, so that they may be readily heard. When not speaking, we request that participants mute your computer microphone, or other phone by pressing star six.

This is the second of four meetings with EPRI. We've already covered the materials reliability issues core type materials on March the 22nd, and I might add that was an outstanding meeting. Very detailed, and very well received. The next meeting is on June the 22nd for half a day on I&C. The final meeting, which is related to balanced plant materials issues, we're working on an agenda, and a time for that, but it's a bit in the future.

So, we planned to take about an hour, and a half for lunch, but it turns out that the food court slash whoever it's called themselves downstairs will be here today. So, since you folks have identified that you'll eat there, we'll do an hour for lunch, but we'll try to leave a little bit early to avoid any lines downstairs. So, we'll try to finish for this morning at around 11:45, and then convene back around 12:45.

Finally, for information purposes, if members have had a chance to look at the Sharepoint site, there's an enormous amount of material that EPRI has provided for us to look through. A lot of it's related to chemistry issues, and there's a recent EPRI report where they use Gothic to identify consequence scenarios for canister leaks, which is an interesting report to take a look at.

And so, lastly, before we get started, since many of us don't know the EPRI folks that are here, except for maybe a few, what I'd like to do before we get formally started is to go around the room so that the EPRI folks can who they are, and what they do, because we're unfamiliar. So, can we start with Dan, and then just loop around, I guess?

MR. WELLS: Yeah, sure, we can start with me.

CHAIR BALLINGER: Yeah, thanks.

MR. WELLS: Dan Wells, I'm the director for fuels, and chemistry. I've been in the role since November of last year, been in EPRI around 12 years. I have oversight for the chemistry radiation safety decommissioning programs at EPRI, as well as both the operating fuels program, fuels research, and back end used fuel high level waste program. Ron, before we keep going, do you want to do the EPRI stuff that's on the phone line too, or just in the room?

CHAIR BALLINGER: I didn't realize that. Yeah, why not?

MR. WELLS: Okay, we'll go to the phone after we do the room.

CHAIR BALLINGER: Okay, thank you.

MS. AKKURT: Okay, I am Hatice Akkurt, I am a technical staff in EPRI's used fuel, and high level waste management program. In my role I'm leading a number of projects related to that storage, whether it's spent fuel, fuel criticality, or neutron absorber material degradation, or also decay heat. I am also the coordinator for our extended storage collaboration program, which is called ESCAPE.

CHAIR BALLINGER: I should mention that these microphones are very directional, and you almost have to be on top of them. So, be aware that --

MEMBER MARCH-LEUBA: Do as he says, not as he does.

CHAIR BALLINGER: My cord, it won't go any further, but believe me -- okay, next please.

MS. KUCUK: Okay, my name is Aylin Kucuk, I am the program manager of nuclear fuels at EPRI. I have been in this position since December, and I am overseeing our fuel reliability program, as well as NRYR program at EPRI. And I have been with EPRI since 2006.

MR. HALL: My name is Robert Hall, I am the outgoing program manager at used fuel, and high level waste, the same group that Hatice is in. I've been at EPRI about two, and a half years, and did some other things before that, mainly in criticality, and spent fuel pool primarily, and also core design.

MR. SMITH: Fred Smith, I'm the senior

technical executive for the fuel reliability program. I primarily work on high burn up, high enrichment ATF projects, I've been at EPRI for six years, and before that I spent 40 years in the industry with various organizations.

MR. MUFTUOGLU: Hi, my name is Kurshad Muftuoglu, I am a technical executive with the fuel reliability program at EPRI. I've been with EPRI since last September, and I'm working on the fuel issues, advanced fuel technologies, and coordinating the CRAFT framework.

CHAIR BALLINGER: I'm not sure how we do the folks that are sitting here -- that'll work, okay.

MR. HARRINGTON: So, I am Craig Harrington with -- I'm actually not part of the fuels, and chemistry crowd, I'm with EPRI materials reliability program here supporting the discussion on XLPR. Been with EPRI since 2006.

MR. BURKARDT: Hi, I'm Markus Burkardt, I'm at Dominion Engineering, and I support Craig Harrington at EPRI in a lot of different areas, but here I'm supporting regarding work that we're doing for XLPR.

MR. WELLS: And then online, I think we have David Perkins.

MR. PERKINS: Good morning, my name is David Perkins, I am a senior technical executive working in the water chemistry, radiation safety area. My main focus has been working in the radiation safety area for the last couple years, and I have been with EPRI since 2004.

MR. WELLS: Thank you, David. Nathan Glunt?

MR. GLUNT: Hey, I'm Nate Glunt, I am also from EPRI's materials reliability program, working with Craig, and Markus on XLPR.

MR. WELLS: And then Erich Wimmer.

MR. WIMMER: Yeah, this is Erich Wimmer from the company Materials Design, and we are providing for many, many years atomistic modeling services to various EPRI projects, in particular to the ATF, and to the HPU programs.

CHAIR BALLINGER: An my understanding is that Al Santos is stuck in security down below.

MR. WELLS: I just got a message from Al saying where should I go -- where should I meet my escort, that's what he said.

CHAIR BALLINGER: He's a former NRC employee, so he should know the ropes on how to get into this building, but --

MR. WELLS: Do you want me to tell him to meet at the lobby, is somebody already going? Okay, thank you.

CHAIR BALLINGER: Okay, Dan, do you want to say anything else by way of introduction?

MR. WELLS: Well, I have a couple of slides, so just very briefly. So, the programs I introduced myself, and told you what I have oversight for at EPRI, but really we work in the fuels, and chemistry area on the fuel colliding boundary basically, the first primary containment boundary. A lot of the work is on optimization efficiency opportunities, as well as efficient, and safe waste disposal for protection of workers in the public.

So, it kind of just gives you a general overview of the scope of the programs in the fuels, and chemistry department. And then the next slide is just an organization chart, so a lot of the names, and a lot of the staff that introduced themselves are on this, but again, gives you kind of an overview of the different programs, and the different technical areas that are covered by fuels, and chemistry.

So, yeah, just a very brief introduction. So, are there any questions on any of that?

CHAIR BALLINGER: Go ahead then.

MR. WELLS: Okay, so Hattice, if you want to bring up David Perkins' slides, we'll turn it over to David Perkins. Well, I guess we do have the agenda, so the morning will cover the one chemistry topic we have with potassium hydroxide. A lot of the reading materials were papers on the potassium hydroxide effort. We did develop the agenda kind of targeted on the fuels side of the house based on feedback.

But there is, we do cover radiation safety, and in the potassium hydroxide work there is some changes relative to, for examples, potassium-40, which will now be generated in the reactor, we're transitioning that chemistry control regime. So, we'll give you an overview of the general scope of the work we're doing in potassium hydroxide, and then focus on the radiation safety side of the work we're doing for the rest of that presentation.

The rest of the morning is focused on the nuclear research side of the house -- sorry, fuel research side of the house, with a number of different topics. Again, focusing on opportunities where we thought the activities that we're undertaking may be of interest, maybe something you encounter in the future. We would appreciate any feedback you have in the scope, and some of the work we have going on for us to consider as we go forward, and continue these efforts.

MEMBER REMPE: So, Member Ballinger, just briefly, I've been trying to break in, and figure out a good time to say this, but as I was preparing for this interaction, I noticed that there are some aspects of the presentation where I will need to limit my comments because of potential conflict of interest issues. Okay, thank you.

CHAIR BALLINGER: Okay, let's go.

MR. WELLS: All right, so David, we'll turn it over to you, and if you just tell us to advance, we'll advance the slides over here.

MR. PERKINS: Sounds good, thank you Dan. Good morning everyone, we already introduced ourselves, but my name is David Perkins, technical executive with EPRI. A little bit more on the background, I am familiar with this, I was a chemistry manager at a PWR, so I got the chemistry, and the radiation safety background. I do apologize, I'm not there today.

We're on site this week at Sequoia for the demonstration project related to potassium hydroxide, and we're having meetings as we're working our way through the different aspects of that. I'd also like to introduce Keith Fruzzetti, Dr. Fruzzetti is the overall potassium hydroxide project manager. He has put together a very good team of experts in the industry, working through the different issues so we can have a successful demonstration.

Next slide please, Dan. So, the biggest question that we get a lot of times is why potassium hydroxide, why are we making a change from lithium hydroxide? In 2013, Government Accountability Office raised questions about the lithium supply chain. That is driven predominantly because enriched lithium-7, which we use in pressurized water reactors for primary PH control is made in two countries, China, and Russia.

So, if one of those countries had a maintenance outage, or they had something that restricted that supply chain, we've got to have the ability to add to get the PH right for us to start up. But then on the other side is, we're also starting to see a demand challenge. Units are starting to go into flexible operations, we have new pressurized water reactors coming online, that's going to increase the demand.

And even molten salt reactors as we move forward in these next generation reactors. So, you have a supply chain challenge, and we've got an increased demand which was driving the industry, is there an alternate that we can use?

MEMBER MARCH-LEUBA: And Dave, this is Jose March-Leuba, how many kilos of lithium-7 do the power plants in the U.S. use a year, grams, micrograms, kilos, tons?

MR. PERKINS: How many kilos? So to start up one unit, and remember, most of these units are on 18 month fuel cycles, so I'd have to get you more details. But to start up one unit can be anywhere from five to seven kilograms.

MEMBER MARCH-LEUBA: Each unit?

MR. PERKINS: Yes.

MEMBER MARCH-LEUBA: So, it's significant.

MR. PERKINS: Yes, sir.

MEMBER MARCH-LEUBA: And on top of this table we probably have half a kilo of lithium-7 right here, so it's not that the material is not abundant, it's the enrichment from 92 to whatever you need it to be?

MR. PERKINS: Correct. The catch is the enrichment progress. The lithium-6 can create significant tritium issues, the reaction is going to go on in the coolant. So, we enrich the lithium-7 contribution to it to minimize that effect, and that's the challenge. We do not have the capability of doing that enrichment process.

MEMBER MARCH-LEUBA: And out of curiosity, because I mean we get proposals with lithium all the time, is it very expensive to enrich it from 92 percent, which is the natural -- natural lithium has 92 percent natural -- is it expensive to get it to 99.X?

MR. PERKINS: It's very expensive, and also the process that is used to make that enrichment, it's a mercury based process. So, there's not just an overall expense, but even the process itself creates some unique challenges.

MEMBER MARCH-LEUBA: Okay, thank you.

MR. PERKINS: We talked about this, when we were looking at it, it was brought up, cost already. So, there is the potential when we shift from this enriched lithium-7 to potassium hydroxide, these units can save a significant amount of money because of the cost of the lithium-7. And then when we step back even farther though, and we say the other benefits, there is the definite potential from a fuel perspective with lower corrosion.

And also when we're looking at mitigation strategies related to the crud induced power shifts that the industry has observed in the past. If you step back even farther, and we say okay, where has KOH been, it hasn't been applied. KOH has actually been applied in the VDRs since the beginning. It has been successfully applied for decades, and we've been able to leverage that operating experience, as well as part of this project to get us moving forward into it.

And they've helped us significantly on that, next slide, Dan. So, big picture on the program, there's four key areas that we're looking at on this, materials, fuels, chemistry, and radiation safety. We're not going to get into the materials, and fuel testing, but you can see we're looking at initiation, crack growth testing, we're doing some loop testing for the fuel side of the house.

Those tests are ongoing, they're scheduled to finish up later this year. Today we're going to talk a little bit about the chemistry, and give you, really the update on the radiation path on there. All of this is being managed through the station, Sequoia station, and actions, or activities are being tracked through their corrective action program, and the team is working through the 10 CFR 50.59 evaluation process.

Again, we've got to finish the testing on the materials, and fuels side of the house, and we can start rolling forward into 2024. Next slide please.

CHAIR BALLINGER: So, this is Ron Ballinger, I'll display my ignorance, but do you folks think that you'll be able to do this through 50.59, or will it require a license amendment?

MR. PERKINS: Well, that's -- the station is working through that process now, and according to the discussions yesterday, they said they still have to work through the 50.59 process, and then they'll see if they have to go through the license amendment as we get into it some more, so that's to be determined in the future.

CHAIR BALLINGER: Thank you.

MR. PERKINS: Next slide please. All right, so when you look at this from the chemistry radiation safety aspects, there's a couple key points here, before we get really into details on this. The chemistry space itself, because it's been used in the VDR fleet, we've got a lot of operating experience of how to operate these systems, how to manage the resin beds, whether it's a cation bed for day to day operations, we do have a lot of experience there.

We also have some experience on what the observations were related to impurities from the chemistry side of the house too, from these plants. And regularly we can model this with tools that we have within the EPRI space, looking at the high temperature calculations in there. Radiation safety has been, I won't say more challenging.

But it's a little bit more in that we've got to look at we are introducing potassium hydroxide into the coolant. That results in the potential activation, and production of several radioisotopes, and with that comes into considerations about instrumentation, radiation fuel changes, dose to the worker, effluent release pathways, waste streams, waste processing.

So, there are several aspects that we're working our way through with the plant, and we'll talk a little bit about them as we go forward. Again, the main assessments have been completed from the EPRI side of the house, and now we're working with the plant to get through, and help them with their day to day management programs. Next slide please. These are the six key areas just for you.

We talked a little bit about them, but if you see in there where the numbers are in parenthesis, those are EPRI reports, where the assessments have been completed, and documented on there. Big experience on this radioisotopic one, which is where we're going to talk to you, there's no significant issues, we just have to work our way through the process on different radioisotopes, and the impact on the fields on there.

Dan mentioned Potassium-40, there are several other radionuclides, potassium-40 to argon, chlorine-36 that come into play in there. So, we do have to step back on that, and look at this from how the site manages this. The vendor reviews are completed, phase two of Westinghouse review is completed, and documented in the report 0959. And again, the plant demonstration.

We're looking at trying to get this plant demonstration started next year with Sequoia. It will be three cycles, each part of the cycle from the chemistry radiation safety aspects we have baseline monitoring that we'll evaluate for changes as it goes forward. And again, as of today, right now, we do not see on the chemistry radiation safety side of the house, any significant challenges moving forward.

CHAIR BALLINGER: This is Ron Ballinger again, are these EPRI reports public?

MR. PERKINS: Dan, I would have to look, I don't know --

MR. WELLS: They're not public. We did provide -- the references that are public are the ones that we provided in the reading materials.

CHAIR BALLINGER: But for purposes of any submittal to the NRC, these reports would be available?

MR. WELLS: So, we have relationships with the funders of EPRI, so if they need materials to submit to the regulator, they can submit them. But again, I think the indications are all that we do expect we'll be able to process this, and the site will be able to process it through a 50.59 evaluation, and then that would negate that need.

CHAIR BALLINGER: Okay, thanks.

MR. PERKINS: Okay, next slide. So, chemistry control, one of the things that we've talked about, and we had pretty significant discussions on this. Right now, we normally have just lithium hydroxide in the system. Lithium is working in that from an alkaline chemistry perspective on the PH program. When you look at potassium, and lithium from an equivalent standpoint, it's about a one to one.

So, from a management standpoint, as far as the beds go, and the operations of the beds, it's a manageable condition, and operations on there, obviously that they will have to take some procedure changes, and address them on the procedure, but chemically it's -- we don't see a major challenge there. Now, one of the things on this though, is we still will be producing lithium.

Boron-10 neutron reaction will still produce lithium, this is why they're going to have to manage both potassium, and lithium controls. So, they're working on this not only from a technical aspects, but from human performance aspects in how they're going to manage both of these potassium, and lithium to maintain PH control of the plant. It does require some additional monitoring from the station.

They're looking at the capabilities, their existing instrumentation, they'll actually be monitoring now, potassium, and sodium to really look, and evaluate, and track, and trend the system performance. Resin management, we don't see an issue with the resin management. Resin management we've talked a little bit about on there. The biggest question on there was -- that we've had so far, is dealing with the caddine resin.

And the impacts with now, potassium, and lithium is they're going to create an issue where we need to make potassium additions, compared to when we put the caddine bed in service. Do not think that's going to be the case, because as the lithium builds up, we'll reach a point in the cycle where lithium will actually be aiding us, and helping us more with this PH control program.

Connectivities from an analysis standpoint, again, looking at a human performance aspect on this part, technicians in the field, the technicians when they take a sample, and they're doing some of these samples, they are going to see differences, especially in the connectivity. In the connectivity space they use this as an indicator for potential impurity, or changes.

There's going to be some changes on that, we're working through that, identifying that with the station, the different analysis, and the potential impacts on there. So, key thing on here is the control, using both potassium, and lithium, if you look at it from about a one to one equivalent, it's equivalent to lithium only.

So, we think from that side of the technical aspects side of the house, we're done. Now we're addressing the human performance side as well.

So, next slide please. We've got six radionuclides that we're concerned with. Argon-39, argon-42, chlorine-36, potassium-40, phosphorus-33, sulfur-35. Now, if you look at the total activity in the coolant, it's less than 0.1 percent of the total activity.

So, in most cases this is not even going to be an issue as far as the total activity in the plant goes, it's such a small fraction. What we're looking at, and working with the station RP staff is the potassium 40 issue. What's the impact on whole body counting, other monitors, whether it's a gamma spec system, we're looking at the effluence, waste disposal, and dose impacts.

MEMBER MARCH-LEUBA: This is Jose again.

MR. PERKINS: Yes.

MEMBER MARCH-LEUBA: Most of the activity is nitrogen-16, that has a very short lifetime, is there a long term issue with activity?

MR. PERKINS: Yes, your correct, nitrogen-16 in the PWR is one of the largest contributors. What has calculated from the model is potassium-42 has the potential to be a significant contributor. Potassium-42 is a very short lived radionuclide, so that would not be more of a long term effect. So, we're having to look at this both in that short term effect on the activities potentially from the potassium-42.

Potassium-40, you brought that up, is there an impact that we see differently from a whole body monitoring, and that's what we're having to work our way through. We don't think there's a long term impact, especially in the waste streams, and waste effluence. Most of the waste streams are already analyzing, if you get your results back, it has potassium-40 identified in it. So, I don't think it's a long term issue there.

MEMBER MARCH-LEUBA: Yeah, the purpose of my question was to figure out if you guys are thinking about it, I don't want to know the answer, it's -- you have to think one millisecond, and two years after.

MR. PERKINS: Yeah, correct, yes.

MEMBER SUNSERI: This is Matt Sunseri, I've got another question on this slide. So, regarding the coolant activity, and these residual amounts, or these minimal amounts, have you looked at the impact of the -- I'll call it the post crud burst cleanup activity levels, I mean would there be any -- .1 of the total pre-crud burst would be one thing, but .1 after might be a bigger number.

MR. PERKINS: Yeah, so when you look at

this, Matt, and good to hear from you, the issue on the crud burst, you're dominated by activated corrosion products. That is cobalt-58, cobalt-60s, when you look at that release, this is even a smaller fraction. And then when we clean up that activity, we're still going to be a very low fraction of that activity.

MEMBER SUNSERI: All right, thank you.

MR. PERKINS: Next slide please. So, here's the key things, Jose, I think this will start answering some of your questions on here. What the station did is they asked a series of questions related to day to day programs for the RP side of the house. And we worked with the industry subject matter expert, and we developed a white paper on this. These are the key areas that we've looked at, as far as in this white paper.

The item in green up at the top right, radionuclides of concern, those were identified in the EPRI report. Items that have kind of a gradient green, those are the ones they're actively working on right now, today in the station it's being tracked in there. It's not just long term rad waste, but instrument responses, personnel monitors, we're looking at this how it's going to impact them today, and potentially in the future all the way up through the rad waste, and effluence perspective.

I'll kind of go through a couple of them, just to give you kind of an idea what the scope is here. Next slide please. Radionuclides, we talked about this, the radionuclides, you can kind of see in the image on the left, there is a significant number of radionuclides that are potentially produced at some level. Most of them are very short lived, we have the six that are greater than 24 hours, but most of these are very short lived.

But they will have an impact on day to day operations, the gamma spec libraries, the whole body counters, they're going to have an impact on potentially what we see on baseline monitoring for these systems. Now that you have new radionuclides, we're looking at the reactor coolant systems, the filters, core filtration, cleanup resins. Evaluating trying to look at if you've got a liquid discharge, or a gaseous discharge, what's going to potentially change on that one.

And we started this with a baseline monitoring program looking at getting these samples beginning of the cycle, middle of the cycle, end of the cycle, and now we can compare them as we move forward. Next slide. I talked about the baseline, so here's kind of a bigger picture of that. We focused on five areas, so I think it was Jose that asked a question on there.

We started with sampling, source term, rad waste, worker dose, and surveys. Under each one of those areas, we've got a look at today on the baseline, and then as we move forward through the demonstration, looking at the longer term considerations, as well as monitoring for these changes. Whether it's a chemistry sampling program, whether it's a radiation fuel changes through cadmium zinc telluride technology.

Whether it's waste changes, what's going on with the Part 61 analysis, or what potentially could be changing in the Part 61 analysis. Ultimately, we've got to get back to is this having an impact on the worker? Are we seeing a difference in the dose? Based on the BWR fleet, they do not see any differences specifically from these radionuclides, including potassium-40, but we've got to document, and understand those effects.

MEMBER HALNON: And this is Greg Halnon, quick question. Do these five areas encompass the technology, or the systems required for injection, and storage of the chemical on site?

MR. PERKINS: You're talking about like the chemical add system in the system, or are you --

MEMBER HALNON: Yeah, I mean there's modifications required, is there any additional hazards storing this on site?

MR. PERKINS: No. We will use the same chem add system that they use to add lithium hydroxide. The storage, and handling requirements will be consistent with lithium hydroxide as well.

MEMBER HALNON: Thank you.

MR. PERKINS: Next slide please. So, instrument -- went too far, one too far, back. Okay, real quick, on the instrument -- I know we're starting to run out of time here, don't want to get behind on the first presentation, but instruments we do have to look at, and we have now with these radionuclides, we have different energy responses that these systems may be exposed to, or this equipment may be exposed to.

Now, the good thing like with the portamonitor, portamonitors actually have a chlorine-36 efficiency rating to them. So, from that perspective, we're able to actually bring some of these harder to detect, more infrequent radionuclides, and we're able to say yes, these monitors do work.

So, the plant station staff is going through each of their instruments, whether it's a tool equipment monitor, a small article monitor, or a portamonitor.

And they're reviewing that based on the energy spectrum differences that we may be seeing in there.

CHAIR BALLINGER: This is Ron Ballinger again, can I get a little bit of perspective? What kind of dose are we actually talking here? If a worker brings a banana to work, the content of potassium-40 in that banana gives them, I don't know, .01 millirem just by eating the banana. What are we talking about here?

MR. PERKINS: Right. So, potassium-40 is not such at issue. When we're talking about dose, and we're talking about dose rates, the ones that we're more worried about to the worker from a day to day perspective is impacts like potassium-42, because of the amount of activity in some of these shorter lived, which is what we're monitoring for. Potassium-40 dose, you're absolutely correct on that.

CHAIR BALLINGER: Okay, thanks.

MR. PERKINS: Yes. Next slide please. So, just real quick, before we go into it, we're trying to -- I shouldn't say everything, we're trying to capture the things from basic gamma spectroscopy libraries up through bio assay programs. With the introduction of some of these new potassium hydroxide, and those six longer lived, does that, or how would the bio assay programs have to change?

They're currently working with the vendor now to make sure the bioassay program can detect some of these longer lived radionuclides. And on potassium-40, you're right, they already can detect it, and that's a normal part of the body, we have it in our bodies every day. So, they're looking at like chlorine-36, and some of the other isotopes in the bioassay program.

Next slide please. So, real quick, kind wrapping it up, right now we're in the middle, they're working on the 50.59 process as we talked about. They are expecting that to complete by the end of the year, and as long as we can get through the 50.59 process, that would roll into the engineering change package to support a fall 2024 start, or injection. Going kind of refresher on this, we are holding in person meetings this week on site.

We do have regular conference calls with them in the chemistry, and radiation protection area. Those areas deal with anything from human performance, plant operations, up through equipment monitoring, and sensitivities. And with that, I think that's the last slide. Is there any other questions? Yes.

MEMBER MARCH-LEUBA: Yes, not a question, but a comment. You know, when you submit things to this building, to NRC, the staff has to follow the regulations to the last semicolon.

MR. PERKINS: Correct.

MEMBER MARCH-LEUBA: ACRS, we're allowed to think outside the box. I'm thinking right here, the banana comment, are we overdoing this? I mean, are you over killing it? In 1960, we'd have just gone to a reactor, and dumped some sodium hydroxide, or sorry, potassium, and see what happens. I mean, you do some scope, and calculation, you run, and see that it's not a real serious problem.

But we are in an area which we're supposed to be a modern risk informed regulator, you risk inform this, and say let's just put it on, and see what happens.

MR. PERKINS: So, how can I say this? I agree, we have minimal risk that we see on these from the chemical radiation safety side of the house. I do think we do need to finish the material, and the fuel testing though, before we can say, and just inject something. I think we really have to finish that testing first. The chemistry radiation safety aspects of this are manageable, they're well documented from the VDR side.

We're just addressing those as they come forward, but again, I focus the material, and fuel side of the house, we need to finish that testing.

MR. WELLS: I would, if I can, Ron, just add to that. When we started this effort back a number of years now, we did evaluate what is the kind of baseline testing that's required to move, versus answering all of the materials, and fuels questions. So, I think from the beginning, our testing matrix, which a lot of that is outlined in the reading materials, we're not testing everything.

We tried to identify what are the most limiting materials, what are the materials, whether it's fuels, or primary system materials that are different from the VDRs that we can't point to that experience, and bring that into this fleet. The NSSS reviews did identify a couple of materials that we need to add to that, but we have on the testing side said what is the required?

We don't have to do everything, the rest of it can come over time once we transition.

MEMBER MARCH-LEUBA: Yeah, but I'm pretty sure for the last year we have not been importing any lithium from Russia, and the political situation with China is very questionable. So, if I was a manager, and was running my plant, I would like you to speed up on getting me potassium, or getting me 100 kilos of -- a strategic stockpile.

MR. WELLS: And there is strategic stockpile within the DOE, it's not very clean --

MEMBER MARCH-LEUBA: Very small.

MR. WELLS: It's not very clean material, it --

CHAIR BALLINGER: It doesn't have to be clean in there.

MR. WELLS: In 2015 we went through this, so the shortage we saw in 2015 was associated with production going down. Basically all of the Chinese production went away, and the only remaining production was coming from Russia, which is not supplied predominantly to the U.S., so we did see that, and had lots of discussions with where is the stockpile, who has it?

Our understanding is that most utilities have now a cycle, or two in reserve at the site after what happened in 2015.

MEMBER MARCH-LEUBA: It used to be Y-12 in Oak Ridge was the production facility for lithium, and I live in Oak Ridge, we used to spend oodles of money cleaning all the mercury, and still there. So, they don't want to run it anymore. I think they're forbidden by law from doing it.

MR. BLEY: This is Dennis Bley. I kind of agree with Jose on whether we're over killing this, but you seem to be relying a lot on the VDR data. Why do you have much confidence in the data you get out of the Russian system? I've run into problems with how things are reported over there over the years, and it's a political process as much as a scientific one.

MR. PERKINS: So, and Dan, you can jump in on this as well. But here's the thing, we have members that are in Czech Republic at Temelin, we have members that are in Hungary at Paks. We have very good working relationships with them, and they have provided this VDR information. We have not relied on the Russian VDR fleet itself.

MR. BLEY: That's much better, thank you.

MR. PERKINS: Yes.

CHAIR BALLINGER: This is Ron Ballinger again, I need to make a comment that was made yesterday. While we on this committee think outside the box, what you're hearing is our own opinions. We only speak through letters, if it comes to that, that's the only formal thing that we have, thanks.

MEMBER SUNSERI: Dave, this is Matt, one more question. You went through this really fast, and you may have said it, I might have missed it, but are there any what I'll call adverse side effects from creating this product, like the enriched lithium with mercury?

MR. PERKINS: No, sir, potassium-40 is readily available, and it is very easy to manufacture, so we don't see any side effects from that.

MEMBER SUNSERI: And that the purity level that you need for?

MR. PERKINS: We're working through it right now. The manufacturers today, based off the feedback we've gotten, again from Czech Republic, and Hungary, is that they've been able to meet all of the more restrictive purity levels.

MEMBER SUNSERI: Okay, so we don't have the down side of the enriching --

MR. PERKINS: No.

MEMBER SUNSERI: All right, thanks.

MR. PERKINS: We're going to monitor for it as we go through it, and also they all have their materials specifications, but we don't anticipate a problem there.

MEMBER SUNSERI: Perfect, thank you.

CHAIR BALLINGER: This is Ron for one last question. It sort of begs the question why are we using lithium to start with? Is there somebody here that's old enough to remember?

MR. WELLS: We have yet to find the answer to that question. The only easy answer is it's simpler chemistry control, because you're only managing one alkaline.

MEMBER MARCH-LEUBA: It was a cheap byproduct, they didn't know what to do with it. I'm serious, they were depending on the market.

CHAIR BALLINGER: Okay, thanks.

MR. WELLS: Thank you, David.

MR. PERKINS: Thank you.

MR. WELLS: Okay, so we'll transition topics. The next set of topics are all in our fuel reliability, nuclear fuels area. So, we'll start off with a review of the program, and then step through some specific topics that may be of interest.

CHAIR BALLINGER: So far that -- you've got to get it as close as you can.

MS. KUCUK: I'll try.

MEMBER MARCH-LEUBA: Basically you have to hear the feedback from the room. If you can hear yourself from the ceiling, you're doing good.

MS. KUCUK: Okay, is that good? Okay, awesome, great. So, I think I introduced myself earlier, but again, my name is Aylin Kucuk, I am the program manager of the nuclear fuels at EPRI. I will mainly talk about the fuel reliability program. We do have another program called NFIR, we are going to mention it.

But since a majority of our work is related to ATF high burn up, and high enrichment, within fuel reliability program I will mainly talk about that program. So, in this fuel reliability program we have three main objectives. The first one is to support current operating fleet to minimize, and avoid fuel failures, and fuel performance issues.

So, we may need to develop an updated fuel reliability guidelines, tools, and handbooks. We also inform industry for the operating experiences with the -- we're supporting them for the change management. We collect old operating experiences, and response to those experiences, and we provide the necessary guidance, how to react, and give them some specific guidance through those documents.

So, we also perform research to capture constant operational efficiencies. There are so many varieties of options available that utilities are looking at. So, some of them are listed in this slide, I'm not going to go over them. And utilities are not looking at all of them, but these are kind of options that appropriate as needed, they are considering those options.

And we are working on developing some technical bases, and some research to help them on their decision making. So, the third area is the developing technical bases for regulatory, and safety issues. These are -- three things are on the table right now, the ATF, high burn up, and LEU+, and FFRD. And you will hear what we are doing on these areas later today.

But previously LOCA, and RAA were the main topics that we worked, and performed a variety of different research to support the technical basis for those issues in the past. So, a little bit about our research. So, I'm going to go over very high level just to give you guys an idea on what this program is really doing. The first one is PWR crud, and corrosion area.

The main issues are the CIPS, and CILC.

CIPS is crude induced power shift, and CILC means crud induced localized corrosion. Those are kind of potential risks during operation in PWRs, and especially for high duty PWRs. We have a tool, a CIPS, and CILC risk assessment tool called BOA. We mainly perform research on developing the necessary models within this tool to have a better prediction of these conditions during core operation.

So, you heard about potassium hydroxide issue. As the fuel reliability program, our responsibility is to perform the necessary testing to make sure that potassium hydroxide can be successfully implemented with no fuel performance issues, and I think they already mentioned about the type of testing that we are doing to support that program. And we also have a PWR fuel crud, and corrosion guidelines.

Mainly including the operating experiences, and previous research related to CIPS, and CILC issues, and providing necessary guidance to utilities. Similarly, for BWR crud, and corrosion area, we do have a similar tool, CORAL, to perform the crud risk assessment, and as needed, for BWR, crud issues is not as operational as PWRs. Like CIPS could be happening, and causing a lot of issues during operation.

But for BWRs, it's more of CILC type issues. So, the CORAL tool is used for if there are any known water chemistry transients, or some issues occurring. But we do have a significant amount of operating experience included in our guidelines, and very robust guidance, how to do a risk assessment, a cycle risk assessment to utilities.

Advanced fuel technologies area, this is an area started with the ATF program, started in the industry. We mainly focus on the fuel reliability, and performance benefits of ATF, high burn up, and LEU+. And then as we move forward, as these new technologies become more robust, and get into an implementation stage, we are going to update our guidelines, tools, and handbooks for a safe implementation, and proper implementation of these technologies.

But in the meantime, we are also performing research to enable safety, and economic benefits of these technologies, and you're going to hear some of that work later today. So, bringing the failures, that is the main failure mechanism for fuel today, and it has been like that for a long time. And what we are doing is really generating guidance, and training on how to control for material exclusion.

But we also look at some fuel technologies that -- you know, how resistant fuel cladding could be. Like coated cladding that fuel vendors are developing could be an additional resistance to debris threading that some utilities are looking at as a more robust type cladding for debris in these fuel failures.

MEMBER MARCH-LEUBA: Chromium coated, or something more sophisticated?

MS. KUCUK: So, it could be an old -- I mean, the main known is the chrome coating. There are some preparatory coatings that BWR vendors are developing. There are -- we have not done the well defined testing yet. So, we'll be looking at the debris resistance of all type of coatings.

MEMBER MARCH-LEUBA: Have they tried heat treatment, hardening?

MS. KUCUK: Yes, the goal is to really do some testing in an ultra cooled with high temperature, and high pressure condition, and doing the bare within that environment to simulate the real environment. To be able to compare bare cladding to the coated cladding, see if there's any bigger differences between that.

CHAIR BALLINGER: This is Ron Ballinger

again, there's a lot of acronyms up there. One members might not be familiar with is OLNC, which is online noble chemistry.

MS. KUCUK: Thank you for pointing it out, yes. Is there any other acronym that --

MEMBER MARCH-LEUBA: Yes, all of them, but you don't need to do that.

MS. KUCUK: All right.

MEMBER KIRCHNER: Aylin, this is Walt Kirchner, I'm offline in Santa Fe, New Mexico. What is your dominant interest about your risk assessment tool? What do you see as the major driver in crud, and corrosion problems for PWRs, and then BWRs? I mean what does it correlate mainly with, is it water chemistry, or is it burn up, or?

MS. KUCUK: So, I think both of these tools really look at the thermohydraulic conditions in the fuel bundle, and also the crud, for PWRs, coming from the corrosion of steam generators, generator tubes that iron, and nickel comes in. And then this tool really estimates the amount of the corrosion product release, and then depending on the boiling, and thermohydraulic conditions in the bundle, estimates the position of those corrosion products on the fuel.

And the associated consequences of the areas that these crud layers could present from a conductivity --

MEMBER KIRCHNER: I see. So, even with the advanced steam generator tube alloys, that's the major source of foreign material then that's impacting the actual fuel clad?

MS. KUCUK: The corrosion products like the new steam generated tubes have much less corrosion, so it impacts the source of the crud. So, that's a big benefit to those PWRs, that they have less crud coming in. But when new steam generators are replaced until the protective layer is established, there is a large amount of corrosion products coming in. So, it all depends on what stage these new steam generators are, and the material.

MEMBER KIRCHNER: So, is there a way to pre-age the steam generator tubes with the installation, so that you don't have this source threatening the actual --

MS. KUCUK: You want to take it, Dan?

MR. WELLS: So, we have done some work looking at -- there's a number of different technologies that have been applied to reduce general corrosion once the material is put into service. No one has tried it on steam generator tubes yet, we have done some work over the last couple of years updating the way the spec has been written so that you could utilize it.

Because the way the spec was written was a shiny metallic surface, and if you're pre-oxidizing, you don't have a shiny metallic surface anymore. So, we've done some work in that area, but it hasn't been applied yet.

MEMBER KIRCHNER: Okay, thank you.

CHAIR BALLINGER: I might comment that you probably ought to be talking with the Navy people if you haven't already.

MEMBER MARCH-LEUBA: Talk to the food industry people, I mean --

MR. WELLS: There's a lot of work that's been done in that area.

MEMBER MARCH-LEUBA: Whenever you put a new tank of stainless steel on a juice factory, the first thing you do is oxidate it.

MR. WELLS: And there's a lot of work that's been done on, especially in new plant space, can you utilize your hot functional testing period in order to pre-oxidize, and clean up those materials before you bring the fuel online. There's been a lot of work in that area, but hot functional testing, the data, it's hard to interpret over the long term, what the benefits were.

But new material being inserted, a lot of the steam generators that were replaced in the U.S., the bowls were electropolished to help with that, but actual pre-oxidation has been limited with applications.

MEMBER MARCH-LEUBA: All you have to do is fill up the metal with the proper chemical, and let it there. That's what they do, they just fill up the pipes until the stuff moves past it. We've used an amount of time simply because this is an interesting topic, but on the crud, the issue is a chemical, atom by atom deposition, or do we have concerns of flakes coming out that would become a loose part that can impact?

MS. KUCUK: So, mostly for the fuel failure risk wise, it's more on the how the crud is deposited, and then how is a heat impact to the cladding --

MEMBER MARCH-LEUBA: Chemically, atom by atom?

MS. KUCUK: Correct. There were some concerns on the BWR side whether the spalation of crud, as well as the zirconium oxide layer, because the -- along with the corrosion products, the cobalt-60 can also deposit, and very high dose type of crud that you can find especially on BWRs, if it just falls off, we were concerned about whether that would distribute those in the system, or not.

But it has not been really demonstrated, and looked at in detail. But that was one of the considerations. But other than that, I think for PWRs, that is not the concern.

CHAIR BALLINGER: So, let's be -- I guess clear it up, I think. The major fuel failure path, mechanism, at least in BWR is debris threading, right?

MS. KUCUK: Correct.

CHAIR BALLINGER: In PWRs, I'm not sure what it is, because that's usually much less of a problem.

MS. KUCUK: So, debris failures in PWR are much less than BWRs, since it's a closed system. But as of today, when we look at statistics, it is still the only failure mechanism that is happening.

MEMBER MARCH-LEUBA: The primary research, because we've become so good at the other mechanisms, and debris, you can only filter it.

MS. KUCUK: Correct.

MR. WELLS: I mean, it's hard to say it's causal, but if you look at the data for when our initial guidance on managing crud, and corrosion, and the risk assessment tools were developed, and the application of fuel reliability program in general, you do see many of these mechanisms have dropped to basically we don't see them anymore, because we're good at predicting conditions under which you will find them.

And so, we've largely mitigated it. We still see debris failures in one, or two other types of mechanisms.

MEMBER MARCH-LEUBA: So, going back to personal opinions of the banana type, this is 00 we're reaching a little bit like on the wine scoring. If you buy a wine on the 85, and you buy a 90, maybe an increase in price is three dollars, and it's a big change in quality of wine. But if you go from 98 to 99 the difference is 50000 dollars, and you can't see the difference.

So, there's a point in which we reach the 98 score in wines, and maybe we start to over kill. Personal opinion, right?

CHAIR BALLINGER: This is certainly fun.

MS. KUCUK: All right, so the other areas that we're working on, the hydrogen, hydrogen impacts on zirconium. So, we have a kind of funny acronym since the name of this area is a little long. But we mainly look at the scientific, or mechanistic understanding of how hydrogen is picked up by the cladding. That is a kind of -- a long studied area that we are trying to improve on, and also trying to model in how hydrogen is picked up.

As well as how hydrogen impacts operational issues because of hydrides, and so on. But we try to get as much hydrogen measurements as we can. It is a costly activity that involves a PIE, but we do those type of measurements on really unique cases just to expand our database, and everything. For guidance methods, and tools, we have a bunch of other guidelines, documents, handbooks, and databases.

The fuel surveillance, and inspection guidelines just give guidance to utilities on how to inspect fuel, when to inspect fuel, the frequencies. It really provides some sort of process systematically utilities can implement. For non-destructive evaluation technologies, we develop non-destructive evaluation technologies for failed fuel identification, and anomaly identification.

Either eddy current, or some visual

inspection tools to ease identification of anomalies on spacers, and row wall, and things like that. But besides that, we also are developing eddy current technologies for better measurement of cladding oxide, and crud thicknesses. And if we can be successful, we are also targeting to be able to measure hydrogen cladding in the pool side.

So, it has not been demonstrated yet, but that is still in works. Control components, and structural components, there are some work going on in FRP as well, mostly for life time prediction improvements, and for PWR control rod modeling area. So, additive manufacturing of fuel components is an area that is becoming more, and more interested.

That especially fuel suppliers are looking at some really advanced debris filter designs to screen, or filter out debris, or much smaller, very light debris as possible. This will be an area that we'll be probably focusing on in the next several years.

CHAIR BALLINGER: Since we're compiling banana comments, SHIZAM, by the way, there used to be a TV show when TV was still black, and white, which is probably before all of you were born, where there was an actor, singer who used to use that word as part of his act. And I now forget the person's name, so anyway, another banana comment.

MR. WELLS: I guess while we're taking banana comments, a lot of this is maintenance of stuff we've done. Just Jose, to your comment, we are, if you think back to Aylin's first slides, we are focusing on some of the newer things that are pushing into new areas, and not just kind of these are -- a lot of the previous ones at least, are maintenance of those products to make sure when things change, they're still effective, things like that.

MEMBER MARCH-LEUBA: Your day time job should be keeping the operating plants running, it shouldn't be unsafely. But have to keep a little bit of one hour a week to think this moon shot, how are we going to get the silicon carbide cladding, things like that. But remember what your day time job is.

MR. WELLS: Absolutely, thank you.

MS. KUCUK: That's exactly what those slides are mainly for. However, now I'm going to go into kind of new stuff that are kind of on the table. But before I go into what EPRI is doing, I just want to give you guys a kind of overview about what is going on in the industry related to those new technologies. For ATF concepts in the U.S., there are some near term, and longer term concepts.

For near term concepts, mostly coated cladding, and for longer term concepts, non-zirc type cladding like silicon carbide, and iron clad, iron chrome aluminum type cladding. So, this slide really highlights, and gives a lot of the details in terms of who is doing what, and what type of technologies. For PWR, mostly focusing on chrome coating on their base cladding material, depending on the supplier.

For BWRs, there are only two different kind of preparatory coating, and Framatome, and GE have been working on, and GE is also working iron chrome aluminum type cladding. This is the kind of technology between near term, and long term. It's not really long term, but it's sooner than the long term type technology. In the long term the only material is silicon carbide that only PWRs are looking at.

For fuel, the U2 is already approved by NRC that Framatome, and Westinghouse has developed. But Westinghouse is also looking at uranium nitrate type fuel, which has some leads already in reactor. And fore burn out, all suppliers are looking at burn up up to 75 with LEU+. So, this is on the development side. And then going into the LUAs, or lead fuel, or test rods, this is a slide that NEI prepared.

And it's up to date as of right now, the variety of different options are under irradiation since 2018. And some of these test rods were already completed at least one cycle, and some of them already shipped to a head cell for PIE, and then the further irradiation for a bunch of them are ongoing. Those variations are coated cladding, and an iron clad, iron chrome aluminum is being irradiated in two different reactors.

And different variety of fuel add pellets, as well as uranium nitrate type high density fuel. And with two different high burn up LUAs are ongoing. And with LEU+, I think that's the -- Vogtle is the only one, LEU+, and high burn up, and coated cladding all together, that will likely start in early 2024. So, this is a kind of very easy chart with a lot of colors, and a lot of shapes.

But I try to summarize what's going on in the industry, and then what is the kind of interest that utilities are looking at with these new technologies. So, ATF started with only coated cladding as a near term technology. The main focus is the safety benefits, and it has shown that there are some operational safety benefits, especially with higher burn up, and enrichment.

But also severe accident benefits for each of these technologies, but the benefits could vary from technology to technology, or whatever condition is being evaluated. There are some desired fuel performance benefits that needs to be demonstrated, but that's the kind of incentive for utilities. Like I talked about debris resistance.

If these coatings can provide some additional debris resistance to prevent debris fuel failures, that's a big economical impact, or savings to utilities. If it can get low corrosion, and hydrogen pick up, it could really provide -- excuse me? Okay, all right. If I continue, so, I think the other benefit we are looking at low corrosion, and hydrogen pick up.

It could really provide some flexibility to operation, and a lot of simplifications, and also if it can be demonstrated as robust to whatever chemistry transience that may happen in reactor, that could also provide an additional margin to plant operation. And then the other big benefit, and that's the one that we're going after the most, is fuel cycle economics.

That's where the high burn up, and high enrichment piece really comes in. So, PWRs are trying to transition 24 month cycles, right? So, especially for longer -- I mean high duty PWRs, if the burn up is extended, and if the higher than five percent enrichment is allowed, then those high duty PWRs could go to 24 month cycle. There is a kind of significant economical benefit in that case.

That's the main purpose that high burn up in LEU+, and FFRD is the main technical challenge, that's where we are kind of focusing on developing some technical bases, and looking at different approaches to address that FFRD issue. And Fred Smith, and the XLPR team are going to mainly talk about what is our approach to address FFRD to open up the path for high burn up licensing.

CHAIR BALLINGER: You know that there's a rule change that's coming down the pipe on high burn up, and FFRD is explicitly required to be accounted for.

MS. KUCUK: Right, yeah.

MEMBER MARCH-LEUBA: We talked about acronyms before and for the transcript, when you say that something is really, really important, tell us what fuel fragmentation and dispersal is?

MS. KUCUK: Yes, FFRD means fuel fragmentation relocation, and dispersal.

MEMBER MARCH-LEUBA: So, only when you say in the transcript this is extremely important, as a member of the public I'm reading it --

MS. KUCUK: Okay.

MEMBER MARCH-LEUBA: They don't know what it is.

MS. KUCUK: Good reminder. Any other questions?

MEMBER MARCH-LEUBA: Yes, we are officially way over time, so I'll shut up.

MS. KUCUK: Okay.

CHAIR BALLINGER: Not officially way over time yet.

MS. KUCUK: Okay, I'll --

CHAIR BALLINGER: But we have a break scheduled at 10:25, so we're working at it.

MS. KUCUK: Okay, maybe I can speed up here. So, the other pieces, the power up rates that Inflation Reduction Act really provided some huge incentives to current operating fleet to do power up rates. And we're looking at time, and temperature criteria changes to support the utilities to look at increased power up rates. So, with that, I think I have a few slides just to go over in terms of what EPRI is going.

Maybe I'll just go over very quickly with you guys, because some of them the team is going to talk in depth. So, the high burn up area is one that alternative licensing strategy, is that what's we're pursuing to address FFRD, that Fred Smith is going to talk in detail. And our main goal is really to resolve this FFRD issue, and open the path for industry to expand burn up up to 75.

Besides the alternative licensing strategy, there is also some work going on related to FFRD consequences assessment for large break LOCA, and fuel suppliers are developing the methodologies, but EPRI's role is to develop some technical bases for some of these phenomenon to support them, and also provide some clarity for those issues.

MEMBER MARCH-LEUBA: Any plans for experimental data to back up your analysis?

MS. KUCUK: For alternative licensing strategy --

MEMBER MARCH-LEUBA: For 75 gigawatt FFRD.

MS. KUCUK: So, the experimental data wise, we're doing the -- working with fuel suppliers to do some hot salt PIE in terms of characterizing the fuel fairly high burn up for fusion gas release measurements, and fuel conditions, and so on to support the fuel performance tools implement. We are also doing some testing, mostly on cladding burst, and dispersal phenomenon related work.

MEMBER MARCH-LEUBA: So, you're using --

MS. KUCUK: There are, yes.

MEMBER MARCH-LEUBA: Okay, because the trend is to go 100 percent computational basis, and you need to validate the basis.

MS. KUCUK: There are some also very large set of LOCA test plan at the treat facility at INL, and to really -- I mean that is going to be the kind of main set of data to support the fuel suppliers, methodologies, and qualification as well.

MEMBER MARCH-LEUBA: Thank you.

MS. KUCUK: Okay. And we have this CRAFT framework that Kurshad is going to talk more on that. We are trying to coordinate those efforts within the industry through that framework. CRAFT means collaborative research on advanced fuel technologies. And then I already talked about this relaxed time, and temperature criteria that we're working with. And I think I'll just kick it to Kurshad to talk about it a little bit more.

And then for fuel performance assessment wise, that's where a majority of our focus is, within the fuel reliability program, and we have been doing a lot of work on this area by doing some laboratory testing for the presentative chemistry conditions for both Ps, and Bs looking at testing these materials in kind of extreme conditions to identify the boundaries, and limits.

We are doing some atomistic modeling of coating behavior, which you're going to hear more later today looking at whether chrome coatings could be a barrier for hydrogen, or not. And developing new technologies, and I talked about the assessment, and also the general characterization of fuel to support the term mechanical model improvements.

For safety benefits, for high burn up safety benefits, I think Fred is going to talk more, and back end impacts, there are some benefits that we're seeing there that Bob Hall is going to have a detailed presentation there. We have done a MAAP analysis for accident benefits for near term, and long term of these concepts, and each of these concepts provides some additional coping time.

But depending on the scenarios, and the material, their coping time results could be different. But currently we are working on upgrading the MAAP pool by bench marking the QUENCH-ATF data coming from the OECD-NEA program. And then once the bench mark is completed, if there is significant changes in the tool, we may reassess these previous analysis for the coping time calculations.

So, there are some accident source term guidances coming from NRC that I think there are some impact to the current fleet, but those changes, or impacts are not driven by fuel burn up increase at least so far.

MEMBER REMPE: So, before you leave this, I've seen some of the other prior MAAP updates, and I'm all for increased enrichment, and higher burn up, and fuel economies, but one needs to think about that we don't have accident tolerant control rods. There's other components in the vessel that oxidize. With BWRs, there's the channel boxes, stainless steel components that can produce combustible gas, not just hydrogen, but other types of combustible gas.

And we don't regulate severe accidents, but one needs to not get too enthusiastic about accident tolerance of these advanced technology concepts. And is EPRI thinking about that? Because you might need to think about what might happen if the fuel rods stay there when the control rods are there, and that's more than what, I think MAAP can do.

MS. KUCUK: Right, so there are some work going on, I believe in Japan, looking at advanced control rods, accident control rods area. We are engaged, and monitor their developments, and so on. And for channels, definitely I think there were some considerations for silicon carbide channels instead of zirc channels. I mean, EPRI had done some work on this area before, but silicon carbide dissolves in oxidizing environments.

So, we need to look at some improved coatings, and things like that if that would be the case. But there's not any current activities on this area, I'm only referring to what was done previously.

CHAIR BALLINGER: This is Ron Ballinger, I'm going to make a semi heretical comment here. I divide these kind of things into two categories. The kind that says we need to be alive in the morning, and that is improved chemistry, improved burn up, improved things like that. And then the group which I consider to be hypothetical, and that is accidents that we're spending a ton of money on to categorize the behavior in a hypothetical accident.

And it makes me wonder whether, or not applying risk informed, or risk based even techniques might be a significant advantage. And I'm wondering whether you're doing that anyway, because as far as I know, the amount of coping time you get, increased coping time for ATF is within the noise of the uncertainty on any of these calculations. So, and to go to 24 month cycles, we hear that people are thinking about a single batch core in order to do that.

So, I'm wondering whether you're dividing your work into the first class, and second class, where there's a risk threshold that you're trying to apply here. I'm probably not stating it clearly enough, but you get my point. I mean we're not going to have another Fukushima in New York City.

MS. KUCUK: Right, I think the -- I mean, if I go back here, not this one. I think a lot of the effort is really on longer fuel cycles. I think that's where a lot of the PWRs are focusing on. So, the comment that you're making, those new technologies, how they are going to benefit the current fleet to keep them running, and operating right now rather than focusing on the very rare severe accidents, right?

So, I think that's where the kind of fuel performance, and fuel cycle economy, that's where the utilities are mainly focusing on, or interested with these new technologies. And all these 24 month cycle core designs they are looking at, and it may vary from plant to plant, depending on their duty levels, or what class of plant it is, some of them may already implement it, and some of them we really need to go to high burn up to be able to do that.

CHAIR BALLINGER: We'll get in this argument on the FFRD discussion, I'm sure.

MS. KUCUK: Yes. That's all I have. Yes, all right.

CHAIR BALLINGER: We are pretty much on schedule.

MR. WELLS: 15. All right, Craig, I think you're going to present, is that correct? If you want to come up?

MEMBER SUNSERI: By the way, Dr. Ballinger, the television show you were talking about is called The Adventures of Captain Marvel, the protagonist was Billy Batson, and he would yell shazam, and turn into Captain Marvel.

CHAIR BALLINGER: Remember Gomer Pyle? You do remember, he used to say that. It was kind of funny, because the man, I forget his name, maybe it was Gomer Pyle, he had a fantastic singing voice. So, he had two careers, he had this TV show --

MEMBER SUNSERI: Jim Neighbors.

CHAIR BALLINGER: What was it?

MEMBER REMPE: Jim Neighbors.

CHAIR BALLINGER: I was going to be on a mission from God to find out his name.

(Simultaneous speaking.)

CHAIR BALLINGER: On that note.

MR. HARRINGTON: On that note we'll switch gears from more directly fuels issues to XLPR. I'm Craig Harrington from EPRI's materials reliability program, and my colleague is on the line, who is not able to come today. And then Markus will be stepping in momentarily to handle the back half of this presentation. Our focus in this work is primarily to support the fuels alternate licensing strategy.

So, we'll be covering materials in that regard. The outline, I will cover the brief overview of XLPR, and the work scope, and then Markus will pick up the last three topics. So, throughout this project we've found that there are times when piping, and fuels speak a little bit of a different language. Sometimes we even use the same words, the same acronyms in different ways.

So, we've created a list of acronyms that you can refer to. Most I think will be fairly clear, and familiar to you, but a couple that maybe not so much are RVIN, RVON, SGIN, and SGON which are reactor vessel inlet, outlet nozzles, and steam generator inlet, and outlet nozzles. We'll be throwing those terms around quite a bit. So, XLPR was created jointly between the NRC Office of Nuclear Regulatory Research, and EPRI over about a decade, or so.

Intended to be a state of the art probabilistic fracture mechanics code, really for piping applications at this point. The intent behinds its creation was to provide new quantitative capabilities to analyze risks associated with nuclear power plant piping. When we refer to risks in this particular context, we mean either leakage, or rupture due to active degradation mechanisms such as fatigue, and primary water stress corrosion cracking.

So, it has an extensive set of modeling capabilities in addition to the fatigue, and stress corrosion cracking. A couple in particular worth noting are ability to address in service inspection, multiple types of mitigation, calculate leak rates, and also to consider the seismic effects. This is one of these.

So, XLPR is a probabilistic code, and so we wanted to point out a few differences between deterministic approaches, and probabilistic approaches. In a typical fracture mechanics analysis, we combine inputs including stress, materials toughness, crack growth rates, and crack size to determine a component life. In the deterministic space -- I didn't prepare these, so I didn't realize that I was going to have to click each time, but that's good.

Focuses everybody's attention in the right place. So, in the deterministic space, we would assume a particularly low -- no. We consider conservative values, high values in stress, low value in fracture toughness, and also high values for crack growth, and for an initial crack size. And you combine all of those to produce one single calculation that has embedded margins that are, and conservatisms that are really typically fairly poorly defined at best.

The probabilistic approach though, is a bit different. We performed numerous calculations, produced numerous results sets with inputs that are different for each realization of the problem, and those inputs are sampled from distributions that represent each uncertain input. Results are then aggregated to produce the failure probability over time, determine the best estimate value for each of the various figures of merit that we are trying to evaluate.

And it gives us the ability to control the level of conservatism that is included in those results. XLPR is a probabilistic code, internally it really just performs a set of deterministic calculations. But it does so repeatedly. Makes sure to fully interrogate the problem space. Traditional --

MEMBER DIMITRIJEVIC: Sorry to interrupt you, but I was curious always, performed sets of deterministic calculations, is there a certainty associated with those calculations, those equations other than just uncertainty in input data? I don't know, did I make myself clear? Is there multiple uncertainties considered also?

MR. HARRINGTON: Missed part of that.

CHAIR BALLINGER: Yeah, for the record, that's Vesna Dimitrijevic, can you repeat the question then Vesna?

MEMBER DIMITRIJEVIC: Yes, so my question here is the uncertainty -- the difference which you explained to us is that we have uncertainty in input data considering probabilistic approach versus just using the value in deterministic approach, conservative value. My question is, is there also uncertainty in the model, in the calculations, equations connecting those input data considered?

MR. HARRINGTON: Yes, there certainly is. There's uncertainty in the models, there's uncertainty in the data, so yes, we have to deal with uncertainty throughout. And in the particular context of XLPR, in its development, we put a lot of effort into trying to, as best we could, assess that, evaluate that, characterize that. So that in the end we have a reasonable understanding of where those uncertainties fall.

And to what extent we have bias in the overall code. So, we looked at that for each of the component modules within the code, and then actually have a report, an uncertainty report from the development of the code that looks at those individual pieces, and rolls that up into an overall understanding of the degree of uncertainty, and the degree of bias that is present there. So, yes, it is an important aspect of this whole problem.

MEMBER DIMITRIJEVIC: All right, thanks for that, that's what I was curious about, thank you.

MR. HARRINGTON: Sure. So, traditional

deterministic analyses are typically much more straight forward to perform. But probabilistic approaches, while more complex, can also be more informative. So, to shift to the work scope, this is specific to the support that they're looking for within the fuels area for the alternate licensing strategy.

We have looked a bit more broadly at the topics here, but what we'll focus on today is specifically the fuels area. So, we are attempting to validate the LOCA frequency estimates that are published in NUREG-1829. This is a report that's been out for a decade, or so. NUREG-1829 produced LOCA frequencies as a function of line size, and it implemented an expert elicitation process, a very complex process, I might add.

That process also involved probabilistic fracture mechanics, but in a less direct manner. They were used to assist in bounding the problem, and in forming the elicitation process, but it was not a directly analytical approach, it was an elicitation approach. XLPR in our case is being used to directly provide an order of magnitude comparison to our results, and the 1829 results.

In addition, XLPR can provide statistics on the time between detectable leakage, and rupture. This is essentially a leak before break type of check. Classical deterministic leak before break analysis procedures basically just compared leakage, and rupture flaw sizes, and time is not a factor in the analysis.

With an XLPR, we track crack progression from initiation through growth to leakage, and eventually to rupture. So that the time between critical stages is easily obtained from those results. The key outputs that we've investigated in this particular project as it relates to fuel fragmentation, relocation, and dispersal, FFRD, are the probability of LOCA.

In this case we use rupture as a proxy for LOCA, and the probability that leakage as a precursor to rupture will be detected in sufficient time to shutdown the reactor. Detecting a leak allows the operators to shutdown prior to the rupture, eliminating the loads that drive crack progression, and thus preventing that rupture.

Alternatively from a more conservative ECCS perspective, postulating that despite having shutdown, the crack does still occur, or the rupture does still occur, there will be a beneficial plant shutdown, reduction -- or post shutdown reduction in decay heat levels before that postulated rupture would occur. So, this is a little bit of an eye test, not really intended to have to dig in, and read all these numbers.

But line size is a primary attribute of the analysis scope that we've undertaken. This slide shows tables 1 on the left, and 3.5 on the right, both from NUREG-1829. Table one shows the LOCA frequencies with respect to line size, and it's broken down by PWRs, and BWRs, we're focused on the PWRs part at this point. Line 3.5 shows the different systems.

They looked at piping both in size, and various systems, and the kinds of degradation mechanisms that would be relevant, or those systems along with other attributes of consequence. Separately from our work, the ALS team has generated line size information at which FFRD becomes a concern. They will cover that in more detail in, I think the next presentation probably. Fred's shaking his head.

But for our presentation, FFRD was determined to only really be a concern for line sizes above NPS14. From the NUREG-1829 tables, the only lines, and not just from their tables, but from what we know of the plants, the only lines that then are relevant, that are above NPS14 are the reactor coolant loop piping. So, for our XLPR analysis work that we'll talk about today, the scope is limited to the hot leg, the cold leg, and the crossover leg.

And it's worth noting that in the NUREG-1829 analysis, they broadly considered a range of degradation mechanisms that could be relevant to the piping involves. XLPR presently only evaluates explicitly fatigue in primary water stress corrosion cracking. And in general, PWSCC is limited. So, in this light, to try to close that gap between the broader degradation mechanism approach in NUREG-1829, and the more limited capabilities of XLPR, we referred to the EPRI materials degradation matrix.

Which is really the guidance document that industry uses to proactively manage materials degradation issues across the fleet. The materials of interest here are the 300 series stainless, and alloy 80 to 182, the similar metal welds. So, those, we've looked at the degradation methods that are included in the MDM, rigorously identified, assessed.

And for the ALS scope of interest, the broader set of degradation mechanisms covered in NUREG-1829 are either evaluated explicitly in the case of fatigue in PWSCC, they are well addressed by other industry guidance such as in the case of thermal fatigue, or simply are not applicable to the piping systems that we're dealing with.

So, based on that comparison, we have concluded that XLPR studies that we've performed do provide valuable insight into the conservatism, or non-conservatism of the NUREG-1829 LOCA frequencies despite the differences in degradation mechanism, scope. So, with that, Markus will pick up, and go through the details of analysis.

CHAIR BALLINGER: This is Ron Ballinger, before you get up, I have a couple of questions. XLPR, Dave Rudman is not here, so otherwise he would be jumping up, and down back there. But XLPR, first off, initiation is not considered, so the crack has to exist already --

MR. HARRINGTON: No, we do consider initiation.

CHAIR BALLINGER: Okay, I thought that --

MR. HARRINGTON: I mean this is really the first time in analysis space that we've built a code that does incorporate initiation.

CHAIR BALLINGER: Okay, good, that's new then. The second thing is has anybody done an analysis where we've compared the XLPR crack progression, and leakage, the crack size results, with when you hit the unidentified leakage limit in a PWR? And so, in other words, you propagate until you get rupture, but rupture is never going to occur we hope, primarily because of the unidentified leakage limit, which is there.

So, has anybody ever done the analysis of will we actually have a problem because of the unidentified leakage limit that would be exceeded long before we get rupture?

MR. HARRINGTON: Yes, all that is very thorough, in fact Markus will cover a lot of this, but just a couple points on initiation. We do have initiation models built in to the code, but we also have the ability to side step that, and use an initial flaw size. So, we have both of those capabilities. Within the code then, we also have leakage evaluation, leakage results.

So, we monitor from the very beginning of when the pipe crack first goes through a wall, and we start calculating the leak rate through the wall. So, we have the progression of leak rate over time. And in this work we focus a lot of attention on that progression, so we can -- and Markus will report results looking at either one GPM, or in leak before break terms, ten GPM is an important number as well.

We evaluate both of those numbers, or other leak rate values that we might choose to look at. And can therefore easily determine when you would reach the officially detectable leakage of one GPM typically in tech specs, as opposed to pipe rupture. So, in LBD space, you're just looking at crack sizes, you don't care how long it takes to go from A to B.

In our work, both in this project, and in prior work with the NRC research on leak before break, we've looked very rigorously at what is that time between detectable leakage, and rupture, and find most of the time, not surprisingly, it's a significant amount of time. So, yes, it's a very important part of this work, and one that we'll spend a lot of time on.

MEMBER HALNON: Craig, one the one with the -- some are hot leg crack, did you compare, or overlay that whole even with what you're doing here, does it make sense when you set that?

MR. HARRINGTON: We certainly made a significant attempt to do that. It is problematic to compare a single event to a probabilistic result. But this -- the code calculates probabilities, and so those two things -- yeah. Those two things are just hard to reconcile directly. But we did, in the verification, and validation of the code at the end of development, we looked at those kinds of events.

That one in particular, and others that at least, obviously we've not had a rupture, so we don't have that to bench mark against. But as different precursor events of leakage, and cracks developing, and things like that. So, we did take a comprehensive approach at trying to bench mark as best we could against those kinds of events.

CHAIR BALLINGER: Yeah, and we have to be careful, I'm going to use a little bit more colorful language, but the VC summer weld was anything but typical.

MEMBER HALNON: It's true.

MR. HARRINGTON: And that creates part of the challenge, because there's a lot of uncertainty in, in particular the weld residual stresses within that weld, and --

MEMBER HALNON: In the early times during the root cause, since I was there leading it, we were concerned about the safety implications relative to a potential rupture, how much time did we have before the rupture? And obviously this code didn't exist then, but it really factored into the regulatory response to it. And that's what I was getting to, is were we unfounded in our fears, just because of the potential?

Or were we well within our reasonable sphere of being concerned about the impact of the rest of the fleet, and potential other welds? I mean we certainly had the rest of the fleets attention, and I say fleet as in PWRs, especially the Westinghouse, attention on it, and that was a concern that may have caused some unnecessary inspections, shutdowns, other things.

So, that's what I see the value of this is. When we do come up to the event, we can maybe not react so emergency, or over react, I guess, not under react obviously, but certainly not over react.

MR. HARRINGTON: I think we maybe did a better job in not terribly over reacting. In that case, we struggled a little bit more with your Wolf Creek thing, but that was a bit more of a dynamic response by the fleet, or part of the fleet. But yeah, those were fun times.

MEMBER HALNON: You don't have to tell me.

MEMBER DIMITRIJEVIC: Hi, this is Vesna Dimitrijevic again. I just want to clear something, you already responded to Ron sort of on the initiation. So, this is a time step process, right? So, what is your time zero situation? You have a certain distribution of the flows in the sizes, right? That's where you tart. What is the situation in time zero where you start your progression?

MR. HARRINGTON: What's the situation at time zero?

MEMBER DIMITRIJEVIC: Yeah, what is happening -- when you start your progression in time zero, you have a different flow distribution, right, size distribution?

MR. HARRINGTON: You mean in general with XLPR --

MEMBER DIMITRIJEVIC: Yeah --

MR. HARRINGTON: Or for this particular project? In general the pipe is assumed to be pristine, and not cracks. The initiation models within XLPR account for the incubation time for crack initiation. And then we don't get into the very tiny details of initiation. When initiation occurs, we consider it to be a crack of engineering scale that responds to fracture mechanics kinds of concepts.

But we do account for that incubation time, and then a crack appears, and follows a growth line. Does that help?

MEMBER DIMITRIJEVIC: So, what you said to Ron is exactly right, you didn't assume an existence of the cracks, or flaws, you just used pristine piping, I see.

MR. HARRINGTON: We can, but if we use the initiation process, then a bit part of that is to recognize that cracks typically do take some time to incubate, and develop before they become meaningful.

MEMBER DIMITRIJEVIC: Right, because in probabilistic versus deterministic you show distribution of the crack size as input into the process. That's why I sort of got confused after you said that no assumptions were made of this.

MR. HARRINGTON: And the crack size is represented by distribution. So, when a crack -- when we work through the initiation modeling process, there's an incubation time, and then the crack initiates, and the actual size of that crack at initiation is also something that we can represent by distribution, and sometimes it'll be a very tiny crack, sometimes it'll be a larger crack.

MEMBER DIMITRIJEVIC: So, what are your assumptions then? So, then what is the leak rate which you assume the leak occurs, do you have, or you are analyzing different leak rates, you know, are they like one gallon, five gallon per minute, or ten? And then when do you assume -- what is the leak rate when you assume the actual rupture occurred?

MR. HARRINGTON: Well, the rupture end is determined by fracture mechanics stability calculations. It's not driven by leak rate. So, at that point we're just tracking what the predicted leak rate is at each point, and then we can look at that after the fact to understand what the leak rate was at different points in time, different points in crack size on both ends.

MEMBER DIMITRIJEVIC: So, what is your definition of failure then?

MR. HARRINGTON: That's a complicated question, it's one that we've spent a lot of time discussing, and considering in the early development of XLPR. Failure depends on a lot of different things, and often failure is defined as rupture. But with the details that we have at our access in XLPR, you can define failure as one GPM leak rate, as some other particular LOCA size in flow rate terms, or rupture of the pipe. So, it gives you a lot of flexibility.

MEMBER DIMITRIJEVIC: Well, you are comparing this with the new LOCA data, which is clear the size of the LOCA is defined there, so that's what I was wondering, how you compare that. So, do you have -- if you want to compare the results, you should have a similar failure definition, right?

MR. HARRINGTON: Yeah, the failure definition is a challenge. In the regulations it just refers to rupture. But that is something we spend a lot of time on, and in my comments a few minutes ago on one of the earlier slides, I had said that we're using rupture as a proxy for LOCA. With LOCA, that's really defined in flow terms primarily. Rupture is defined in stability terms, and those are not exactly the same.

MEMBER DIMITRIJEVIC: Definitely not the same.

MR. HARRINGTON: But in this context, we're using rupture as a proxy for LOCA.

MEMBER DIMITRIJEVIC: Okay, I've got good news, and bad news. The good news -- well, the bad news is that we're a little behind. The good news is it doesn't matter, because we're going to make up a half hour for lunch because we can meet downstairs. But what I'd like to do is we're scheduled for a break now, this is a sort of semi convenient way to do a break.

So, I would like to take a break now, our normal break from now until 10:40, and then we'll pick up the XLPR thing, is that agreeable folks? Thank you. So, we're all recessed for 15 minutes.

(Whereupon, the above-entitled matter went off the record at 10:26 a.m. and resumed at 10:40 a.m.)

CHAIR BALLINGER: Okay, so let's go back in session, and I think Markus can pick it up.

MR. BURKARDT: Great, thank you very much.

Yeah, Markus Burkardt at Dominion Engineering, and I'll continue the discussion of the xLPR analysis work that we've been doing.

And so the xLPR analysis cases that we've been looking at have been applying PWSCC and/or fatigue material degradation mechanisms.

And so here when I talk about fatigue, I'm talking about fatigue that's driven by plant transience rather than like fatigue driven by local thermal fluctuations or vibration. So those are the kind of key material degradation mechanisms that are modeled within xLPR.

The analysis cases, as Craig mentioned, consider flaws of engineering scale. And so those flaws, we can either have them be present at the start of the simulation and grow over an 80-year plant life period, or we can use initiation models to calculate the time to initiation

And then the flaws start growing at an engineering scale of, you know a couple millimeters, something that has k-controlled crack growth. And then have them evolve over time from there.

And in this work, we performed many sensitivity studies to determine the impact of changes to key analysis inputs. And I'll go into a little bit more detail on the specific sensitivity studies looked at later. But some of the parameters that were changed in these sensitivity studies include geometry, loading, welding residual stress profiles, or initial flaw sizes.

There are several outputs that we looked at as part of this work, some that are output directly by xLPR, and others that require a little bit of close processing. For the directly output outputs, the probability of rupture is kind of a key output that we're looking at. And that's used to calculate the rupture frequencies from comparison to NUREG-1829.

With xLPR, when looking at the probability of rupture output, you have the option of conservatively not crediting in-service inspection or leak rate detection, or you can credit in-service inspection, or you can credit leak rate detection. Or you can credit in-service inspection and leak rate detection.

And so for cases the utilize the initial flaw model, those results are then conditional on crack initiation actually occurring. And so to kind of consider all of these cases and results on the same baseline, we also consider probability of crack initiation for cases that model that explicitly.

Additionally, if we're looking at time between detectable leakage and rupture, the leak rate is a key output that we look at as well.

For the results then that are post-process, using the leak rate data as well as the rupture time, we then calculate the time between one gallon per minute detectable leakage and rupture. And so in some cases in the slides, I might have used some shorthand here and called this the lapse time.

Then for cases where we obtain probability of rupture using the initial flaw model, you know, that's conditional on crack initiation. So as an approximation, we then scale that by the probability of initiation to approximate the probability of rupture, given crack initiation.

And then another approximation that we make as part of this comparison to NUREG-1829 is we take the probability of rupture at 80 years, average that over the 80-year time period, to come over an average 80-year rupture frequency. And as Craig mentioned, we use rupture as a proxy for LOCA in this comparison. So there's --

MEMBER DIMITRIJEVIC: This is Vesna Dimitrijevic. Again, I have a question. Sorry, I'm trying to understand, I'm very interested in your results on how they compared with the LOCAs since I'm PRA person.

So is this -- the input output you get from xLPR, what is this, is this per weld? For all pipe, per, you know, foot? What is the, you know, tell us the tactic of this ATS. What is the piping? The old plus-one piping? The, you know, per foot per weld?

MR. BURKARDT: So xLPR, in xLPR we model flaws within just one weld. And so this is basically a per-weld type result. But then in our work, we've looked at many different types of welds and considered, you know, the results from those many different welds.

MEMBER DIMITRIJEVIC: So but you compare

it with the, the number that you get, you assume some average number of the welds in the, you know, plus-one piping, is that how do you get total result? I mean --

MR. BURKARDT: No.

MEMBER DIMITRIJEVIC: in reference in size of the piping. So you know, in the -- how did you compare the results?

MR. BURKARDT: I think I'll get to this a little bit later in the presentation. And if you still have questions on the comparison, at that point maybe we can speak to that then.

MEMBER DIMITRIJEVIC: All right.

MR. BURKARDT: If that works for you.

MEMBER DIMITRIJEVIC: Yes, of course.

MR. BURKARDT: Thank you. So there have been some other recently performed xLPR studies that have been published by the U.S. NRC in two technical letter reports in the context of leak-before-break analyses for alloy-82/182, dissimilar metal piping butt welds in PWR piping systems.

And so this work was performed under a memorandum of understanding between NRC and EPRI, where the NRC and EPRI teams worked together in developing the set of cases to be considered. But then each team independently developed the inputs for those cases, ran those cases, and interpreted the results.

So these two technical letter reports basically document the NRC's work and from -- so this is NRC Research who performed this work and then published those reports.

The first of the two technical letter reports that I'll speak to is the piping system analysis. And so this looked at a representative reactor vessel outlet nozzle and inlet nozzle in a Westinghouse four-loop PWR. And so for this technical letter report, we looked at an extensive set of sensitivity studies.

From the learnings of this report, the xLPR generalization study then looked at a much broader range of welds, looking at all the other outlet-82/182 dissimilar metal piping butt welds that had prior leak-before-break approvals from the NRC staff.

But the set of sensitivity studies per component was then greatly reduced based on the kind of findings and results of the piping system analysis and basically focused in on the ones that had greater effect and greater importance.

And so the results from these two studies in the work that I'm presenting were used where possible, but then supplemented with additional xLPR analysis cases, as needed.

And so there's many, many inputs to xLPR, thousands of inputs. And so it's tough to summarize all of those, but here are some of the key analysis summary inputs for the bases cases for each of these piping systems.

Here, colors on the plot highlight consistent wording. And then the blue box highlights the main loop piping welds, which are the focus of the ALS work.

And so here we basically looked at several different welds, welds in the reactor vessel outlet nozzle, the reactor vessel inlet nozzle, the steam generator inlet nozzle, the steam generator outlet nozzle, as well as the reactor coolant pump inlet-outlet nozzle welds.

And then also in the pressurizer surge nozzle. And then in CE hot leg branch lines and CE cold leg branch lines.

And so for the base cases, we model PWSCC crack growth rather than fatigue crack growth. We also explicitly modeled initiation. In most cases, we modeled both axial and circumferential flaws, but for the piping system analysis, only circumferential flaws were considered initially.

Some different approaches were taken to model in seismic occurrences across the collection of these analyses. Most of the base cases don't consider mitigation. And in-service inspection leak rate detection are optional in the outputs that are considered.

And this cart kind of highlights the sensitivity studies that were performed. As you can see for the piping system analysis, there was a much longer list of studies that we considered.

We looked at initiation, at welding residual stress, at earthquakes, at normal operating thermal loads, changes to those, change to leak rate detection, changes to ISI modeling, application and mitigation. You know, what does fatigue mean. Changes to initial flaw size, geometry, consideration of axial cracks, hydrogen concentrations, different temperatures.

And so then the learnings from here we then had a narrowed scope of sensitivity studies that were considered in the generalization study. We were really focused on initiation, on welding residual stress, on mitigation, in some cases fatigue.

So now getting into the comparisons with NUREG-1829 and xLPR. There are some differences that go into this comparison. And so kind of do the best we can here.

For context on the 1829, the LOCA frequencies that we're using for comparison were based on expert elicitation processes Craig discussed. And we're taking the results from Table 1 for this comparison. These are kind of the base case results from NUREG-1829, and that table summarizes median, 5th percentile, and 95th percentile results.

Here, those results are total PWR LOCA frequencies after over-confidence adjustments using an error factor scheme. They are 40-year fleet average values. And they consider typical in-service inspection and leak rate detection, as required by plant technical specifications.

And so these results are presented on a per-plant basis for each distinct LOCA category or LOCA size. And also the results in that table consider both the contribution of piping and non-piping passive systems.

And so there are a couple of differences between these results and the results that, you know, we're showing from xLPR. But we'll get into that. And so the xLPR results, they are 80-year results. We're looking at results from just one weld.

There's some differences in the material degradation mechanisms that, you know, that are considered in both of those efforts. But we're still trying to make comparisons here as best as we can.

Now, these two plots show with the gray, orange, and blue lines the LOCA frequencies from NUREG-1829 from Table 1. And then in -- with yellow points on both of these figures, we show the xLPR results that are basically rupture frequencies when considering leak rate detection and taking credit for that.

On the left figure, we're only crediting leak rate detection, on the right figure we're crediting both leak rate detection and in-service inspection. And so there are only, of those cases that I mentioned earlier, there are only three which actually have a non-zero occurrence of rupture with leak rate detection or with leak rate detection and in-service inspection.

MEMBER HALNON: Markus, do -- when you say leak rate detection, are you starting your lowest threshold 1 GPM, that's when you start detecting the leak?

MR. BURKARDT: Yes.

MEMBER HALNON: Okay, because most plants can do -- weld better than that, especially since the Davis-Besse.

MR. BURKARDT: Exactly, yeah, the one -- the one GPM is a conservative --

MEMBER HALNON: These are even conservative from that standpoint.

MR. BURKARDT: That's correct.

And so yeah, so we show those points explicitly. And so also all three of these points are sensitivity studies that are cases where there's modeling in xLPR that's not fully representative of plant conditions and operations, like cases where application of an overlay ultimately leads to the cause of a rupture. Or where there's flaws that have initial depth deeper than the depth of an inlay.

So cases like that. But so as relevant to the ALS, we also further investigate those cases and speak to those a little bit more later in the presentation.

Then for --

MR. BLEY: It's Dennis Bley. Checking points across those, it looks like the ISI essentially has no impact all on here. Is that because they're so far apart, or do you know what's going on? Is there enough here to say ISIs are not doing us any good? Leak rate detection?

MR. BURKARDT: You're looking at the yellow dots between the left figure and the right figure?

MR. BLEY: Yeah.

MR. BURKARDT: And so the yellow dots drop by two orders of magnitude in terms of LOCA frequency once you credit in-service inspection. In addition to leak rate detection.

MEMBER MARCH-LEUBA: The lines don't change from left to right because they're --

MR. BURKARDT: The lines? Oh, the NUREG-1829 lines already consider both in-service inspection and leak rate detection, as I mentioned in my opening.

MEMBER MARCH-LEUBA: So they're both the same.

MR. BURKARDT: So they're both the same, and it's just the xLPR results that I'm showing relative to those numbers and how those change. And so I show the cases with non-zero occurrence of rupture with leak rate detection explicitly on both of the figures. And then on the figure on the left, also showing cases where there were no ruptures with leak rate detection.

And so here I calculate a 95% upper bound based on a one-sided conference interval using a binomial distribution. And so that considers the number of realizations.

And for cases that are utilizing the initial flaw model, we're also scaling back then by the probability of initiation. So that given that those cases would otherwise be conditional on crack initiation.

And so those are all shown with the green points with arrows pointed downward to indicate that if additional realizations were evaluated in xLPR. If there are no ruptures are predicted, then those probabilities would be even lower.

MEMBER DIMITRIJEVIC: So did you assume the probability of detection is 1, both in ISI and in the, you know, leak detection?

MR. BURKARDT: So for in-service inspection, the probability of detection is based on a logistic model that's informed by the EPRI performance demonstration initiative. And so no, it's not a -- not a probability of 1 there.

For leak rate detection, yes, that is closer to like a probability of 1 type model. There is some uncertainty applied to the leak rate in comparison to the leak rate detection thresholds within xLPR. And that's more of deterministic assessment as to whether that leak is detected or not prior to a rupture occurring.

MEMBER DIMITRIJEVIC: This is all on one weld, right, so that's --

MR. BURKARDT: That's correct.

MEMBER DIMITRIJEVIC: Okay, so interesting, yeah.

MR. BURKARDT: And so another output, or I guess the last point here was just that when considering in-service inspection and leak rate detection, the LOCA frequency is estimated from xLPR, albeit with slightly different, you know, assumptions and considerations that go into the comparison or on a similar order of magnitude as the median LOCA frequency estimates from NUREG-1829.

So then another --

MEMBER DIMITRIJEVIC: But your estimates are just for one weld. So even if we assume there is only one weld in Class I, you know, exposed to that degradation mechanism, I mean, I don't really know how good is this comparison. We need to think a lot about that, actually, you know, so.

Because this, you know, maybe they are, you know, comparing apples and oranges. You know, that if you're looking just in one weld exposed to that specific degradation mechanism and on a different timeframe, I don't know how well that compares to the NUREG, so.

MR. BURKARDT: So for these welds, the kind of key -- or in these piping systems, the welds are of particular interest or concern are the 82/182 dissimilar metal welds. And in like one loop or one plant, there's really only a handful of those.

And so at that point, you're considering whether you're looking at results from just one weld or, you know, maybe up to eight welds.

And there you're then -- some things to consider is how do you combine the probabilities of failure from those individual welds and do you just like do you combine those as a individual probabilities that are -- that are unrelated, or do you consider them as having like a related probability of failure.

And so there's different approaches and methodologies that have been discussed for that sort of comparison within like the xLPR work that's been done.

MR. BLEY: Hey, this is Dennis again. I'm just trying to understand this picture. The main -- well, we're looking at two main things, I think. One is the three orange dots at about a break size of ten and a little over 30. And they drop by two to three orders of magnitude.

Then you have the little green ones, which are the 95% upper bound. And two of three orange dots are above that 95% upper bound. And that's just a result of the calculation of the probability, the uncertainties in there?

MR. BURKARDT: Yes, so those three dots are fairly extreme cases that aren't really representative of plant conditions and operations. And so that's why those dots are, you know, have higher probabilities.

And for the green circles with the arrows, there no ruptures with leak rate detection are predicted by xLPR.

MR. BLEY: Okay, and the length of the arrow means something?

MR. BURKARDT: The length of the arrow does not mean anything.

MR. BLEY: Okay.

MR. BURKARDT: Just the position of the circles means something, and that's based on the number of realizations that's evaluated and also the probability of initiation.

MR. BLEY: I think I'm beginning to get it, but not wholly. Okay, thanks.

MEMBER KIRCHNER: Yeah, may I follow on and ask what dominates the results that you're getting? Is it the assumptions on crack growth or crack size that actually dominate the results?

MR. BURKARDT: So crack initiation is a dominating factor.

MEMBER KIRCHNER: Yeah.

MR. BURKARDT: And that's a secondary factor, factors leading to more rapid crack growth are kind of the two key mechanisms.

MEMBER KIRCHNER: Like corrosion or whatever.

MR. BURKARDT: That's right. Yeah, basically for like primary water stress corrosion, cracking, higher stresses, higher temperatures , things of that nature.

MEMBER KIRCHNER: Now, how does the overlay of seismic impact these results? Is that a major contributor to the stress, or does it just shift the curves, almost, you know, doesn't change the shape or anything? It just shifts them in terms of probability space?

MR. BURKARDT: So seismic in xLPR is modeled in two different ways. One, in one way it is basically modeled as additional stresses that occur at some periodicity. And then that's considered in stability calculations that are being performed.

And so when seismic is modeled, basically you have slightly elevated probabilities of rupture due to that. But there's no -- yeah, we haven't, so far we haven't see that to be --

MEMBER KIRCHNER: You don't see a cliff edge effect with the seismic considerations. In other words, something equivalent to like brittle fracture, where you just get a large, a resultant large rupture that goes beyond just from kind of propagating a crack over time with stress and corrosion factors.

Do you see a step change with the seismic stresses added to the model?

MR. BURKARDT: No, we do not.

MEMBER KIRCHNER: So you don't see any cliff edge effects? So this is within the SSE spectrum and -- I'm just trying to understand.

Usually, for a lot of these kind of issues, seismic is often dominant. So you're staying within the SSE and then using the spectrum, the implant structural motion spectrum, or?

MR. BURKARDT: So xLPR doesn't include like spectral seismic analyses. We can only input like a specific earthquake like magnitude, like in terms of piping stress. And then a specific frequency of occurrence, rather than a spectrum of frequencies of occurrence and a spectrum of, you know, seismic stresses.

And so yeah, the kind of stresses that we apply for this are the, yeah, for the SSE-type seismic events. And we pick a typical like seismic occurrence of that event as part of those analyses.

MEMBER KIRCHNER: And those are factored in these results we're seeing here, or they are --

MR. BURKARDT: They are.

MEMBER KIRCHNER: They are. Because you took the seismic out with those two dots have a much lower frequency?

MR. BURKARDT: Not substantially, I believe. Slightly but not substantially.

MEMBER KIRCHNER: Okay, thank you.

MR. BURKARDT: Go ahead and move on.

There's another output that we look at, which is the time between detectable leakage and rupture. And so here, just I want to provide a little bit of context first just on kind of help unpack some of the later slides that I have here.

And so xLPR models many, many realizations in one analysis case. And so each individual realization is basically looking at flaw growth in evolution within a specific weld.

And so here I've just picked an example case to kind of depict what that looks like in terms of detectable leakage to rupture and how the leak rate evolves over time for that type of like sample case.

I have the details listed on the left here, but that's not important for the purpose of this discussion. And so really it's just that you go from a part through-wall flaw to a transitioning through-wall flaw. That transitioning through-wall flaw then starts to leak, and then continues to grow until you get an idealized through-wall flaw.

That flaw then continues to leak further as the flaw grows more and more around the circumference of the welds, until eventually rupture occurs. And so the leak rate basically evolves over time and we're calculating the leak rate based on the flaw size as part of this assessment.

So this is for one realization. And in an analysis case, there can be many realizations that result in rupture. And so you can kind of build a distribution from that. And then we use a couple of terms to help describe that distribution that I'll summarize in some of the later slides.

And so, you know, we look at the mean. We look at the standard error. And so in later slides I'll have error bars on the mean that show that. And here by standard error, I mean the standard deviation divided by the square root of the sample size.

We also look at the minimum, as well as a 95 tolerance interval, assuming that the data are locked normally distributed. So this is kind of we have a distribution of these times from detectable leakage to rupture for an individual case.

And so then we look at the collection of cases and look at the summary statistics on the times from detectable leakage to rupture for all of those cases for additional context.

And so this is just kind of a screening exercise that we perform. The slide that I'm showing now basically shows the mean times from detectable leakage to rupture for all of these cases.

I just want to highlight the cases of importance for the ALS work or the reactor coolant piping hot leg and cold leg. So those are shown in orange and blue.

And then also I wanted to make a distinction between the base case results, which are circled, what have points circled in black, and the sensitivity studies, which have no black circle around the points.

So in addition to looking at the kind of distribution of mean times between detectable leakage and rupture, we also looked at the minimum times. So this is now for an analysis case that could have many, many times between detectable leakage and rupture.

The very minimum of those individual cases, we look at those and we use that as a screening exercise where we then do further investigation of the cases that have relatively short minimum times from detectable leakage to rupture, under three months. And so I'll get into those specific cases further in the presentation.

But all of those cases are sensitivity studies, and they either considered unmitigated welds subject to primary water stress corrosion crack growth at either the hot leg or the pressurizer temperatures.

Or they included modeling that was not representative of plant conditions and operations.

And so I just want to emphasize the unmitigated statement here in that all currently existing 82/182 welds at pressurizer temperature are now mitigated. And a large majority of the components at hot leg temperature are also mitigated. So it's just kind of the detail to point out there.

CHAIR BALLINGER: Yeah, this is Ron Ballinger again. Those last two slides, we recenter ourselves. xLPR was originally built to just look at crack growth. You're applying gear to LOCA issues.

And so those last two slides are pretty key, with all the caveats that are involved, what they're telling us, and by the way we're likely to see this kind of analysis as a committee going forward for other -- you know, we haven't see this yet.

The chances of a non-detectable leak giving us a problem is very low. Is that the message I'm taking away from here?

MR. BURKARDT: That's correct.

CHAIR BALLINGER: Because like, remember Halnon said one gallon a minute, that's an upper bound. We'll see that way below one gallon a minute. And so if you factor that in over here, that reduces the likelihood, I won't say probability, that you'll miss something. So very, very low value.

MR. BURKARDT: You'll reduce that detectability threshold for unidentified leakage, and then that brings the time at which you would detect that to the left giving you further temporal margin to eruption.

CHAIR BALLINGER: You're basically risking forming the LOCA analysis. That's the whole purpose of this.

MR. BURKARDT: Mm hm.

CHAIR BALLINGER: Thanks.

MEMBER KIRCHNER: There's some detail on that slide that's hard to extract, but let me see if I can put it into a question. So the surge line I presume is the pressurizer, and that sees a lot more transient. Is it dominated by fatigue or stress, corrosion, cracking?

MR. BURKARDT: A lot of the pressurizer cases there are modeled as being unmitigated. So a lot of that is just due to the elevated temperature.

MEMBER KIRCHNER: Right.

MR. BURKARDT: Elevated pressurizer temperature and subject to PWSCC growth in an unmitigated component.

MEMBER KIRCHNER: So those show on this probabilistic trajectory earlier detectable leakage before rupture. Or earlier rupture. How do you --

MR. BURKARDT: Shorter times.

MEMBER KIRCHNER: How do you read the cumulative distribution?

MR. BURKARDT: It's just a way to sort the data. It shows -- so the, you have a surge lines cases here show shorter times from one gallon per minute detectable leakage to rupture. And that's largely attributed to the faster crack growth, the pressurizer temperature. And --

MEMBER KIRCHNER: But how does one make this --

MR. BURKARDT: That's what I'm pointing out --

MEMBER KIRCHNER: Yeah, so but let's just pick a point that happens to fall on one of your grid lines. So you're showing the surge line there at .2 cumulative distribution and roughly minimum time. A least rupture two months.

MR. BURKARDT: Mm hm.

MEMBER KIRCHNER: And then the other green dots are just because of the different variations in input that you put in.

MR. BURKARDT: That's correct.

MEMBER KIRCHNER: So in a worst-case scenario, then, how do I read those dots that fall on the ordinate?

MR. BURKARDT: Those --

MEMBER KIRCHNER: That has no time to -- from detectable leakage to rupture? How do I read that?

MR. BURKARDT: So those are cases where you basically have rupture either prior to or at the time where you would have detectable leakage. And so those cases are cases that we then want to sharpen the pencil on, better understand, and look into further. So that's where my presentation is going next.

MEMBER KIRCHNER: Okay, thank you.

MR. BURKARDT: Okay, so looking at those cases, we basically performed further investigation of limiting cases that had either minimum times between detectable leakage and rupture, less than three months.

So that includes the ones with zero months times that we pointed out. And then also the cases, the three cases that had non-zero occurrence of rupture with leak rate detection, we looked at those as well.

And so then we reran these cases with refined time stepping or updated input model parameters that we felt were more realistic. We also investigated inputs to xLPR, intervening variables and outputs to better understand the applicability of the scenarios being modeled.

And so after this dispositioning, for the cases that are relevant to the ALS, the minimum time from detectable leakage to rupture for a base case is 14 months, and for a sensitivity study it is 0.8 months.

So then I mentioned we had also wanted to look at an additional figure of merit, the 9595 tolerance interval. And so for the cases that had these limiting minimum times, we computed a 9595 tolerance interval using a log normal distribution and explaining the data.

And so this is defined such that there's a 95% probability that the constructed limits contain 95% of the population of interest for the surveillance interval that's selected. And so when we look at this, then the 9595 lower bound for the most limiting of the sensitivity studies that's representative of the U.S. PWR fleet is 3.8 months.

And so just again highlighting the fact

that the sensitivity studies, they were less constrained to maintaining fidelity to realistic plant conditions. So some of them were perhaps a little bit more extreme and unrealistic.

And they're also defined to informed understanding of the base case results by investigating some of the key inputs that are known to have influence on the xLPR results.

CHAIR BALLINGER: This is Ron Ballinger again. Once again, these are postdictions.

MR. BURKARDT: Mm hm.

CHAIR BALLINGER: Primarily because all of these wells are mitigated.

MR. BURKARDT: Correct.

CHAIR BALLINGER: So we're basically predicting something that can't happen because the welds are -- the welds are mitigated.

MR. BURKARDT: Some are unmitigated. And in the sensitivity studies, we also looked at mitigated weld cases as well.

CHAIR BALLINGER: But I thought that you said that as far as you knew, in the fleet, all the pressurizer surge line wells have been mitigated or probably replaced.

MR. BURKARDT: But for the cases relevant to the ALS, it’s the main loop piping.

CHAIR BALLINGER: Okay.

MR. BURKARDT: And so those cases, there are plants that still have unmitigated hot weld hot leg components. And so those do need to be considered as part of that effort.

CHAIR BALLINGER: Okay.

MR. BURKARDT: I’m now jumping to my conclusions. So when we looked at crediting and service inspection and leak rate detection, the occurrence rupture results were on a similar order of magnitude as the NUREG-1829 LOCA frequency estimates. Acknowledging that there are some differences in better -- made as part of the comparison.

The only non-zero results that were found were for cases including modeling that was not representative of plant conditions in operations. And for the cases with zero ruptures with leak rate detection, we, for purposes of comparison, computed a 95% upper bound based on a one-sided conference interval.

Then for all of the base cases and most of the sensitivity cases, considered minimum times from one gallon per minute detectable leakage to rupture exceeded three months. And the 9595 tolerance interval for the limiting sensitivity study that was representative of the U.S. PWR fleet, the lower bound there came out to be 3.8 months.

That's all I have for the prepared presentation. Open to any other questions you may have. Thank you for listening.

MEMBER DIMITRIJEVIC: I'm curious, did you look in the -- did you compare that with the results form like in-service inspections, you know, through the history of that? I know you don't have anything on the ruptures, but you may have detected leaks, and you know, and all that degradation mechanisms.

Did you try to compare your, you know, soft results with empirical data?

MR. BURKARDT: So the initiation models were calibrated considering like in-service inspection results. Not only those but also laboratory data. So they were considered in that manner.

And they've also been, I think not directly the plant data, but the performance demonstration initiative, like calibration mockup or test mockup specimens, which are meant to be, you know, fairly realistic to plant components. Those have been used in calibration and development of the in-service inspection probability of detection and sizing models.

There's been other validation that's performed for, you know, all of the individual sub-models. And as part of those efforts, if field data was available, field data were used in those validation efforts. But in some cases, as you pointed out, like for rupture, there wasn't a lot of field data available, or any.

And so then, you know, laboratory data had to be considered instead as part of those validation efforts.

MEMBER DIMITRIJEVIC: Thank you.

CHAIR BALLINGER: Questions from members? They're open. We just eliminated the quandary.

This is a convenient place to break for lunch, even though it's bit early. The folks downstairs are open. So we can do that. And I understand, do we still have a problem with one of the presenters?

MR. WELLS: Maybe, maybe not.

CHAIR BALLINGER: So we --

MR. WELLS: Suresh, I think you're on, do you want --

CHAIR BALLINGER: Do we have a cumulative distribution for that?

MR. WELLS: Suresh, do you want to try to unmute yourself and just do a quick audio check?

CHAIR BALLINGER: Well, what I'm going to propose is that we break for lunch now.

MR. WELLS: Yeah, okay.

CHAIR BALLINGER: Take an hour.

MR. WELLS: Okay.

CHAIR BALLINGER: And then within that hour, hopefully we can get that sorted out. Because he's a presenter, so we got to be sure that we're under control on that one.

MR. WELLS: Okay.

CHAIR BALLINGER: So unless there are other circumstances that would say we don't do that, that's what I would like to do.

MR. WELLS: Well, the only consideration we've been trying in the background to figure out is Erich is actually in France. So the longer we wait, the later it gets there. But --

MR. WIMMER: That's no problem.

MR. WELLS: Yeah? Okay.

MR. WIMMER: For me, I'm fine.

MR. WELLS: Okay, all right. Thank you, Erich.

MR. WIMMER: I'm actually in Vienna.

MR. WELLS: Oh, you're still in Vienna, okay.

MR. WIMMER: Okay, so that's fine.

CHAIR BALLINGER: You don't need to throw him under the bus.

MR. WELLS: Yeah.

MS. KUCUK: So I think in that case Mr. Kucuk and Erich will be --

CHAIR BALLINGER: I'm trying to --

MS. KUCUK: He may not be on yet, so.

MR. WELLS: He may not be on yet. Okay, so we'll get him on the lunch break. And maybe try to come back to test him maybe ten minutes before we plan to start.

CHAIR BALLINGER: Very good, thank you very much.

MR. MOORE: So this is Scott Moore. We can turn off the room audio until ten minutes prior to we pick up? Okay, thanks.

CHAIR BALLINGER: All right, so I'm going to propose that it's now 11:20 -- we meet, we'll come back here at 12:30. Thank you.

(Whereupon, the above-entitled matter went off the record at 11:23 a.m. and resumed at 12:30 p.m.)

CHAIR BALLINGER: Okay, we're on the air. So thanks for coming back after lunch. If any of you have, any members have actually looked at the schedule, you're probably now going to be confused, because we've changed the order. So stick around and we'll eventually get to the presentation that you thought you were going to hear. So we're going to start with Fred Smith, right?

MR. SMITH: Yes.

CHAIR BALLINGER: And then we deviate from there. So go ahead, thanks.

MR. SMITH: Yeah. Thanks. So I'm going to keep along the same path as the xLPR work, because we're using that as part of the ALS strategy. And ALS, you probably wonder, alternative to what. And so the kind of traditional deterministic approach is what we would consider the normal approach to dealing with FFRD. ALS includes risk insights. They're still fundamentally a deterministic analysis. But there are deviations that we're including risk insights to modify the approach.

CHAIR BALLINGER: You might know that we reviewed the RIL, or risks, or whatever they called it.

MR. SMITH: The RIL, yeah.

CHAIR BALLINGER: And we wrote a letter, and that letter suggested that they use a risk-based informed approach, let's put it that way.

MR. SMITH: Yes. And I agree.

(Simultaneous Speaking.)

MR. SMITH: So the objectives of the ALS includes both deterministic and risk-informed insights. And the objective is to obtain NRC approval of the generic method to address PWR LOCA induced FFRD in an expeditious manner.

The activities for the traditional deterministic approach are ongoing and will take additional research and time. And so this will provide the benefits of higher burn-up sooner. So we want to -- we don't intend to rely upon additional integral LOCA tests. So we're not tying ourselves to the TREAT test program, for example, although there are some cladding tests that are used to support this, particularly to address burn-up effects, but limit the licensing complexity and risk by largely relying upon previously approved methods and strategies.

But of course there are burn-up effects that would need to be incorporated into the currently approved methods. And those would be part of the submittal.

And then minimize the plant-specific, simplify the plant-specific implementation. So this would be a generic topical that would be easy to demonstrate compliance with the boundary conditions for the analysis. And then a plant can then add whatever other licensing basis issues that need to be addressed.

CHAIR BALLINGER: Okay, you've used the magic word, topical.

MR. SMITH: Okay.

CHAIR BALLINGER: What's the schedule for that?

MR. SMITH: Well, what we have communicated and continue to support is either between the end of this year and the end of the first quarter next year.

CHAIR BALLINGER: Thank you.

MR. SMITH: This is confusing me, because this is slow here.

So the basic approach is that the large, intermediate-break LOCA, we're going to show that there's no clad rupture using traditional deterministic methods.

So in the previous conversation we talked about pressurizer surge lines potentially having a different performance. So the pressurizer surge line is inside the deterministic LOCA and can be evaluated just like any other LOCA analysis.

The large-break LOCA will be determined based upon more realistic treatment considering xLPR calculations that you've seen. We're crediting leak before break for piping. It's already qualified for leak before break.

All PWRs in the country that have main loop piping, already analyzed and qualified for leak before break, demonstrate that there's ample time between a precursor event, detectable leakage, and rupture to address large amounts of potential uncertainty and risk, and then crediting the existing tech specs that require you to shut down the plant and therefore reduce the KT. So if there could be a hypothetical LOCA in those conditions, it would have no consequence to fuel fragmentation.

So the rationale is, you know, we have the capacity under those conditions to address large-break LOCA for 5046. We're not proposing to change the treatment of LOCA only with regard to FFRD. So plants will still need to do a full 5046 analysis. But for -- yeah, okay -- but for FFRD we're going to justify that credit for LBB is appropriate as a more realistic treatment.

There are examples here of the history, if you look at the types of events that have been justified, LBB of course, the initial asymmetric loads events. But somewhat more representative of our condition are credit for control rod scram and fuel mechanical loads do credit LBB and are not addressed in large-break LOCA today.

So, I'm sorry, it's a little delay here.

So we apply the xLPR analysis, as we've already discussed that, when we limit it to large pore cooling systems, then the probability of the initial event itself goes essentially almost to zero. It doesn't go all the way to zero, because they stop the computers. You know, they don't even run so many realizations. But it is approaching a very, very small value.

And then the time between detectible leak and rupture is sufficiently large that, considering operator response, uncertainties become extremely low. And in fact, it becomes not credible that somehow you could have operating crews over the 3.8 months not react for the tech specs and shut the plant down.

MEMBER KIRCHNER: Fred, this is Walt Kirchner. Could you just elaborate on the green arrows on the bottom. I think Craig mentioned 14 inches earlier in his presentation. Is there some kind of spectrum there where you've done sensitivity analyses to determine, you know, what that threshold is for inducing FFRD?

MR. SMITH: Yeah. So for the green arrow on the left, that's all intermediate and large-break piping for the various in Triple S configurations. So the actual diameter changes a little bit with in Triple S design, so we took the numbers off. So the deterministic methods will include everything from the pressurizer, surge line, and accumulator line on down. So the full spectrum analysis is being done for all the in Triple S that's being considered.

CHAIR BALLINGER: So I'm looking at those green arrows again, like Walt is. And the one goes to the left says we're not going to do anything, you know, get clad bursts, so we don't have to worry about it.

MR. SMITH: We're going to demonstrate that you don't get clad bursts.

CHAIR BALLINGER: Oh, okay, demonstrate that you don't get clad bursts. If that's true, then you don't have to do anything.

And the one to the right says the LBB

evaluation does not include FFRD in LOCA evaluation and model.

MR. SMITH: Right.

CHAIR BALLINGER: Does that mean you're going to get rid of FFRD as well there?

MR. SMITH: We're going to say that there's no credible scenario for LOCA induced FFRD for large core piping based upon the fundamental probabilities of large core rupturing and the large time between detectible precursors and rupture that gives you more than adequate response to shut the plant down.

The heat loads are very low, and therefore you've lost the motive force for LOCA to occur. And even if you decide to simulate that with a LOCA model, you would get no rupture.

CHAIR BALLINGER: So in this path forward, what's the long pole in the tent in terms of coming to the agency and saying we don't want to do this? We want to implement this. What's the long pole in the tent, which is going to be the hardest to --

MR. SMITH: Well, you brought it up earlier. And so as you mentioned, the Commission's directed the staff to begin rulemaking, include FFRD. We've engaged with the staff on a number of occasions.

But, of course, we're not privy to their deliberations.

There is a policy from 1986, '87, that says that LBB should not be used for ECCS evaluations. And so I think the long pole is to convince the Commission to reconsider that policy.

Now that policy, when you read that they did take an active effort to consider that in '87, they said we're not going to close the book on this. But industry hasn't identified any safety benefits. And so we're going to put this on the table. And so we're identifying safety benefits associated with high burn-up. And I'll talk about those later.

And so that, to me is the challenge to get alignment with the staff and the Commission on crediting LBB.

CHAIR BALLINGER: But one of the criteria that the staff is proposing, where you have to consider FFRD or relocation and fragmentation, is 55,000 megawatt-days per.

MR. SMITH: Yeah.

CHAIR BALLINGER: That's way below high burn-up.

MR. SMITH: Yes. So the cases in the arrow on the left, we'll consider the fuel that's susceptible, and evaluate the LOCA, and then determine that --

CHAIR BALLINGER: If you don't get burst, it doesn't matter.

MR. SMITH: Yeah. We're not worried about burst for fresh yield, but everything above the acceptance criteria.

MEMBER KIRCHNER: And now, Fred, this is Walt again. So to elaborate on Ron's question, obviously you could get clad failure with a small-break LOCA under certain circumstances. But you're basically saying you don't have the differential pressure that's the driving mechanism for FFRD.

So you're not necessarily -- on the left arrow you'll analyze where there is an issue or not, of course, but you're basically saying that on the left arrow you don't have the differential pressure that would result in the driving mechanism for FFRD.

(Simultaneous Speaking.)

MEMBER KIRCHNER: And will the staff accept that?

MR. SMITH: Ha, ha, ha. I don't have my crystal ball with me, so --

MEMBER KIRCHNER: No. But are they indicating technically that's the way they view the issue?

MR. SMITH: So let me just clarify one thing. And you're right, there's a differential pressure component. But, you know, for the ballooning rupture that's associated with FFRD, you need to get to it at least to hike clad terminal pressure, terminal temperature. So it's 600, 700 degrees C. And so the combination, if you get some ballooning, then you'll have relocation potentially. And we'll evaluate that. But if you don't get enough ballooning, then you won't get rupture. And so that's --

MEMBER KIRCHNER: But you won't get beyond three percent strain. To me that's their criteria.

(Simultaneous Speaking.)

MEMBER KIRCHNER: So, Fred, to support this, with the high burn-up fuels it seems to me you're going to haves to demonstrate to the staff that the fission gas release doesn't result in a buildup of excess pressure within the clad.

Given that you've got a fixed elevation for the fuel to fit into the existing plant designs, are you confident with the high burn-up that you're not going to see a high fission gas release, a rapid fission gas release, and a pressure spike from that on the left-hand arrow spectrum of events?

MR. SMITH: We are including, and it is a work in progress, but we are including transient fission gas release effects in the model. And so we expect to be able to demonstrate that those are adequately modeled and those effects are credibly captured. Next slide --

MEMBER KIRCHNER: Are your vendors considering, you know, it's desirable obviously to have helium in the clad fuel for heat conduction reasons. Are they looking at lowering the charge pressure for new fuel?

MR. SMITH: Probably not. You know, the helium also suppresses fission gas release early on. So you wind up with a lower total internal pressure if you have a charge. But you're correct in that the internal pressure may be a challenge against the no clad liftoff criteria. And so there are some changes to the rod configuration that will accompany this to make sure you make those design objectives met.

CHAIR BALLINGER: But the no liftoff criteria doesn't apply in a LOCA.

MR. SMITH: But it applies before the LOCA.

CHAIR BALLINGER: Yeah, yeah. Okay.

MR. SMITH: Yeah. And this is kind of on topic just summarizing, you know. And details of this are proprietary, but this part is not. But it kind of goes to what we were talking about.

So from a LOCA model perspective, we're looking at bounding PWR ECCS models and really one set of parameters with bounding assumptions for in Triple S configuration. And so we've done the fuel management for 18 to 24 month cycles with high burn-up and high enrichment.

And so the nuclear characteristics, we will bound those in a way that we believe can be incorporated in to the reload validation check list and the tech specs in the future. The fuel rod design, we will not apply this to all cladding types and all rod designs. It will be a select subset that will be licensed to meet these criteria.

And then broadly speaking, there will be conservatism included to envelope this in the future, so avoid to coming back to re-look at this again. And so this results in a different ECCS analysis for each in Triple S configuration, two loop, three loop, and four loop.

And we already talked about the burn-up performance, so transient fission gas, re-burst, relocation, and justification for the material performance will be included in the methodology updates.

The cladding rupture is existing, cladding rupture models today. And so those will be extended with measurement as justification to high burn-up conditions. And all this information will be part of the topical as a list of requirements that are required to demonstrate applicability.

So the utility can say, yes, I've got this fuel, yes, I have this ECCS injection rate, you know, yes, I'm willing to operate within these COLA limits, et cetera. So they can easily say, yeah, this applies to me. And therefore the conclusions apply to me.

CHAIR BALLINGER: Okay.

MR. SMITH: So, you know, we talked this morning about the LBB specs. And sort of just as a reminder, so there's a 72-hour LCO, so every three days at least, and I think in practice it's really almost a semi-continuous thing. But at a very minimum, every three days you expect compliance surveillance on undetectable leakage exceed the 1 gpm limit, then you have to be in Mode 5 in 36 hours.

So what's real important to me from this perspective is if you compare this to the approximately 100 days between detectible leakage and burst, then the likelihood of operating staff, multiple operating staffs and multiple conclusions using, you know, highly trained operators in specific procedures, missing this is virtually infinitesimal.

So we're not planning on doing it in the reliability analysis, but if we were it would score out at a probability below the lowest level of human reliability, I mean, the highest level of reliability, but it would be below ten to the minus six. So that could be convoluted with, really, the LOCA initiation frequency. So, this becomes a really incredibly improbable event.

MEMBER MARCH-LEUBA: And remind me, is Mode 5 depressurized? Mode 5, is it depressurized?

MR. SMITH: It's cold shutdown, below 200 degrees Fahrenheit.

MEMBER MARCH-LEUBA: Pressure? What pressure?

MR. SMITH: It depends upon the reactor, but it's slightly above atmosphere.

MEMBER MARCH-LEUBA: Yeah. So, depressurized?

MR. SMITH: Yeah.

MEMBER MARCH-LEUBA: So, what other force you have to break up, continue to break, it won't be there.

MR. SMITH: Yeah, that's right. So, as we said before, the motive force is gone. And of course, if you have an unidentified leak, you're in there to find it and fix it. And so, you're not going to continue to be pressured.

Next slide, please. Yeah, so as I said before, all BWRs have licensed at least the hot and cold leg for LBB, and many have gone to below the 1 gpm. And so this relatively long period using statistically conservative evaluation of the analysis results provides a large amount of temporal margin to any clad piping rupture.

And we already talked about motive force. And then, of course, it's not really credible that you could go from a 200 degree Fahrenheit or 100 degree Centigrade without temperature with the decay heat loads that would be present after 100 days to anywhere close to, you know, 600, 700 degrees C. So you have no clad rupture.

Next slide, please. Yeah. So one point here that, you know, this is essentially the condition that you're in at the very end of long term cooling. And so if you had a LOCA, however you have it, we don't know, but if you did have a LOCA, you know, essentially you're going to drain the top of the vessel. And you're going to have a big mess on your hands.

But you're not going to expect that you're going to uncover fuel in the RHR capacity to make up. Any steam generation is a small fraction of one side of the RHR system. And so you're going to just sit there, kind of like an outage almost. And so any of the consequences of a LOCA, such as the challenge to equipment qualification, environment qualification, radiological release, won't be present in this kind of scenario.

MEMBER MARCH-LEUBA: And the other argument is that seismic -- the design basis earthquake, will not break a good pipe. The pipe has to be already broken, or there is no possibility that an earthquake would cause the one, you know, the break that we always assume.

MR. SMITH: Yeah. It has to be damaged, as we talked about earlier.

MEMBER MARCH-LEUBA: I know.

MR. SMITH: Yeah.

MEMBER MARCH-LEUBA: That's what you have to convince us of the -- and we always have the frame of mind.

MR. SMITH: Yeah.

MEMBER MARCH-LEUBA: Of course there's the LOCA, the earthquake, not any safety and --

MR. SMITH: Yeah. I understood. I was listening very carefully to those questions.

MEMBER MARCH-LEUBA: Yeah. That's an argument that needs to be done very -- with ten to the minus six probability.

MR. SMITH: Okay, yeah. So for defense in depth considerations, this is another key point. So LBB applications don't explicitly consider defense in depth. This is somewhat implicitly, I believe. And they certainly don't, you know -- and this is out of the LBB federal register that the substantial range of pipe crack sizes are stable for an extended period of time.

And so this is what supports the ability to detect leaks. There's not a cliff effect. And the probability of rupture is extremely small. So that's already kind of in the framework when you say LBB.

For defense in depth considerations, the only thing that exists, really, is the very conservative assumptions in the deterministic fracture mechanics. And so when you make those assumptions, you're defending at every line the capacity of the piping system to withstand a rupture. So there's a degree of defense in depth in the original licensing of those.

xLPR takes a very different approach in that it is using a whole range of probability distributions, and sampling those probability distributions, and then we're drawing a NAIFA line on the limit. And so, I would contend that those have an equivalent function. And so many elements of defense in depth are built into the xLPR analysis methods.

(Simultaneous Speaking.)

MEMBER DIMITRIJEVIC: What are the safety margins? You know, you have two things to address, safety margins and defense in depth. And all of those, conservatively, falls in category of the safety margins, you know.

MR. SMITH: Yes, that's right. And, you know, we'll have to address those, but I think the two principal arguments are that you've lost the motive force, and so a credible earthquake, I mean, a credible rupture isn't going to happen.

You have large amounts of time margin. So if we take this 3.8 months of time margin and, you know, cut it in half, cut it in three-quarters, you still wind up with the same assumption. So there’s large amounts of margin in that result that can accommodate many, many potential challenges.

So next slide, please.

MEMBER KIRCHNER: Fred, this is Walt again. Could you just refresh our memory, or at least mine, on what the typical in-service inspection intervals are for the primary coolant boundary?

MR. SMITH: I'll ask my piping experts to answer that. I don’t really know.

MR. BURKHARDT: So, the typical in-service inspection?

MR. SMITH: Yeah, okay.

MR. BURKHARDT: The microphone's right here.

MR. SMITH: Yeah.

(Simultaneous Speaking.)

MR. BURKHARDT: So, this is Markus Burkhardt at DEI. In-service inspection intervals depend based on the specific component and temperature. But they’re typically in the every couple of years to every ten years for things that are hot lag or cold lag temperature. Mitigating components are sometimes, on a sample basis, a little bit less frequent than that. And components at pressurizer temperature are more frequent than that, to your point.

MEMBER KIRCHNER: No, I was just thinking that you can use the ISI as a defense in depth argument to bolster your case.

MR. BURKHARDT: Yeah, that's --

MEMBER KIRCHNER: To look for what I would be looking for is something beyond, you know, a small incipient crack or something, some obvious indication of something, either in the environment that the piping is exposed to or something else, other factors that might lead to a more significant leak problem or probability of a leak.

MR. BURKHARDT: Yeah, thank you for that feedback. I think discussing the conservatisms built into the ISI program is another layer of events.

CHAIR BALLINGER: Division 2, Section 11, Division 2 was written not for LWRs, but is it useful here?

MR. SMITH: I don't know. I would have to look at that. I'm not familiar with it.

CHAIR BALLINGER: Because it allows probabilistic identification of things.

MR. SMITH: Yeah.

MR. HARRINGTON: This is Craig. It may

be useful, but I don't know that anybody's going to apply it.

MR. SMITH: Okay, well --

MR. HARRINGTON: You know, it could be, but everybody, all the existing fleet operate under Division 1, well established ISI programs. They do monitor it for those things just described. So it's really their intent.

MR. SMITH: Okay. So I've talked about this a little bit, but this is a cartoon just to illustrate the, you know, response from a plant operation staff.

And so, again, we talked about the large time margin between consequences. And the shutdown of the plant is being performed by highly qualified, trained operating staff, proceduralized processes. In many cases it's automated, and they're independently reviewed. And so the human reliability opportunities are relatively low.

And then the cartoon on the right shows we have a diverse set of indicators. And so we're not relying upon one single parameter to detect slow leakage. And so we have multiple means of getting feedback. We may have something that's not understood or anticipated.

Next slide, please.

CHAIR BALLINGER: So this is a lot more than addressing FFRD. This is a huge change in the way you address LOCA.

MR. SMITH: Yeah, so --

CHAIR BALLINGER: The FFRD is kind going along for the ride here, I think. Ha, ha, ha.

MR. SMITH: Well --

CHAIR BALLINGER: I mean, this is --

MR. SMITH: So my friend Al Santos is here. And so there's a potential, certainly a potential interest in that kind of change. The scope of our topical report is limited just to FFRD and so -- but yes, it does kind of set up a framework that maybe there could be a change to fully risk inform large-break LOCA in the future.

MEMBER KIRCHNER: So, Fred, this is Walt again. Ron asked a question earlier in one of the presentations about the long pole in the tent. Does rod ejection then become the long pole in your tent in going to LEU+ and high burn-up?

MR. SMITH: Well, it's something that needs to be addressed, you know, the full spectrum of accidents, fuel handling. Fuel handling may be surprisingly fuel assembly drop, it's also called, can be surprisingly difficult, or ejection as well.

One of the big differences was rod ejection, as it's a very local event in that it's going to only cause failure in a relatively small part of the core. And so it's different from that perspective.

MEMBER REMPE: The approach is different somewhat, but is it so different than what was done with GSI-191 for a lot of plants, to investigate for why they didn't have to -- they could address it?

MR. SMITH: That's right.

MEMBER REMPE: And so I don't, you know, I applaud your work, but I don't think it's just -- this is so different that we haven't --

MR. SMITH: Well, remember that --

MEMBER REMPE: -- the application was successful.

MR. SMITH: Similarly, the transition break size rulemaking almost went through, but they failed to identify enough benefits.

MEMBER REMPE: Yeah.

MR. SANTOS: Can I add onto that?

MR. SMITH: Sure.

MR. SANTOS: Fred just said -- I heard what you said, Ron, and I heard some other comments as well. Oh, this is Al Santos from NEI. And, you know, what Fred is doing here with the ALS approach is really, almost you want to call it an offshoot of the 5046 Alpha rulemaking that was initiated back in 2010, and it was discontinued in 2016 from the staff, where it was a risk informed LOCA activity.

They used it when -- Fred was talking about the transition break size which was looking at this criteria where you could change some of the design criteria based upon -- or minimize, you know, some of the issues, and I won't go into that, but going into the smaller break sizes up to the transition break size. That was, you know, calculated through a reg guide that was proposed in that rulemaking.

What Fred is doing here is really updating that type of approach to the modern fracture mechanics, probabilistic fracture mechanics tools that we have here, and focusing it to a specific application.

So I think that, like you said, Ron, this has got other places that it could be very useful. But for the short term and what the utilities are looking for, this is a specific, targeted application to see how we could use this risk informed LOCA approach going forward.

(Simultaneous Speaking.)

MR. SANTOS: Yeah, go ahead.

MR. SMITH: So one of the areas that we haven't talked about is that, you know, we're doing LOCA analysis for piping systems and addressing large-break, but there are other potential failures that could result in a LOCA. And so we're going to be reviewing all those.

And, of course, the easiest ones are pipes that will result in a LOCA that's smaller than the deterministic analysis that we're doing. And there are some larger systems like a steam generator, manways that have been historically addressed based on some fracture mechanics and some measurements.

And we're reviewing those and going to provide justification for why the probability of those rupturing and causing a LOCA is acceptably small. And then there's also the potential that you could have active component failures that could result in a loss of coolant.

MEMBER HALNON: But how do you reconcile the Davis-Besse event with what you just said?

MR. SMITH: Well, that's something that, you know, certainly the Davis-Besse event was extremely unfortunate and --

MEMBER HALNON: In many ways.

MR. SMITH: Yes, in many ways. And it is, at the very least, as a former utility person, it violates almost all the standards that we were upholding.

MEMBER HALNON: So that's a one-off --

MR. SMITH: Well --

MEMBER HALNON: -- isolated point somewhere that we --

MR. SMITH: We can certainly hope so.

MEMBER HALNON: But notwithstanding all the causes of the event itself, it did show that human error, I guess, is the best way to put it.

MR. SMITH: Multiple cultural breakdowns.

MEMBER HALNON: Yeah. It could cause the problems here. I mean, it was demonstrated that it did. It wasn't theoretical.

MR. SMITH: No, I understand. And I'm trying hard not to try to justify it.

(Simultaneous Speaking.)

MEMBER HALNON: -- the employees. So you can't offend me.

(Simultaneous Speaking.)

MR. SMITH: Yeah.

MEMBER HALNON: I wasn't there during the event. But I was the first energy survey.

MR. SMITH: Yeah.

MEMBER HALNON: So, don't worry about hurting my feelings.

MR. SMITH: No, I wasn't worried about that. But the --

MEMBER HALNON: I brought it up.

MR. SMITH: -- the methods of detection that we showed on the other slide is what --

MEMBER HALNON: Okay.

(Simultaneous Speaking.)

MR. SMITH: -- allowed them to detect --

MEMBER HALNON: Right, so --

MR. SMITH: -- the event.

MEMBER HALNON: The leakage was much less than one gallon per minute.

MR. SMITH: Well, there's that, but there was radiological evidence.

MEMBER HALNON: There was a lot of look-back and --

MR. SMITH: Yeah.

MEMBER HALNON: Yeah, I understand that. And so basically that event set up the preconditions that you're talking about here, basically.

MR. SMITH: Right.

MEMBER HALNON: We strengthened our pre-screening. And if we --

MR. SMITH: If I thought this was going to be a recurring event, I think there would be a lot of other things to talk about.

MEMBER HALNON: We'll go with that, yeah.

MR. SMITH: So I would just highlight that if the control rod drives had been injected that would be much less than the LOCA event that we are analyzing. And so that would still be covered.

MEMBER HALNON: Okay.

MR. SMITH: Next slide, please. So just as background, we have been talking with the staff on this in a number of forms. So we wrote an initial draft approach on this. And we have since refined that. So it's there, you can look it up. But it's not really what we're doing today. So that's just included for completeness.

We've had two xLPR public meetings the with staff, most recent one in January. The ALS, we presented ALS to the NRC several times including last August at their high burn-up workshop. And then right after that we had a pre-submittal meeting, and had some good, hard questions but some general, reasonable feedback at the results encouraging us to be persistent but be prepared. And so that's what we've been working on doing.

And we meet with them every quarter to update. We were very anxious to hear what their first phase of the rulemaking provides. And so we expect to respond to that as appropriate.

Next slide. So safety benefits, so high burn-up does produce a lot of potential safety benefits. So high burn-up would, in broad terms, just as a rule of thumb, we expect to reduce the reload requirements by 20 percent and, therefore, the back-end requirements by about the same amount. Of course, that varies by plant design, but that's just kind of a rule of thumb.

And so the risk of transportation across the entire fuel cycle is reduced. The risk of fuel handling in the plant due to reload, smaller vat size, is reduced. High level waste, this is a very substantial benefit to me, but the amount of high level waste that you have to store at the site, load into the dry cast, eventually transport it to a repository --

MEMBER MARCH-LEUBA: Are you 100 percent sure about that? You know, high level waste is the outcome of the power times time. That's how many isotopes you produce. You just concentrated more in high burn-up fuel. It's more dense. So in a sense, I want to ask you, when you come in, are you hitting your dry cask, because you're packing more use into it.

MR. SMITH: Well, certainly the atoms are somewhat similar, but you're also burning some, and you've got fewer packages.

MEMBER MARCH-LEUBA: Again, it's more concentrated.

MR. SMITH: That's right. So if you have an accident, you have --

(Simultaneous Speaking.)

MEMBER MARCH-LEUBA: Is it concentrated enough to cause some -- well, I know you have some heat load in your dye cask. You put high burn-up fuel, the heat load will be higher.

MR. SMITH: Well --

MEMBER MARCH-LEUBA: It's not going to be lower.

MR. SMITH: Not necessarily. So, yeah, we'll talk about that later, but the heat load is --

MEMBER MARCH-LEUBA: Somebody is out there --

MR. SMITH: Well, that is one of the topics that's in one of my presentations a little bit later. So we can --

MEMBER MARCH-LEUBA: We'll defer to you.

MR. SMITH: We can table that until we get to that.

MEMBER MARCH-LEUBA: Sure.

MR. SMITH: Okay. So where was I, here. So economic performance sites that are at risk of early shutdown because of their economic performance, this provides some tangible economic benefits that would allow them to continue to operate and continue to support U.S. and international environmental goals and Green House emissions.

Core design efficiency reduces the uranium requirements and the effect on the environmental and radiological impact for the whole fuel cycle. And this has been proven for all of the previous burn-up upgrades. Staff is in the process of doing the environmental evaluation for this. And there's no reason to expect they wouldn't have the same conclusion.

Longer high burn-up fuel enables longer fuel cycles, so there's fewer outages, lower outage risk, and less personnel dose, because they're not in an outage which is a major source of personnel dose.

And then last but not least, because we are precluding burst, there are any number of phenomenon that we would not have to develop models, and do research for, and tie up industry, personnel in the NRC, personnel in designing, I mean, evaluating those models and phenomenon. So that contributes to the overall effective use of scarce resources.

So this is kind of what the submittal looks like in cartoon form. So we will have a topical report in blue. It's on the left side. And supporting that would be an xLPR analysis that you've heard about today as well as vendor-specific LOCA application reports.

The methodology, the vendors will hold that as proprietary, and so they will submit that separately in coordination with this report. And then, as I said before, there will be a application section that will allow the utility a clear path to adopt this.

And so in summary, we're going to do deterministic, small, intermediate-break LOCA, apply LBB to the large-break LOCA, and then address non-piping ruptures. And the plan is to submit this before the end of this year, between the end of this year and the end of the first quarter of next year.

MEMBER HALNON: Fred, could you just summarize? After you said the utility can adopt this, what do they -- summarize what the benefits to them will be.

MR. SMITH: To doing this?

MEMBER HALNON: Yeah.

MR. SMITH: Well --

MEMBER HALNON: I mean, it's a big deal. You have to put the license -- I mean, it costs money, and a lot of time, and stuff.

MR. SMITH: So when you compare it to a fully deterministic approach, there is fair amount of research that's ongoing, the tree test. I know Joy can tell you about the governor of Idaho allowing the lease of environment rods into the state of Idaho. It's been, I don't know, ten years maybe. So those tests will begin sometime next year.

There are other phenomenon tests that we're working on. And so I don't have a crystal ball on how long that backup would take. But it wouldn't be unreasonable to say that it would be another five years.

MEMBER HALNON: Well, but practically, are they going to be able to expand their PT curves as they're --

MR. SMITH: Well --

MEMBER HALNON: -- what physical benefits to an operating control room will this provide?

MR. SMITH: So obviously when you improve the -- reduce the vat size and improve the efficiency, you're going to reduce the leakage. And so that gives you the opportunity to either stretch out PT curves in time or to revise them in a more favorable way so you'd have less neutron leakage.

MEMBER HALNON: So you're -- license renewal will be easier to get to, you know, even beyond the --

MR. SMITH: Those are all potential --

MEMBER HALNON: But we're setting ourselves up, yeah, longer running, lower leakage plant. But the control room operator really --

MR. SMITH: We would like him to not know that --

MEMBER HALNON: Not have to worry. That helps, thanks.

MR. SMITH: Okay.

CHAIR BALLINGER: So probably a dumb question, I'm pretty good at them. Who owns xLPR?

MR. SMITH: It's jointly owned by the NRC and EPRI, I believe.

CHAIR BALLINGER: That's what I thought.

MR. SMITH: Yeah.

CHAIR BALLINGER: So now you have the industry using xLPR. And the staff is going to use the same xLPR, the same code to check the industry's use of xLPR?

MR. SMITH: Well, perhaps.

CHAIR BALLINGER: Well, there's no other code that's going to do that.

MR. SMITH: That's true. Craig, if you want to --

MR. HARRINGTON: This is Craig. That was actually one of the intentions in jointly developing the code, is then we're in a position where we're arguing over the details of the application of the code as opposed to arguing over what's in the code. And a probabilistic code is such a big black box, you could spend all your time arguing over what's inside the box. And, you know, both groups very aggressively worked to make sure that the things in the box were to the best of our ability to represent reality.

MEMBER MARCH-LEUBA: Yeah, that's the definition of conflict of interest.

MEMBER REMPE: That's not the, actually,

that's not the only application. Your SCALE code from Oak Ridge is a good example. And we've had a design certification where both the applicant and the NRC used MELCOR. And then they argued about the input and how they, you know, did the novelizations, like --

MEMBER MARCH-LEUBA: The point is --

MEMBER REMPE: -- I can tell you, I know that it's not the first time.

MR. HARRINGTON: -- is the same way.

MEMBER MARCH-LEUBA: Yeah, there's --

(Simultaneous Speaking.)

MEMBER MARCH-LEUBA: -- that has been done in other applications doesn't mean that you are in love with your code. And IS and NRC review it, I mean, all with the same code. And we're looking with bankers' eye shields. So during the review, they have to be careful about that bias. The engineers fall in love with their methodology.

MR. SMITH: Yeah. But there are many other, I mean, the first term change in Reg Guide 1.183, it's owned -- tallied by the NRC. There's very little capacity in the industry to do those kind of calculations. So that's the opposite.

PARTICIPANT: After the stupid question.

MEMBER KIRCHNER: Fred, this is Walt

again. Could you address your middle bullet there? By component bodies, I'm assuming you're including valves.

MR. SMITH: Yes.

MEMBER KIRCHNER: Yeah.

(Simultaneous Speaking.)

MEMBER KIRCHNER: Is this really a profitable area for -- I could see manways which could be quite large. Valves, by and large, may be bracketed by your deterministic small-break analyses. I'm just thinking that, you know, valves, say you had undetected, very small leakage that led to corrosion. You could have a valve bonnet just blow off. But trying to develop a database that would justify a probabilistic approach to that strikes me as tenuous.

MR. SMITH: Well, many of these things have been already addressed in life-extension applications and have been addressed back in the dark ages when I was a young engineer. And so we're really discovering the basis for those. And I don't expect to do fracture mechanics on valves or research coolant pumps either. But I'm just --

MEMBER KIRCHNER: Well, like you, I'm an ancient mariner. And I've been on ships where valves had blown right off the boilers, so admittedly not nuclear but, yeah. I just wonder whether -- I get it for the primary coolant boundary and piping. It may prove, just may prove difficult to develop a good enough database to justify its application to all these different components. But I'm willing to stand corrected.

MR. SMITH: Well, I think, at least our plan is that we should be able to disposition all but a few because of the largest deterministic LOCA is conservative. And if the valve blows off, then the choke flow becomes the piping going into the valve.

MEMBER KIRCHNER: Exactly --

(Simultaneous Speaking.)

MR. SMITH: And the piping is smaller than the analyzed piping systems, and that dispositions that.

MEMBER KIRCHNER: I was just musing out loud with you, Fred. Because I'm thinking it's very elegant what you presented for the primary coolant piping. But I'm just wondering whether it would be difficult to apply to individual components.

MR. SMITH: Yeah. And I appreciate that. We are working on that even as I speak. So we don't have all those questions resolved. But we are committed to provide a basis for why those components don't pose an undue risk.

CHAIR BALLINGER: Okay. So now we are on Number 7 which is the -- is that the way we're going, to the atomistic modeling?

MR. WELLS: Yeah, we were planning to go to atomistic modeling, that's seven on the published agenda, or eight on the published agenda. Yes.

So, Erich, are you still on the line, hopefully? We're pulling up your slides.

(Simultaneous Speaking.)

MR. WIMMER: Yes, I'm here.

MR. WELLS: He is, he's still here.

MR. WIMMER: Yes. And how is the audio? Is it okay?

MR. WELLS: Yeah, we can hear you.

MR. WIMMER: Okay. Welcome, then. Well, I really appreciate the opportunity here to present atomistic modeling here of the cladding coating behavior. And this is work done for and together with EPRI. And I wanted to mention right in the beginning here my colleague Mikael Christensen who did the really heavy lifting here.

Okay. So on the next slide, there is an outline of the present (Audio interference.)

CHAIR BALLINGER: Uh-oh. Now we're

running into trouble.

MR. WELLS: Erich, your audio is breaking up. Maybe try just turning off your video, see if that helps your bandwidth.

MR. WIMMER: Okay. I turned off the video. Is this better now?

MR. WELLS: It improved a little bit. We'll have to see.

MR. WIMMER: I'll just double check. Let me just double check one thing here in the microphone.

(Simultaneous Speaking.)

MR. WIMMER: Can you hear me okay now?

MR. WELLS: It seems to be better. We'll have to see.

MR. WIMMER: Okay. So hopefully that will work.

So I will review first the motivation, the objectives of this modeling work, and then give the key results up front, say a few words about the modeling approaches but then -- and show the results.

So one of the key questions we wanted to ask is chromium coating to zirconium. Is chromium a barrier or a window for hydrogen? And what is the bonding between the chromium coating here and the zirconium substrate? And what happens if you have defects, in particular, through coating defects. And how does the system respond to mechanical deformations? And then summarize and provide some engineering implications.

Next slide. It's kind of slow. Can you give the next slide?

MR. WELLS: It did not advance on this side.

MR. WIMMER: Hum.

MR. WELLS: We're on key results in the meeting. Are you not seeing that?

MR. WIMMER: I am not seeing that yet. Okay, so we have a big delay here. Hum.

MEMBER REMPE: This is Joy Rempe.

(Simultaneous Speaking.)

MR. BLEY: -- and it's perfect for me.

MEMBER REMPE: So this is Joy. We often have problems this way. And if you'll, say, go to Slide 98 out of 224, the folks in the room will do it. And by the time it gets back to you on the Internet, it'll be a lot longer. So just know the folks in the room will take care of it. Okay?

MR. WIMMER: All right, good. So if you see the key results, well, I know what they are, so it's basically good news. So the chromium of metal, as well as a thin layer of chromium oxide to scale, is a barrier for hydrogen ingress. So that's good news.

But then also we see from the simulations that the bonding between zirconium and the chromium is very strong. So this chromium coating resists even strong and large strains and mechanical deformations.

And so finally, if you have a through coating defects then we were concerned about that the corrosion will then set in at the boundary between chromium and zirconium and thereby lead to delamination. And this is not the case. So those are really the key results that we can conclude from these simulations.

Okay. So then a few words, and let's go to the next slide, Slide Number --

(Pause.)

MR. WIMMER: -- and do you see actually the Slide Number 5?

MR. WELLS: Yes, we're good.

MR. WIMMER: Okay, good. So the methodologies that are being used are atomistic simulations on two levels. One is the first principal is quantum mechanics. And the power of this approach is that there are no system-specific parameters.

That means you have a very high predictive power from thousands and thousands of compilations that have been done in the community with these methods. We have a good sense of the error bars. So we predict thermodynamic mechanical properties and also interface energies.

Then we perform molecular dynamics as simulations, and that gives us diffusion, plastic deformation, and behavior of larger defects at these locations, voids and cracks. And to do this end, we are using state of the art so called interatomic potentials derived using machine learning techniques.

So I won't go into details, but it's truly the state of the art, giving very high fidelity in the molecular simulations. And the software that we using here is for the quantum mechanical calculations, VASPA. And for the dynamic simulation, Landspin, developed at the Sandia National Lab. And that's all embedded in a molecular modeling environment that our company produces and supports they call Media.

Okay. Let's then move to Page Number 6. And that shows the following computational experiment, if hydrogen atom on the surface, what energy does it take for this hydrogen to get into the bulk chromium and then to diffuse into the chromium?

The results show very clearly hydrogen

does not like to go into chromium metal. So chromium metal is a barrier. If, on the other hand, hydrogen would be inside the chromium, it would diffuse relatively fast.

So the barriers for diffusion are low. But it just doesn’t want to go in. So in other words, the solubility of hydrogen and chromium is very, very low. So that’s why chromium metal is a barrier.

Okay. So on the next slide, now on Page Number 7, we see the same question being asked. Well, what happens to the chromium oxide? Because we know that if you expose metallic chromium to an environment it typically forms a very thin but very nicely protective chromium oxide scale. So how does hydrogen behave in that?

And the answer here is actually somewhat different. It can go in with a modest barrier. But then inside it has great difficulties to diffuse. So it really gets blocked and trapped inside. So yet for a different reason, chromia is also a barrier. So both chromium metal and chromia are barriers for hydrogen ingress. Very good news.

Moving on to Page Number 8, so how is the bonding between chromium coating and zirconium? And again, that’s a computational experiment. We simply place chromium onto the zirconium metal from an interface and then pull it apart.

And what actually happens is that the interface does not break between the top zirconium layer and top chromium layer, but it breaks inside the zirconium such that, at the end of this separation, you have one monolayer of zirconium atoms attached to the chromium coating.

So it can quantify that the work of separation of the chromium coating from zirconium is 2.62 joule per square meters. And for comparison, if you would take bulk zirconium and just cleve it, that will cost you 3.2 to the square meter. So in other words, the bonding between chromium and zirconium is really quite strong.

Okay. Now moving on Page Number 9, hopefully we'll see that.

CHAIR BALLINGER: This is Ron Ballinger. You're talking about a basal plane --

MR. WIMMER: That is correct, yeah.

CHAIR BALLINGER: -- in the zirconium. But we know that the cladding primarily has about a plus or minus 30 or 40 degree texture. And not only is it a 30 or 40 degree texture, but the pilgering or the process of making the tubing results in sort of a helical pattern on the cladding. So you're not really talking about a basal texture that's sticking out.

MR. WIMMER: Yes. And that's a very good question. In fact, when we carried out the project later Rob Baum exactly mentioned that. So then you will see later on simulation that where the basal pole is tilted by 30 degrees, and then we looked at what happens.

CHAIR BALLINGER: Thanks.

MR. WIMMER: But basically this chemical bonding between zirconium and chromium is, to some extent, independent of the crystalline orientation. So it's very local but, of course, it's a very good point. And yes, we did address exactly this issue. Because the surface of the cladding is not to be the basal plane itself, but it's tilted to a granular, a grain structure, of course. Yeah. But, I mean, those are models that give you a sense of what is the bonding here. And, of course, you make in this kind of model some simplifications.

So now we ask the question what about the chromia scale being attached to the chromium surface? How strong is that bonding. And again, we do the same computational experiment. You know, we create the interface, again, using the basal plane, that's true, to pull it apart.

And again, the bonding is very strong. So in other words, the chromia scale sticks very, very tightly to the chromium metal. And that is also, of course, good news.

So moving on to the question of, kind of, the response to mechanical deformations, and hopefully that we'll see this soon.

Okay, no, first the, sorry, first the issue about what happens if you have a through coating defect? Then, of course, the coolant gets access to both the chromium but also the underlying zirconium.

And what that leads to a situation where the interface between chromium, zirconium starts attracting oxygen and hydrogen, which is the product of the disassociation of water, and thereby destabilized and ultimately lead to delamination.

Or would the oxygen and hydrogen that's being produced by the dissociation of water actually defuse into zirconium? That will be Scenario B. And the calculations very clearly show its Scenario B that actually takes place. And it's much more likely.

So there is no preference really for oxygen, hydrogen to accumulate in the zirconium, chromium interface or perhaps precipitate oxides or zirconium hydrides which will be detrimental.

So again, through coating defects probably remain kind of localized and don't lead to a spread out of this defect which, again, is good news.

MEMBER KIRCHNER: Erich, this is Walt Kirchner.

MR. WIMMER: Yes?

MEMBER KIRCHNER: Did you look at this phenomenon under different thermal conditions?

MR. WIMMER: That's what --

MEMBER KIRCHNER: In other words, with the zircaloy, the chrome zircaloy at temperatures that you would normally see in a PWR when you have sub-cooling nuclear at boiling.

MR. WIMMER: Yes. Now what happens is that, of course, the diffusion rate changes, and it depends strongly on temperature. And we do have explicit expression for temperature dependent diffusion coefficients of oxygen and hydrogen in zirconium. Also the reaction rate itself is temperature dependent.

But the mechanism itself of oxidation at the boundary and diffusion into the bulk is, well, the timescale, the rate changes, but the overall mechanism remains the same. So while we did not explicitly do a temperature dependent simulation, under operating conditions you will probably see exactly the same behavior. But it’s a good point, yes.

Does this answer your question?

MEMBER KIRCHNER: Yes, thank you. Yeah, I was thinking that the interface may be a little bit more prone to attack under, you know, operating conditions versus just, you know, after it’s been manufactured.

MR. WIMMER: Yes. I mean, it’s of course the interface. When it’s manufactured, A, as we say it, zirconium, it has a texture, it has a grain structure that are tilted, and you have grain boundaries. It’s much more complex. But the overall mechanism, I think, that’s indicated here, and resulting from the simulation, is credible.

And, of course, with increasing temperature things go faster in an exponential way so, of course, the speed of the reaction changes but not fundamentally the mechanism.

MEMBER KIRCHNER: Thank you.

MR. WIMMER: Okay, great. Well, let’s move on to Page 11. Now it gets exciting, because hopefully you can see some movies. What we’re doing here is we are now deforming molecular dynamic simulation at the 600 K.

So now the temperature is explicitly taken into account. And so that's approximately normal operating conditions. And now we apply a strain perpendicular to the interface, and you see how it breaks. Well, again, it's the same situation that after the breakage, one layer of zirconium remains stuck to the chromium overlayer.

Okay. So now let's move on to a case where the strain is not perpendicular, so you have to just the delaminate it by pulling it apart, but rather lateral, so parallel. And that could be in the case of ballooning or simply thermal expansion of the substrate.

And now what you see is that, of course, you have first an elastic domain. Then you get a plastic deformation, and you activate slip planes. But you see very nicely that the chromium just really hangs on to the zirconium surface. Eventually, of course, it breaks, opens up and exposes the zirconium substrate. So again, even under very extreme strain, the chromium coating remains basically intact until, of course, you expose it.

So then, yes, if we now have, for example, a grain boundary or small crack in the chromium coating, and we deform at the same simulation, maybe you can start the dissimulation. But I'm glad it worked. That is beautiful.

Do you see it okay, in the room?

MEMBER REMPE: Yes.

Mr. Bley: Yep.

MR. WIMMER: Okay. So now, of course, the zirconium substrate is exposed right from the beginning. Then that's a weak spot. And you see the formation of kind of a pit in the zirconium.

Now that's not so good news, because that may simply lead to some greater exposure of the zirconium substrate if you had these kind of defects in the chromium coating. But again, those strains are enormous, and you will probably see them only when you have extreme ballooning here or other effects like that.

All right. Moving on, if in the manufacturing you would have voids between the zirconium substrate and the coating, and those are just the beginning of the end of several snapshots, so not from animation.

But what you see is that if you had voids in the interface from manufacturing, they could grow to fairly substantial voids under strain. But then again, the remaining part of the chromium coating really sticks up there to the zirconium substrate. And under this 30 percent strain, which is enormous, you get these kind of enhanced voids and perhaps this kind of pitting. But overall, this defect remains essentially local.

Okay. On Page 15 we address exactly what you said earlier, namely Rob Baum, who sadly left us not too long ago, but he pointed out, he said what happens if you actually include a tilt in the substrate?

And fundamentally, I guess, some subtle differences but overall the picture remains the same, that under strain the chromium coating adheres to the zirconium substrate. And eventually you expose a piece of zirconium, as we have seen in the more idealistic case of just the basal plane being coated. So we investigated that question, but it fundamentally does not change the conclusions.

And then on the next page we looked at the possibility that, of course, we know that zirconium and chromium does form an intermetallic laves phase which is more brittle. It's no longer cubic. And so what happens if you have such an intermetallic phase at the boundary?

And again the simulations indicate that you have on the top row the case with the zirconium, chromium intermetallic precipitate and the bottom row without. And you see that fundamentally it remains the same except that the system starts to break also inside of the zirconium, chromium to intermetallic which is more brittle. And this is kind of what you expect. But the presence of zirconium, chromium to intermetallic -- if it ever occurs, would not be a disaster.

Okay. Now on Page 17 we can now draw some conclusions. And we did many more simulations also of the influence of niobium and other elements. But overall from these simulations we can conclude that chromium coatings on the cladding of PWRs, as we know, it works under non-oxidizing conditions.

Actually, these simulations did not reveal any kind of red flags. Furthermore, the chromium itself oxidizes to a less extent and less rapidly than zirconium metal. So you get less production of oxide, less production of hydrogen to begin with.

Then both the thin chromia scale and the chromium metal coating are barriers for hydrogen ingress. Chromium coating adheres very strongly to the zirconium surface, through coating defects remain localized, and oxygen, hydrogen atoms created from the dissociation of water in such a crack, and the through coating defect, will remain local and not lead to deterioration of the interface between chromium and zirconium.

And the presence of oxygen on the zirconium substrate, and I didn't show this explicitly, but the simulations show also that they wouldn't be a big problem and, similarly, if you have niobium dissolve, for example, in niobium containing alloys.

And the second set of summary engineering applications are on Page 18. We talked about the presence of precipitate, the tilting of the basal plane, and the void formation. And so it might actually be beneficial if initially the chromium coating is under compressive stress. Because then when the system expands, and is subject to a tensile strain, it may even resist better. So that's an opportunity.

Now there are still some concerns. And that's summarized on Page 19. The assimilations, and I didn't have time to show that explicitly, but if you have zirconium hydrides near the interface, they tend to weaken the chromium zirconium interface.

And given the fact that hydrides tend to precipitate in the colder parts, that means closer to the coolant, that could be an issue. The intermetallics, like these laves phases, may increase the brittleness of the interface. And pre-existing voids may lead to this kind of pitting in the exposed larger areas.

Then another concern, clearly, is under LOCA conditions, if the temperature rises above the eutectic temperature of the chromium, zirconium system, well, you basically melt away the chromium, zirconium. And that eutectic temperature is much lower than the melting temperature of pure zirconium

And, of course, a big issue and opportunity for modeling, I would say, is what happens under irradiation, so neutron core irradiation but also the gamma irradiation. But those are really the remaining concerns, and I think they're on Page 20. It provides the acknowledgment, as always in its projects, that it's a team effort.

First of all, I want thank EPRI for the continued support of this modeling work. I want to thank all my colleagues at materials design, especially Clive Freeman, Clint Gallop, Ben Rominisny, and Walter Walsh.

And so with that, I hope my audio was okay. And, of course, we'll be happy to answer any kind of questions. So thank you.

CHAIR BALLINGER: Your audio came in great after we fixed it. Questions from members?

MR. WELLS: Okay. So we're going to change it up on you again a little bit. And we're going to let Suresh go into the fuel fragmentation threshold now. That way, we'll wrap up all of those related topics before we shift --

CHAIR BALLINGER: That's probably a --

(Simultaneous Speaking.)

CHAIR BALLINGER: -- better path.

MR. WELLS: I think we can make it work.

So, Suresh, we have your slides pulled up in the room if you want to start when you're ready.

MR. YAGNIK: Okay, thank you again. Good afternoon, everybody. Can you hear me okay?

MR. WELLS: You're good.

MR. YAGNIK: Good, so thank you. I'm glad to be here, virtually, though.

Please, next slide. So the outline of my talk is going to be, I'm going to introduce a best estimate fuel fragmentation threshold that we have published, then talk about some scoping experimental tests that were done some time ago, and then move on to some mechanistic studies that we performed on especially the irradiation that was done under a bilateral EPRI program called IFA-649, these were special fuel lists that I will talk about a little more, and then finally wrap up my discussion with a summary and future outlook.

Next slide, please. So the best estimate fuel fragmentation threshold is based on, you know, numerous separate effects tests that, again, I'm going to talk about.

And the Halden IFA-650 series, there were as many as about 16 integral tests in Halden reactor, as many of you are, I'm sure, aware, and then the SCIP-III program which we collaborated to get some detailed information, as well as the open literature information that was available to us.

And then we evaluated all those taken together, you know, although there were integral tests as well as separate effect tests. But interestingly, they all sort of fell into what we call a best estimate threshold which is plotted down below here.

It's essentially quite intuitive if you think about it, that if you have -- it's plotting local temperature versus local burn-up. And if you have a significantly high enough burn-up and high enough local temperature, you will get fragmentation. So that’s the upper right-hand corner of the L shaped curve which is the fragmentation threshold.

And anything to the left and under, which is sort of high temperature but low burn-up and very low temperature at any burn-up, will not fragment. So that’s essentially what we published back in 2015.

And again, as I said, it’s based on local burn-up, local temperatures. And therefore it can be rather easily implemented in any fuel performance code looking one fuel rod at a time. And we are aware that it has been used in industry codes, especially the DOE’s work on BISON code.

And if you know the threshold, and you apply it to a single fuel rod, one at a time throughout the core, you can help assess the mass that is subject to fragmentation. And this is the key word here. We don’t get into relocation and dispersal, but just the amount that can be potentially fragmented. And then therefore it is available for relocation and dispersal.

Next slide, please. So then I move on to the scoping test. And here’s quickly the schematics of the experimental test. We carried out two different programs. One was based on an induction heated furnace with very small samples to the left here, very fast temperature rims were intentionally applied on a very small sample, about a million-liter cube in dimension.

And then the fission gas release and fragmentation was monitored, and the effect of hydrostatic restraint in the system could be also monitored. So if we apply very high hydrostatic pressure in the system, the fragmentation behavior is interestingly quite impacted as I will show you in my next slide.

To the right here is a laser heating system. So as we reach the -- and these tests were done both on LWR irradiated fuel, so we're talking about irradiated samples at different burn-ups. And we took the samples, and it was heated to a base temperature close to the addition temperature and then pulsed with laser beams on the sample itself, again very small piece of sample here.

And then it monitored how the fragmentation occurred through the optical window as well as then, after the test, the optical metallography or ceramography on the sample was performed to see how the fragmentation progressed. So these are what I call the defragment scoping tests on fragmentation done several years ago.

Next slide, please. Then based on this hydrostatic pressure that I mentioned, it became clear from the scoping test that in a fuel rod, when you have a strong metallic clad mechanical interaction, which is a rather complex state of stress, but we simulated experimentally already hydrostatic pressures.

And you can see on this block here that, as the pressure increased, the fission gas released, which is sort of a surrogate for the fragmentation behavior, especially in a small miniature sample reduced quite a bit.

So one good news was that once you have a high burn-up and therefore a high pellet clad mechanical interaction, your fragmentation naturally gets reduced. This was further confirmed in IFA-659, a fuel test that we later perform again in France. This particular data came out of Japan.

Next slide, please. So here, a little bit of background of especially radiation that I mentioned of IFA-659, U02 based fuel disc. Because of the special disc irradiation, rather than a fuel rod irradiation in the Halden reactor, we were able to get almost homogeneous burn-up and irradiation temperature across the fuel disc.

And each fuel disc was then discharged with a scram of a test reactor to capture at temperature, collect in terms of microstructure and sort of capturing the distribution of the fission gas bubble at the temperature that it experienced in the reactor at different levels of burn-up, which included either no transition to high burn-up structure nor in region, so to speak, a partial transformation to high burn-up structure, or full transformation to high burn-up structure. And I'll elaborate on that a little later.

And this kind of unique sampling which is then very conducive to do testing in hot cell is not retrievable in LWR fuel rod, from LWR fuel rod. Because naturally LWR fuel rods support very steep temperature gradiation, temperature gradients, but not gradients, I must say. So that's the reason, so to speak, of this special irradiation in IFA-649, the disc irradiation.

And then after these samples were unloaded with the SCRAM, they were then subjected to different hot cell lab as a function of burn-up, ramp rate, and external restraint and, at the same time, monitored online the krypton-85 release experimentally. This was very extensively done together with pre and post tests, a characterization of each sample that was pursued.

Next slide, please. So here very quickly, the schematics of the two systems that are in the Laka lab in France, Cadarache, the one to the left is prostrated pressure, what they call Mirark system which, again, is monitoring the krypton-85 release from a sample online.

And to the right here is essentially the same thing, online monitoring of krypton-85. But in a hot cell system which could go up to about 160 megapascal, and 1,600 degrees C maximum temperature.

This, of course, required a lot of safety validation from hot cell point of view. But both these systems were available for our IFA-649 testing that I'm going to talk about subsequently.

Next slide, please. Okay. So now, based on those tests, I'll show you a series of at least four slides here which give insights into the mechanism and kinetics of gas release and fragmentation.

The first one is a typical plot of this online monitoring of temperature which is this red temperature thing that was applied. And at the same time, the instantaneous release of krypton-85 is monitored here in the green part.

So as can be seen here, the significant gas release occurs around 200 degrees C in high burn-up fuel. And the onset of fragmentation is related to the pressure that each gas bubble has. And the gas bubble pressure estimation was done experimentally by this pre-characterization that I talked about, EPMA, SIMS, and SEMS studies as irradiated.

This particular plot is on as irradiated sample. And you can see that the estimated pressure in each of these bubbles, typically shown on the right micrograph here, ceramograph, as a function of the bubble radius, the larger the bubble, the lower the pressure which, again, intuitively sounds right. So this was experimentally confirmed as well.

Next slide. So this was experimentally confirmed as well.

Next slide. Well, another aspect of mechanistic study was that gas release begins at a lower temperature where you have partially transformed structure in the fuel. So if you have partial HBS in the sample, your release, of course, as you can see in the left block here, essentially the same treatment of temperature times protocol. But the sample to the left was at 76.1 gigawatt vapor metric ton local burn-up. And it showed a peak also corresponding to a lower temperature, as well as around 1200 degrees C.

Whereas when you go the high burn-up, a fully transformed sample, again, the same thing. But the burn-up local, burn-up of the sample was 103.5 gigawatt vapor metric ton. Again, through peak characterization we can ensure that it's fully transformed rim structure or high burn-up structure. And the release occurred only around 1200 degrees C. So again, an interesting insight that the modelers could use.

Next slide. So here a very limited testing on what's the effect of RAMP rate. Again, in a hot cell bolstered ideation study we can use in our system two different RAMP rates, .2 degrees per second and 20 degrees per second. In order, two orders of magnitude higher RAMP rate.

And we could see that as the RAMP rate increased, the higher efficient gas release and smaller fragment size were observed. Some of the, we presented you, several graphs are shown to the right here.

Next slide. So, next one is the, gets even further inside into the mechanism of fragmentation process. And that is to say that when you compare untransformed fuel with partial transformed fuel and the fully HBS sample as a function of different burn-up that is plotted on the left slide here. Left plot here.

You can see that fuel with no HBS transformation essentially sees no fragmentation. Even if you anneal it all the way up to 1600 degrees centigrade. But fuel with partial and full transformation starts to release gases and shows fragmentation right around 1200 degrees C.

And the higher the HBS conversation the particle size distribution after the scram application of the ram extender temperature ram, was different. So you can see that the larger, the higher burn-up showed more finer particles.

Again, a lot of these things are quite intuitive. And again, also seen in integral tests, but in a very global way. But in doing the separate effect testes like this, focused on mechanism aspects of the phenomena, we can confirm and provide some modeling quality data in each one of these cases.

Next. Next slide please. So, just to now wrap it up on what we have been doing is that it's been a careful experimental work starting all the way from a very well designed irradiation program in the Halden Reactor several years ago.

Uh, we could get almost homogenous fuel of lone pedigree. And they will then apply to the separate testing in a cost effective way to answer many questions about the mechanism of fuel fragmentation in LOCA situation. Or for that matter, any other temperature transient that could be applied on the fuel material.

The initial scoping studies clearly showed the effect of the PCMI, or what I would say hydrostatic restrain. And that's an important aspect that is very, uh, helpful in sort of mitigating fragmentation if you have strong PCMI.

Also, we are continuing to work on things like fulfilling even more existing knowledge and data gaps on PCMI and stress corrosion cracking, and what we call burst release, or also sometimes equally called transient fission gas release.

We're also trying to establish, because the IFA-649 iteration was very edition. Well, how does that represent what happens in a fuel rod? So, equivalence of the IFA-649 behavior and mechanistic studies to fuel, PWR fuel pellet at the high burn-up is shown on the next slide here. This is still work ongoing.

Next slide please. So here what I'm showing is a ceramograph of a typical high burn-up 68.8 gigawatt day, seven spam cross-section of a PWR fuel. And you can actually see the dark zone right in the center of the pellet there.

And if you go to a final magnification, you actually have four micro-structure zones roughly across the radius of the pellet. Those are the pink, yellow, green and purple there on the second figure, to the right of it.

And what we are doing as we speak actually, literally, is to take two adjacent pellets and take the radial cores in each case, limited only to the central part, the mid-radius part, the outer mid-radius part and the rim part. And then subject them to the similar temperature transient in MIRARG and Mexico facility that I described before.

And tried to then give specific behavior based on the micro-structure of the fuel pellet across the fuel radius. And relayed them to the RAMP rate and other variables that I talked about. And hoped there is then these modelers would have data, modeling quality data, from each of these specific zones.

And when they smear it over the entire pellet cross-section they could come up with a more reliable model of how the pellet itself reacts to a temperature transient. Which of course would also support a temperature gradient. And again, if it is a LOCA then there is a scram situation. And only the decay heat and gamma heat is present.

So, all those could be factored in with careful modeling. But our objective has been in this program to shed light on mechanistic model with careful experiment implementation.

And that's, I believe, is my last slide. I'll be happy to answer any questions.

MEMBER PETTI: Yes, this is Dave. I had a question on the last slide. What's the schedule of doing this?

MR. YAGNIK: We are working on this, the material is available. Some characterization and coding has been done. So I would say another six months or maybe a year. This is in the current phase of our program that we are working on this.

MEMBER PETTI: Thank you.

CHAIR BALLINGER: Other questions? Okay, I've lost track of the order, so --

(Laughter.)

CHAIR BALLINGER: -- sorry.

MR. WELLS: I'll keep us going. I'll keep us going, Ron. So this is Dan.

So I think we're going to shift gears. We're going to skip a few presentations. We'll skip the overview. If we have time we can come back to CRAFT. But we'll skip forward and move to spent fuel criticality and let Bob and Hatice go. So we'll shift to back end a little bit.

MEMBER REMPE: As we get ready for this presentation, if people are out there on the internet and, please, if you're not talking be sure and mute. Because I think that we've got someone coming across the internet with some background noise.

MS. AKKURT: Okay. So I will be presenting spent fuel pool criticality issues for ATF high energy and high burn-up.

I am Hatice Akkurt. And Bob Hall will be presenting the second part, part of this presentation.

Now, for this group, you know, either in front of ACRS, right before the approval of the Regulatory Guides 1.240, which is based on NEI file 16, criticality guidance and EPRI benchmarks for depletion uncertainty.

And just as we thought that we were done and regulatory guidance is issued, but the issue is that, that guidance, as well as up to five percent enrichment and 66 for 2301. So when it comes to LEU-plus, which is the yellow region, you know, of your for higher enrichment, higher burn-up, you don't have much data. And also, the existing guidance does not necessarily apply.

So basically to address, find out what are the issues and to draw (audio interference) have a roadmap, we performed a spent fuel pool criticality for advance fuel working group. And this working group was composed of members from utilities, vendors and NEI.

We had a multi-day working group meeting that we basically did vote for the guidance, recommend regulatory guide. And tried to identify, you know, when we move to a higher enrichment, higher burn-up, for ATF also, what are the issues that need to be revisited and what are the technical gaps.

And first round was identification of gaps and issues, potential issues. And then the second round was (Audio interference) --

I'm hearing myself speaking. There is an echo, right?

(Off microphone comments.)

MS. AKKURT: Okay. So second --

(Off microphone comments.)

MS. AKKURT: Okay.

(Laughter.)

(Off microphone comments.)

MS. AKKURT: Okay. The second round was, you know, who is going to lead this thing. It asks, you know, basically. And some of them were assigned the utilities because you need advanced analysis for some cases, right? And some of them were, you know, vendors needed help with some of their codes and some of the data gathering needs to be coordinated.

But two generic issues, which are going to be the focus of this presentation, were identified. And EPRI is leading those two generic issues. The first one is the criticality code validation. Which is basically, how do you validate PR codes, you know, do we have enough critical benchmarks, or do we even need. This is for fresh fuel.

But when it comes for spent fuel, how do you address depletion, burn-up credit, uncertainty and bias. And as I mentioned, these two tasks were identified to be led up by EPRI.

So, depletion, uncertainty and bias, what we mean here, you know, this slide and the next slide is directly from the ACRS meeting we had in March 2021. So, what is the need here?

Well, for fresh fuel you go to contamination of critical handbook and you use the relevant experiments based on your criticality for your code validation. When it comes to spent fuel we don't have experiments to validate our codes.

And performing new experiments, you know, not many facilities exist and you have to perform many experiments at different facilities. It's not that feasible unfortunately necessarily.

The third option is, you know, using the fresh fuel assumption. Well, given the fact that if you have spent fuel using the fresh fuel assumption is overly penalizing, you know. So how do you account for this?

So, I'll do address this in 1919. The famous CRAFT memo was issued. Basically, if you don't have the data you can use five percent, which will account for your uncertainty, which includes your isotopic composition, cross-section uncertainties and so on and also the bias.

Obviously this being very easy to use. You know, easy to be being implemented. You thought you would use this. And then around 2009 NRC asked for the technical basis.

So at that time two parallel paths were followed. The first path was NRC sponsored work, which is based on chemical assays. The issue there is, chemical assays that were performed in '60, '70s have very large measurement error. And then obviously Venn Daniels is based on the measurements that has very larger errors, you know, it showed that CRAFT memo may not be conservative.

But EPRI used a different approached. Used the flux MAAPs, which are measurements with much lower uncertainty and used in regulatory space. From four reactors, 44 cycles, to do the analysis, develop EPRI benchmarks. And two reports were developed and then submitted for NRC review. And these were finally approved in 2020.

And the final numbers became, for the reactor, reactor is a different uncertainty and bias, as a function of burn-up they are tabulated here. The first thing is, they showed that five percent is ideally conservative and there is additional margins. But again, this is valid up to five percent.

Starting this work we asked ourselves two question. First question is, that we move to LEU+, is the CRAFT memo still valid. And the second question is, MOUs are regulatory approved EPRI benchmarks and extend this for LEU+ for higher enrichment and higher probability.

So how can we do this? Now, to do this, you know, we said, okay, it's always good to start from a physics side under what changed. And you will go to high enrichment, high burn-up.

And also, you know, some of the modeling tools. There are tools like scales, packages, TSUNAMI, which does the similarity analysis. Sampler does uncertainty analysis. You know, basically using perturbation. Use that and look at what's changed with how similar are these two systems.

And then, you know, extend EPRI benchmarks to determine, first of all, you know, is it possible to extent EPRI benchmarks and how we can do this. And to do this, the stochastic sampling was used.

So physics. Well, I mean, all the actinide production and, you know, in terms of number uncertainties and reactor contributions were evaluated. And the table on the left shows, you know, what changed. The biggest change was for uranium-235. Which is good news because uranium-235 is the one we know the best and, you know.

And in terms of the number density contribution, the figure on the bottom shows the number density changes for five percent versus eight percent. But I will, you know, point your attention to the fact that five percent, 60, you got up, is identical to eight percent and 90.

I should also mention that, you know, based on earlier discussions, when you say LEU+, you know, it is, at the moment, no one is planning to go up to eight percent and 90. But you know -- or if you wanted to have, you know, an entire role that will cover in the future if that is enough for the development for the full site.

Now, the graph on the top is showing the reaction to contributions. You know, five versus eight. And again, I will show that. Point you to the fact that five percent, 60, and eight percent, 90 are in the curves.

Now, the second tool of the TSUNAMI says there is uncertainty. For those who are not familiar, this is recently -- a fairly recent tool that has been added to the SCALE computational package. So basically there is two systems. And it basically says, if your similarity portion that is off (Audio interference.)

(Echoing.)

MS. AKKURT: -- from zero to one to a hundred. But based on ISG 10 it says, anything, if the coefficient is higher than 0.95, they are high, very highly similar. 0.9 is, you know, high similarity. And then the caveat it says, you know, at moments code is limited in use, you know, coefficients less than 0.9 should not be used. But I do believe that is based on other sources as well. 0.9 may be too stringent.

But in any event, we did five percent versus eight percent for different burn-up values. The diagonal, I know it's a very crowded table and the subset is shown on the right, but I will point to the diagonal. And the diagonal everything is about, you know, 95 out of 100 or 0.95.

The figure on the right is showing for different burn-ups and arrangements, five versus eight. And surprising here also, that you compare five percent, 60, and eight percent, 90. Similarity coefficient is coming as 0.99. Which is basically telling you that these two systems are almost identical.

Why is that important? Again, it's consistent with the physics, but also you have yielding to the fact that when you go to a new class there won't be much implicit uncertainty.

Now, coming back to EPRI depletion work. As I mentioned, those are based on the facts MAAP measurements for the four reactors for four cycles. You said that, you know, if you didn't -- and those were binned, and you know, that's the classic example if you didn't have the data. Assuming we don't have the data. And what will be the, and why is it uncertainty bias.

The table on the top is based on the measurements. The table on the bottom is based on the stochastic sample. Basically, as you can see, you know, they are very able to reproduce, but it's some conservative. Given that this is done with the lack of data. You know, being in good agreement conservatively is a very good news.

So starting, you know, getting some information from the five percent, extending to eight percent, then if you see that, even if you go to eight percent, 90, still the depletion uncertainty of five percent CRAFT memo is still conservative even if you move to higher enrichment and higher burn-up.

So going back to our original questions in this work, you know, is the CRAFT memo conservative, yes? And can we use EPRI benchmarks in, as they stand for the NEI class, and the answer is yes. And you have done this using multiple extended approaches starting from physics and uncertain analysis and extend. So it supports this full version.

MR. HALL: All right, picking up from there. We also wanted to look at fresh fuel validation in the spent fuel pool and new fuel storage areas.

Next slide. So the question was, are there sufficient critical benchmarks. In the five to eight weight percent range for LEU+. And the second one is a little hint of the results and it says, are they needed. And I'll fill you in on that clue later.

So this was a fairly simply and short story here. We surveyed the available benchmarks from the handbook and the NUREG and some other places that you get these from.

Using the NUREG/CR-6698 approach. We performed fresh fuel similarity analysis using TSUNAMI, which Hatice just talked about. We do this for a huge number of models. So we're looking at new fuel vaults, spent fuel pool, different types of racks, different types of fuel assemblies, with boron, without boron, with poisons in the pool, without poisons in the pool and so forth.

So we're trying to cover the landscape here in application space. But we're limiting ourselves to comparing apples-and-apples. So we compare in a particular configuration, five weight percent assembly. We compare it to a six, we compare it to a seven, we compare it to a eight. And we asked TSUNAMI, are they similar.

I think that's good. Next slide please. So in the first part, the survey of the benchmarks, the NUREG Table 2.3 has a good bit of guidance that was established by criticality specialists at the time and said, hey, these are necessarily conservative criteria in order for consensus to be obtained for this guidance.

So, in the table are characteristics that you should try to match to your application, as well as some boundaries on how far different things could be. And these experiments are still applicable.

And they fall into three general categories. And that's what materials do you have, what is the geometry and what is your energy spectrum.

So if you look on the table, that's a snippet of the table. It's not the whole table. But in blue is what we're interested in for this analysis.

So fissionable material we're interested in U-235. Fort the isotopic composition, we're interest in being within about two percent of our enrichments. The five, six, seven and eight.

The physical form we have UO2 pellets in pins. For the moderator we have water. What form it is, liquid. The density is room temperature liquid water. The HBU ratio, we're interested in things that are similar to fuel assemblies.

And so, as long as your pin diameter in your pitch, and pitch is similar, we meet that criteria. Again (Audio interference) as part of geometry, as close as possible to the actual case but not as important as the materials. We have fuel pins in water, and so those were the experiments selected. And we need to stick to our neutron energy range, which is the thermal range.

Next slide.

MEMBER HALNON: Any poison considered?

MR. HALL: In this case there are experiments that were pulled with poisons and without. We're not making that distinction. We didn't try to limit ourselves to one.

So this is sort of the broad sweep of experiments people would pick for a spent fuel pool application where they, one rack may have poisons, the other rack may not. You pull a set of experiments that will cover the range.

MEMBER HALNON: Okay. It's the same with the water, no boron?

MR. HALL: Boron and no boron. Yes.

MEMBER HALNON: Thanks.

MR. HALL: So, this a fairly simple answer. We go to the handbook, we go to NUREG source and a couple of other places. And the histogram tells you what we found for pins in water sort of applications. And what you see is a pretty good spread all the way up to the upper sevens.

And there is, in particular, some stuff in the mid-sixes that is well represented. So, it didn't take a lot of work to do this particular part. But it's a satisfying answer in that there are a substantial number of that middle of that five to eight weight percent range of enrichments out there --

MEMBER MARCH-LEUBA: This is fresh fuel, right?

MR. HALL: It is.

MEMBER MARCH-LEUBA: Even though you want to apply it to spent fuel?

MR. HALL: No.

MEMBER MARCH-LEUBA: You want to apply it to fresh fuel?

MR. HALL: That's right.

MS. AKKURT: Yes. As I said, there are two --

MEMBER MARCH-LEUBA: You're confusing me with the title of the presentation.

MR. HALL: Right. There two pieces and they're connected together. When you do the fuel pool criticality analysis, you do a fresh fuel validation for your code and then you have to say, well, I deplete my code now, and I have depleted fuel, now what I need to do. I need to know how much uncertainty I have. And that was the first part of the presentation. How do I put a foundation underneath that five percent assumption.

And NRC also currently requires the HDC criticals as well, which are from France. And it's intended to be a mix of isotopes that mimics mid-burn.

MS. AKKURT: Yes. It's part of spent fuel pool criticality. For the criticality portion you still need to do the fresh fuel to make sure that, for code validation.

MEMBER MARCH-LEUBA: You're evaluating the code?

MS. AKKURT: Yes.

MEMBER MARCH-LEUBA: When you burn the fuel, whether you have five percent to start with or eight percent, if you burn five to 60 and eight to 90, you have the same amount of u-235, you have about the same amount of plutonium, and a little higher of the poisons that doesn't count.

MR. HALL: Yes.

MEMBER MARCH-LEUBA: Anyway, go ahead.

MS. AKKURT: Yes. That's why the depletion uncertainty points, ours was pointing to that.

MEMBER MARCH-LEUBA: Yes. You have the same isotopics. You --

MR. HALL: Right.

MEMBER MARCH-LEUBA: -- this --

MR. HALL: I called that, I called that deja vu. You know, we went from five weight percent, we go up to eight. You burn it farther. And we've already been there. When we get to eight weight percent of 90, we've already been there.

When we were looking --

(Simultaneously speaking.)

MEMBER MARCH-LEUBA: -- into the same condition?

MR. HALL: Yes. Same sort of isotopics. Thank you.

All right. For the second part of this second half, the pictures on the right are images of our two-by-two rack configurations. And as you can see, we didn't necessarily fill the racks, we just had some two out of four, three out of four, three out of four fuel assemblies in those four rack cells.

The list on the left tells you all of the different conditions that we looked at. And again, we're going to do a very simply apples-and-apples comparison of five weight percent through higher enrichments and see what the code tells us.

Next slide please. And this is a really boring slide. It says, everything is 0.98 or higher. And, you know, there is some disagreement over where the cutoff should be for similarity, but no one would argue with .98. .98 says, these are almost identical.

And so, for the fresh fuel situation in new fuel storage in spent fuel pools, so in that very thermal environment, we have the same materials, we have the same geometry. We have about the same spectrum. There really isn't anything --

So even the NUREG criteria would tell you the same. Say, look, we have almost the same, we have the same geometry, we have the same materials. We also still have the thermal spectrum, it ought to look similar. The code says it's .98 or above.

So that means, practically speaking is, that if you can successfully validate your code for five weight percent fuel, you don't need to do anything else for five to eight weight percent to go above five weight percent. Or the fresh fuel condition. The code is telling you that you've already covered that.

So, in answer to the original questions, significant, there are a significant number of benchmark experiments in the five to eight percent range. And people will include those when they do the criticality analysis.

However, enrichment is an extraordinarily weak variable for this application. It seems like it should be a strong variable, but it is not. And we get that same result across numerous fuel types, rack types --

MEMBER MARCH-LEUBA: Yes, since we're not assigning anything, we'll take your -- I'll take your word for it, but obviously key is, what does TSUNAMI do because I can bring your code up, .98 without distribution around it.

MR. HALL: Sure. Well, TSUNAMI is written by Oak Ridge to try to --

MEMBER MARCH-LEUBA: I work in that -- I used to work in that floor, two offices down.

MR. HALL: Okay.

MEMBER MARCH-LEUBA: So I know them people. But I don't know this.

MR. HALL: So out take on TSUNAMI is, you know, it's looking at every material, every cross-section. It's looking at every co-variants, point in the co-variants matrix. It's very stringent.

And so, what we've convinced of, is that if TSUNAMI tells you it's a 1.0, as close to 1.0 or very similar, it really is very similar. What we're not as sure of is when the TSUNAMI tells you, you have a .7, we're not as sure exactly what that means. It doesn't mean that there is no value for the critical experiment, but it means it's not identical. So that's sort of the gray zone.

MS. AKKURT: Yes.

MR. HALL: But in this particular case, what we're presenting today, there isn't really a, in my mind, a question that .98 and above really does say the same.

MEMBER HALNON: There was some insight you can gain by the spread. Was clearly expanding the higher enrichments of something --

MR. HALL: Yes.

MEMBER HALNON: -- so you can extrapolate that out to some extent. I don't know if you want to go linear or something else.

MR. HALL: TSUNAMI did say that there is a difference. It said that difference is quite small.

MS. AKKURT: And I should say that, you know, we didn't, in the interest of time, we didn't include the special on-coded as you had all expect, you know, from an attorneys point of view. You're aware that chrome, chromium coating, or other coating materials, impact is negligible so that's why it is not spelled out, should be part of the record.

MR. HALL: Yes. Chrome coating and doping had almost no effect on --

MS. AKKURT: But TSUNAMI can show these similarities. And, you know, the group that is working on an accruement and so on. And God rest your earlier comment.

MEMBER MARCH-LEUBA: Changing subjects. You promised me earlier that you were going to talk about --

MS. AKKURT: Yes, we will.

MEMBER MARCH-LEUBA: -- long-term decay. Oh yes? High burn-up fuel.

CHAIR BALLINGER: I noticed there's a hand up in the --

MEMBER MARCH-LEUBA: Oh, it's the next presentation.

MEMBER KIRCHNER: I did.

MEMBER MARCH-LEUBA: Okay.

MEMBER KIRCHNER: You're right on key.

(Off microphone comments.)

MS. AKKURT: The next one in the agenda is eight.

MR. WELLS: Yes, we're going to -- so is there any more questions on this one, or should we do that and then --

MEMBER KIRCHNER: This is Walt Kirchner. Just an observation. What you're really saying is, when you say the enrichment impact, at least in terms of doing your benchmarks isn't that significant, is that for this kind of lattices in LWR in water, in light water, that's more dominant in the range of enrichments that you're looking at then, in other words, the geometric configuration of materials is probably more dominant than the enrichment. As long as you're staying within a reasonable 15x15 to 17x17

PWR or the same for BWR bundles.

MR. HALL: Yes, I think that's a fair characterization.

MEMBER MARCH-LEUBA: I mean, honestly we're not reviewing the results from your analysis, we're taking your word for it. I would like to see how much gadolinium is in there, how much burnable portions are in there, how much does it affect the spectrum and everything else.

For you to be able to be below an eight percent fuel enrichment in the same core that used to run a five, you need to put a lot more gadolinium in it.

MR. HALL: Sure. I know the, when we said we covered a big-broad spread of types of fuel, we have three and eight weight percent gad. I don't recall how many pins are in those.

MEMBER MARCH-LEUBA: No, I'm saying is, if you have a five percent uranium, you have 25 enrichment --

MR. HALL: Yes.

MEMBER MARCH-LEUBA: -- bundle. And now you're putting in the same slot an eight percent enriched bundle, it goes super critical. You have to cut it down somehow.

MR. HALL: Yes.

MEMBER MARCH-LEUBA: And you cannot change the boron in the water because there are limits. So you have to do it with burnable poison. And burnable poisons are difficult. Notoriously difficult to model.

And I didn't see any mention in your similarity analysis for gadolinium. We're not reviewing that, we're just getting here information. But if we were, I would be asking you questions about, like burnable portions are very difficult to model. They're very dark in particular energies, so, anyway.

MR. HALL: Okay, that's a good point. All right, so changing gears a little bit. This is --

MR. WELLS: So, Bob, sorry, so just so everyone keeps up. So this is 13.

MR. HALL: Yes. I just, I was --

MR. WELLS: So in the PDF it's Page 193. But that's where we're moving to next. Yes, I thought that might be helpful so I looked it up.

MR. HALL: All right. So I'd like to present results of a scoping analysis that we did for a transition from current burnup and cycle designs to ATF/LU+ and higher burnup cycle designs. And, again, I would say it says right on the title slide, preliminary scoping results.

The report that this is in is actually going to publications today so the review process has been completed for that. And the numbers I am going to show you today are similar to what is in that report, very close. I will point out where there is a slight difference.

Okay. So the purpose and goals here are to look at the back end of this transition to higher burn up fuel and ask the question what is the impact on managing your spent fuel pool inventory in the spent fuel pool and in your migration out to dry storage? And, again, this is a high level first effort here. And we had a couple of goals.

First, we wanted to get some realistic estimates of what a move to higher burnups looks like by way of traditional typical core designs for a PWR and then we want to identify key variables. What are the really important variables in whether or not this will affect your storage?

We want to estimate the trends. We want to say something to the extent we can about the ISFSI dose rates. And, again, we are using high level scoping, simplifying assumptions. This is not plant specific or dry storage system specific. But it is intended to identify the key variables and the trends.

All right. As you know, there are tradeoffs with this increase in enrichment and burnup. As we push enrichment and burnup up, the number of assemblies that we use and the number of canisters that we need to load comes down.

As we push burnup up, the decay heat and the dose rate will go up. The cooling time and increased enrichment will push that back down to some degree. And then when we talk about the overall inventory, when we increase cooling time requirements on individual assemblies, that increases the number that you have to hold up in the spent fuel pool. However, you are discharging fewer assemblies to the spent fuel pool. And so there are a lot of trade off effects here that we need to gather up into one net effect and that is the goal here.

Next slide. So a couple of slides just to remind us what the decay heat looks like as a function of cooling time for different burnups. No surprises there. But when you take the plot on the left and you transpose it onto the right to constant decay heat, those are constant decay heat curves on the right.

And what that says is, if we have discharge burnups in the 45 gigawatt day per ton area

-- I think the industry average is around 47 right now -- it doesn't matter that much what your decay heat limit is in your canister. You don't need to wait that much longer, you know, across that decay heat span.

But when you push out to say 60 gigawatt days per ton, that gap opens up quite more, a good bit more. And so the effect on the holdup time is starting to get large as we push out to the burnups that we are talking about. So how much of an effect is that? Is that something we are going to need to be concerned with?

So first, the point we make on this slide is we need realistic burnups and enrichments. Enrichment does play an offsetting role for dose rates, and it also plays an offsetting role for decay heat. Enrichment and burnup, increased enrichments reduce actinide production in particular.

MEMBER MARCH-LEUBA: Because you are burning --

MR. HALL: You are burning more U-235 and less of other materials. And so --

MEMBER MARCH-LEUBA: Hard to believe.

MR. HALL: It manifests. Yeah, it manifests both in decay heat and in the dose terms.

The degree to which depends on how much burnup did you add and how much enrichment did you --

MEMBER MARCH-LEUBA: It's a function of burnup, not of enrichment.

MR. HALL: Yeah, both, right.

MEMBER MARCH-LEUBA: No, sir. No. The production of U-239, the production of U-239 is depending on how much U-238 is in there.

MR. HALL: And also the spectrum. In other words when you burn the fuel assembly with more U-235, you have lower flux in a constant power reactor. And so when you do that you get less rapid build-up of U-239 even though you have the same amount of U-238.

MEMBER MARCH-LEUBA: Yeah. Okay.

MR. HALL: So the actinides will build in slower at the higher enrichments.

MEMBER MARCH-LEUBA: A little bit.

MR. HALL: So it is one of those tradeoff effects.

MEMBER MARCH-LEUBA: You have to look at the calculation.

MR. HALL: Yes, yes.

MEMBER MARCH-LEUBA: I see here that 75 gigawatt, the hermetic term versus 15 you have a factor of 5 in decay heat. Some of those are linear. I mean, if you look at the left figure --

MR. HALL: Oh, yes.

MEMBER MARCH-LEUBA: -- of 40 cooling time.

MR. HALL: Mm-hmm.

MEMBER MARCH-LEUBA: -- the heat is almost linear.

MR. HALL: Right.

MEMBER MARCH-LEUBA: It is a function of -- I mean, there might be a secondary effect on enrichment, but that figure on the left tells me it wasn't a burnout.

MR. HALL: It is more than linear. I would love to have a ratio curve for you.

MEMBER MARCH-LEUBA: Obviously, you burn it wrong, and you are burning more plutonium. So you have to do the numbers. Basically, at the end of the day you have to do the numbers. You cannot guess.

MR. HALL: Yeah, yes. I think that is the point of this analysis. It isn't clear to what degree you will offset. And our point on the next slide is that we need to use realistic burnups and enrichments because of this post-tradeoff. We need to get the tradeoff right. Utilities do not pay for more enrichment and not burn the fuel farther.

They will get -- at least the burnup and the enrichment go together and that is what you see on the right plot. That is the U.S. EIA annual discharge burnup data. And you can see as we have gone through time, we have also gone through enrichment increases. We started out these cycles in the three-way percent area --

MEMBER MARCH-LEUBA: Mm-hmm.

MR. HALL: -- and now we are up to, you know, the mid-fours, and we've gone up in burnup as well. The main point of this slide is what is the batch discharge burnups we should be thinking about for the various limits of the peak rod burnup limit?

So what you see in the current situation is peak rod limit is about 62 gigawatt days per ton. That's shorthanded. It varies a little bit from that. But at 62 we see maximum discharge burnups of about 47. And so why the big difference?

And on the left, I won't go through all of those, but there are a lot of practicalities in core design. I spent 37 years in the core design group. The practicalities of core design is you can't push every assembly up to the limit. You know, there are a list of things on the left there that you have to account for. And so the maximum practical batch average burnup is about 80 percent of that peak rod burnup limit, give or take about 5 percent.

And so we look to that because what we are proposing here is we are going to look at what happens when we go from 62 gigawatt days per ton to 75. So we have to know what is our batch burnup going to be looking like to that not what the burnup limit is because that --

MEMBER MARCH-LEUBA: I can tell you. I don't even need to calculate it. If you go from 18 months to 24 --

MR. HALL: Mm-hmm.

MEMBER MARCH-LEUBA: -- it will go up 25 percent.

MR. HALL: But it doesn't because you have to use more fuel. Unless you keep the same batch size --

MEMBER MARCH-LEUBA: The low to this, yeah, this SFP is higher, 25 percent higher.

MR. SMITH: If you go to 24 month cycles, the loss of efficiency will actually reduce the burnup.

MEMBER MARCH-LEUBA: You will have to speak up.

MEMBER REMPE: Yeah, say your name and lean in.

MR. SMITH: Yeah, Fred Smith. When you go from 18 to 24 month cycles, the batch size increase is so large that the utilization drops and so the average burnup also drops. You can't go from 18 to 24 months without --

MEMBER MARCH-LEUBA: You're wasting --

MR. HALL: -- putting a lot of fuel into the core. And you are wasting. It is not as efficient.

MEMBER MARCH-LEUBA: You are wasting fuel. You are wasting good uranium.

MR. SMITH: That's right.

MEMBER MARCH-LEUBA: Anyway. You just have to do the calculation.

MR. HALL: And this is the calculation done in approximate form. So for current PWR cycles, where we are right now is about 44 gigawatt days per ton at 4 weight percent. I'm not saying we are at 4 weight percent but that is a nice benchmark point from the discharge data. And the discharge burnups increase about 11 gigawatt days per ton per weight percent. So that is a rougher rule of thumb.

So for our scoping, we are using an Oak

Ridge tool from the reference that you see there to do estimates of what the discharge burnups are for each batch and sub-batch. Given the limits that we just talked about, 80 percent of the 62 gigawatt days per ton, we are 80 percent of the 75 gigawatt days per ton.

And that figure on the right is from that Oak Ridge report by the way that -- it just points out that 24 month cycles and 18 month cycles, the relationship between enrichment and burnup, there is a third variable in there as well and that is the specific power of the core. So they are all important in determining what should be -- what is the enrichment and burnup combination going to be to achieve the cycle that you want, whether 18 months or 24 months.

All right. So given that, we have -- next slide, please. That was the next slide, I'm sorry. I was looking at my own slide. Sorry. I didn't tell you. So we wind up with three cases we want to look at. We've got a core size of 157 assemblies, 17 x 17 PWR, 98 percent load factor.

And these reload batches are average in equilibrium. So, you know, when you see two assemblies in a sub-batch, it doesn't literally mean two assemblies in a sub-batch. That is an average. But what you see are three cases that we look at, a base case that is 18 month cycles, similar to what most PWRs are running today and then you have an 18-month higher burnup case and a 24-month higher burnup case.

So we run the tool, trying to stay within our discharge burnup limits. And we push the enrichments up and shrink the batch sizes to the extent that we can, and this is where we wind up. We start out at 4.4 weight percent with the batch sizes you see there, two sub-batches, 41 and 25 assemblies.

For the 18-month cycle, we are running 5.1 weight percent, batch average, with 53 assemblies out of the 157. It is just above 33 percent batch load. And then for the 24-month cycles, we move up to 5.4 weight percent with 74 assemblies loaded. And you can see what the sub-batch discharge burnups are there and what the average discharge burnup is for those situations, those three cycles we want to look at.

So we also need to know what is the canister decay heat limit so how long I have to keep it in the fuel pool depends on that. So these are the three cases that we have looked at for the canister decay heat limits.

The first one is a uniform loading. The second two are zone loadings, similar to some licensing from the last 10 years of modern 37 assembly canisters. By the way, the total heat load in these cases is between 42-1/2 and 50 kilowatts per 37 assembly caster.

Next slide, please. Okay. So we did first a simple first simulation of equilibrium spent fuel pool inventory assuming a uniform canister load decay heat limits. And the assumption here is that we load everything that we can into a canister that qualifies for loading into a canister. If it doesn't qualify, we hold it up in the spent fuel pool. And we run that for several cycles until we get to an equilibrium of how many fuel assemblies are what we call stranded assemblies in the fuel pool waiting -- that have to wait longer before they can be loaded into a canister.

And what you see from the curve on the left is that the impact on how much fuel you need to hold up in the fuel pool above and beyond what the base case is telling us. So on the left axis, on the Y axis, that is how many full core equivalents additional space do you need in the fuel pool to accommodate this transition to higher burnups? And the answer is it depends on your decay heat limit in your canisters.

If you push off to the right and you had a very high decay heat limit in the canisters, there is essentially no effect. If you stay to the left and you have low decay heat limits on your canisters, then the answer is a lot.

And so we have identified our first key variable, and that is what is the decay heat limit in the canisters?

MEMBER HALNON: Bob, does this take into effect the loading patterns that are presently having to be done because of the B.5.b and other aircraft impact issues?

MR. HALL: No. This is a hypothetical high level spent fuel pool. If you had B.5.b, of course, that limits how many assemblies you could --

MEMBER HALNON: Right. So --

MR. HALL: Yeah.

MEMBER HALNON: -- okay. So this is obviously a retro.

MR. HALL: Right.

MEMBER HALNON: Okay.

MR. HALL: So this is really a delta exercise. So this could be for a B.5.b pool first and then second.

MEMBER HALNON: What is the worst number of assemblies in your nominal core that you are looking at?

MR. HALL: In the example that I had on the previous slide was 157 assemblies.

MEMBER HALNON: I'll take a look. I missed that. Thanks.

MR. HALL: We actually did it for 100 assembly, a hypothetical core 157 and 193.

MEMBER HALNON: Yeah. That pretty much bounds what we are --

MR. HALL: Yeah. We wanted to exercise some of those assumptions as well. So on the next slide you will see the slightly finer pencil answer.

Next slide, please. This is all about how much additional space you need in your fuel pool. And the answer is it depends on your canister load. Again, we already saw that in the previous example. But for a 24-month cycle transition, somewhere between .2 and .8 full cores of additional fuel space in your pool. Now this is assuming that the canister decay heat limits have not changed.

MEMBER MARCH-LEUBA: And then regarding the canister revolution, is the big rod limiting the canister or is this the evidence rule?

MR. HALL: I have not loaded canisters either so full disclosure. The limits we are dealing with is the -- we are looking at just the decay heat --

MEMBER MARCH-LEUBA: Yeah. What is the (simultaneous speaking).

MR. HALL: -- and the zoning of the decay heat.

MEMBER MARCH-LEUBA: I have bundles with -- whole bundles and (simultaneous speaking) --

MR. HALL: There is definitely --

MEMBER MARCH-LEUBA: -- it has to be limited error if you are going to put a very high bundle with a weak cover.

MS. AKKURT: There are two limits. One is, you know, on the limits --

MEMBER MARCH-LEUBA: Mm-hmm.

MS. AKKURT: -- there is a limit for that. And the one, the system design, it varies and then one is for the total.

MEMBER MARCH-LEUBA: Yeah. Again, you have to render the simulation for your particular plan --

MR. HALL: Yes.

MEMBER MARCH-LEUBA: -- your particular matter, subject to loading your particular canisters. And is an extra ordinate core reload for spent fuel a problem? I mean, it used to be a problem when we didn't have dry storage.

MR. HALL: Right.

MEMBER MARCH-LEUBA: But today is it a problem?

MR. HALL: That is the question. And the answer is it depends on your particular situation.

MEMBER MARCH-LEUBA: It depends on what you are doing with your canisters, right?

MR. HALL: Right, right. So our part -- I mean, you are coming to the same place we are. We are identifying the key variables and what the trends are.

MEMBER MARCH-LEUBA: Mm-hmm.

MR. HALL: And we are saying, okay, you, utility, you need to analyze this.

MEMBER MARCH-LEUBA: Well, this is what I was telling you. You went to 24 cycles with high enrichment or higher burnup. You are going to cool them longer.

MR. HALL: Yes.

MEMBER MARCH-LEUBA: The percent is -- as we said, 30 years ago do you have room in your pools? But now we will be moving into dry storage.

MR. HALL: All right.

MEMBER MARCH-LEUBA: You are going to have to move it to dry storage faster.

MR. HALL: The dry storage campaigns are, you know, highly intrusive to the site operations. So you have to plan it plus you usually do the -- so you can plan out a full core offload at some point --

MEMBER MARCH-LEUBA: Yeah.

MR. HALL: -- in the spent fuel pool. Now you might have to plan it out for a two core offload in the future. So that is what --

MEMBER MARCH-LEUBA: What I'm saying is --

MR. HALL: -- especially with an 80 year.

MEMBER MARCH-LEUBA: It may be a problem.

MR. HALL: It certainly is a logistics issue down the road. It may not be a problem today because if they just did a campaign, you are going to have much more than what you need, but it is a capital expenditure. It's logistics.

MEMBER MARCH-LEUBA: The engineering to me is if I'm there, the plant manager for Plant X, I will ask you to calculate my next --

MR. HALL: Mm-hmm.

MEMBER MARCH-LEUBA: -- 10 reloads and see if I have room, see what I need to do.

MS. AKKURT: And they have tools for that.

MEMBER MARCH-LEUBA: Yeah. Yeah, I see. But --

MS. AKKURT: No, no. I mean, you know, usually the coal operating plants, you know, they do the cask loading campaigns every two or three years so they can do multiple at the same time.

MEMBER MARCH-LEUBA: Mm-hmm.

MS. AKKURT: Because that reduces your training requirements and also gaining experience in terms of logistics and so on. But, you know, if you have cask loader software, for example, a part of it is, you know, for your existing campaign to plan but you can do scoping for the next 10 years, 20 years and so on.

MEMBER MARCH-LEUBA: Probably for this scoping calculation, you don't need to find it. All you need when you buy the new core, when you buy the new core, you can save one bundle, you save a bundle of money. And so you can recalculate your core distribution. But for this calculation, one extra bundle doesn't make a difference. So, yeah, go ahead.

MR. SANTOS: Can I add one thing? This is

Al Santos, NEI. This study was done evaluating the status quo with the current existing canister designs and limits.

There are efforts underway to try to expand those limits due to the hybrid of demonstration casks that was loaded. When we loaded it, we thought it was at a much higher temperature. And we actually did a thermal couple. It was at a much lower temperature than we anticipated because of the -- I think the many conservative assumptions that we put into the thermal analysis of the materials so in the fuel.

So in this case if we are looking at the future, one of the areas that we can change, you know, this calculus, is like changing or, you know, the limits that are on the canisters now. There is already a PIRT underway looking at trying to change that and go up from a 400 degree C limit total canister individual fuel in its cells and increasing the entire heat bundle, you know, the entire canister bundle limits as well as the cell limits. That's what Hatice was talking about.

If you have individual cell limits and locations where they are in terms of how much heat you put in there, but also you have the total canister limits. And so there are aspects of this that this could change in terms of like how much more you need there if you can offload the fuel faster from a pool to the canisters as well as having a hotter, you know, more assemblies in there.

MEMBER MARCH-LEUBA: It all depends on what you have plan to do, right?

MR. SANTOS: Correct.

MEMBER MARCH-LEUBA: I don't know. If I see a point -- including .2, it could be as high as .8. If that is a problem, you are running a tight ship. You should have that much margin.

MR. HALL: Right. When we started the preliminary analysis, we didn't know what the number would be so, you know, .2 is, I think, everyone would agree that's pretty small. That's a pretty small --

MEMBER MARCH-LEUBA: If you can't accommodate that --

MR. HALL: Right. That is a very small number.

MEMBER MARCH-LEUBA: (Simultaneous speaking) running.

MR. HALL: So, you know, we were kind of happy to see that number.

MEMBER MARCH-LEUBA: Mm-hmm.

MR. HALL: It is quite small. All right. So I think we've covered this slide. The results for the finer pencil approaches are broadly similar to the simple illustration.

Next slide, please. The other thing of interest, we have different fuel in the fuel pool now. And we have a little bit more fuel in the fuel pool so what about the peak decay heat. The peak decay heat in the fuel pool happens right after you finish offloading the core. And so we looked at what does the peak decay heat look like for each of these scenarios? And you can see in the table there what we found. We are assuming that the last assembly from the offload goes in about 140 hours after core shutdown. And the numbers in the final report change slightly. They are about 2 to 4 percent essentially. That is the bottom line.

So, again, it is not zero, but it is a small number. And so when you look at the first box that we saw about how much longer the decay time would be at these higher burnups, you wondered, is that going to be a big problem? And the answer here again is coming up, it's not zero, but it's small.

MEMBER MARCH-LEUBA: It's look like it is counterintuitive. So I hope somebody went in to check you out because this does not seem to be consistent with the left figure on -- the figure on Slide 4.

MR. HALL: Next slide, please. We also wanted to say a little bit about dose rates. And we leaned here on an Oak Ridge study that was done in 2020 where they did a uniform canister loading of fuel with different burnup with all fuel assemblies having the same decay heat. So they used 1,200 watts. So in other words you have enrichment and burnup combinations that you decay for enough time to get it to 1,200 watts and then you put it in the canister.

And so what happens as you go up in enrichment and you go up in burnup and you go up in decay time, but you still come back to the same decay heat limit in the canister and the answer is in that plot.

And what that says is this is normal storage of a dry storage canister inside a concrete overpack. So this is the external dose rate that a worker would see walking around the ISFSI. And it is monotonically decreasing as you increase the burnup. That seems counterintuitive but it is because of the extra decay time that is required for those higher burnups prior to being able to put this in dry storage.

That extra decay time offsets -- it decays off the dose contributors such that the net effect is that for normal storage, which is again gamma dominated, those dose rates come down. Now those dose rates are coming down per canister. And in addition to that, we have fewer canisters that are being loaded. Fred talked about approximately 20 percent less. For this study, we are looking at 15 to 20 percent less. And so in addition to the fact that the dose rates per canister are coming down you also have fewer on the pad.

And just to confirm this -- next slide, please -- we took the 18 to 24-month transition that we modeled for this study and ran is through the EPRI cask loader software. And we asked the cask loader software, the person who ran that, to take our discharge burnups from the 18-month base case and for the 24-month higher burnup, load that uniformly into the cask loader software and tell me what you get.

And what happened is what you see on the slide. We have a combined figure of merit. It doesn't give dose rates, but figures of merits are proportional to the dose rates. For the canister system, we are reduced by 8 percent for the 18 to 24-month transition.

A larger reduction in gamma, a little bit of an increase in neutrons, but these are highly gamma dominated systems. And in addition to the 8 percent reduction per canister, we have 16 percent fewer canisters for this particular example.

And so our broad conclusion from this, for our preliminary work, is that there is a smaller beneficial dry storage dose rate change particularly for the normal storage configurations.

And finally, I think my last slide is just the conclusions which we have already talked about these. So I think the best thing to do is just to see if there is any more discussion or questions on what was done and what the results look like.

CHAIR BALLINGER: We visited a vendor a little while back, and we had a long discussion about increased burnup and increased --

MEMBER MARCH-LEUBA: Get closer to the microphone there.

CHAIR BALLINGER: Increased burnup and increased enrichment. And he said that from an economic point of view, they would have to recommend a single batch core.

MR. HALL: And by single batch core --

CHAIR BALLINGER: That's what it means.

You fill it up.

MR. HALL: You mean load the entire core -- from a 157 assembly core --

MEMBER MARCH-LEUBA: If you can let me a 50 percent swap. I mean, why don't we do 30 percent swap in loading. If you cannot make a 50 percent swap, you have to go to a 100. So it is either 33, 50 or 100. It is discrete. You cannot swap 60 percent.

MR. HALL: So I would say that there was a slide that I had with the two curves --

MEMBER MARCH-LEUBA: Mm-hmm.

MR. HALL: -- that were a function of 24-month or 18-month cycles enrichment and what the specific power of the plant was, right? This is why the BWRs have already moved to 24-month cycles.

MEMBER MARCH-LEUBA: Mm-hmm.

MR. HALL: It is because they are lower on the specific power curve. They don't need the enrichments. And so they are staying well below 50 percent and doing 24-month cycles because of where they are on that curve. It is for the high duty PWRs that becomes more difficult.

But what we are showing here, I think if you looked at our 24-month, we were looking at something like -- our batch loads were in the 40s, mid-40s to upper 40s. So we are below 50 percent with 5.4 weight percent, and we do not exceed the burnup limits.

And so core designers do Whac-A-Mole, right, if you remember that game? You got to meet -- you hit one limit, the other one pops up and so this hits all of the limits. It stays within the enrichment limits, within the burnup limits and under 50 percent. And it was -- you know, we assumed in this case 98 percent load factors and very short outages of 20 days so it appears to us at least from the scoping study that it is quite doable.

MR. SMITH: It is Fred from EPRI. From an economic point of view, they would have -- to make it work financially for a customer, they would have to go to a single batch core.

MR. HALL: Fred, you might want to speak to this. You've done this study, right?

MR. SMITH: So the fuel management that we've done in several three dimensional, not just scoping, but full fuel management, demonstrates that we can get below a 50 percent batch fraction.

And if you go much beyond that, the economics of 24-month cycles become very negative. And that's why you don't see -- only 24 percent of the fleet has gone to 24-month cycles because they can't get there with a reasonable batch size.

And so, yes, if you don't keep the batch fraction down, then the fuel costs overshadow the outage savings, and people don't go there. So their statement is probably true, but it is not really applicable, I don't think.

CHAIR BALLINGER: And just to comment, I was surprised.

MR. SMITH: Yeah.

CHAIR BALLINGER: We were all -- most of us were on that visit so.

MEMBER MARCH-LEUBA: That 40 percent you mentioned is with what enrichment?

MR. SMITH: Well for around 6-1/2 or so for a four load PWR.

MEMBER MARCH-LEUBA: A traditional.

MR. HALL: Yes. It is split fee so --

MEMBER MARCH-LEUBA: You need to heat 795. You can do it --

MR. SMITH: No, no, no. Not for -- the 8 percent really is something that --

MEMBER MARCH-LEUBA: Over the fuel --

MR. SMITH: -- would fill up in a BWR peak 10.

MEMBER MARCH-LEUBA: Mm-hmm.

MR. SMITH: Not an average and not in a PWR.

MEMBER MARCH-LEUBA: I have said often, and people always laugh when I say it, I am not in the business of arguing with computer codes. The bottom line here is we have the validated tools to do this calculation. If a plant manager wants to load a 36-month cycle, you go and calculate it, you know, and see what comes out and then with the ching-ching number for the dollars and tell me whether I want to do it or not.

And from the safety point of view here in this building, yeah, I'm an NRR member. All I care is that you will write a license report that says I satisfied Criteria A, B, C and D. And that's all I care, right? And, again, I'm not a member of NRR, and this is my personal opinion. But don't argue with a computer code. Just go ahead and run it and see what comes out and tell me how much it costs.

CHAIR BALLINGER: Questions? Okay. I'm still not sure what's next.

MR. WELLS: We are going to go to 12 now, I think, yup, decay heat, which is on Page 176 of the PDF.

MEMBER MARCH-LEUBA: Could you possibly tell me how many presentations are there left?

MR. WELLS: We're going to probably scratch --

MEMBER MARCH-LEUBA: Is it because there are two ones? Yeah, I'm for that because -- I'm running out of batteries here.

MEMBER REMPE: Can you give us an idea of how long this next presentation will be because --

MR. WELLS: It won't take very long.

MEMBER REMPE: Well, I know that the subcommittee chairman may have some time constraints.

CHAIR BALLINGER: Yeah, no, no. We got it. I have a -- we have a break at 3:45. That's scheduled. And so whatever we get done by 3:45, I will have to leave, but Dave will take over chair. So there is no issue with respect to --

MR. WELLS: I don't think there is any issue in getting the last one done before 3:45.

CHAIR BALLINGER: Okay. All right.

MEMBER REMPE: That's where I was going.

CHAIR BALLINGER: Great. That's good.

MS. AKKURT: Okay.

CHAIR BALLINGER: I mean, you guys are here out of the goodness of your heart. Okay? So it is important that we hear what you have to say.

MS. AKKURT: Okay. So in this presentation, I will give an overview of our ongoing collaboration for extending the decay heat validation range. And I don't need to tell this group that decay heat is really important for the back end, whether it is your spent fuel pool, rate management, as we have been discussing, your dry storage, you know, both in terms of fuel but also canister rate management, you know, it is proportional temperature and decay heat and transportation disposal, you know. You have limits for that and for disposal also, you know, accurate knowledge of decay heat dictates how many canisters you can store in your repository.

So to give you some background on, you know, how this collaboration started, all mentioned about the higher burnup demonstration project and how the measured temperatures came much lower than estimated temperatures using different cause and so on.

So at that time EPRI coordinated three PIRTs in parallel, fuel, thermal modeling and decay heat. And during this PIRT, you know, we became aware of some of the unpublished clad decay heat measurements. But before going to that, you know, the

PIRT report placed on expert's panel is published. And normally PIRT reports including the important parameters ranking and recommendations. But for this one, since we have many members who are interested and so on, it was more like overview of the decay measurements, overview of decay heat calculations, sources of uncertainties. It is a more comprehensive report. That's what I am getting to it.

So the gaps that were identified is, you know, we need decay heat measurements, you know, where we don't have the shorter cooling times here. We are talking about, you know, one to three years and then higher burnups. And then also for advanced fuels, you know, we need some measurements for higher enrichment.

MEMBER MARCH-LEUBA: So I don't ask you the wrong questions, you are worried about long-term spent fuel pool and canister heated load. You are not worried about transient decay heat in each cycle.

MS. AKKURT: That's clear. The presentation, when I refer to decay heat here, we are talking about decay heat beyond one year.

MEMBER MARCH-LEUBA: Okay.

MS. AKKURT: Yeah.

MEMBER MARCH-LEUBA: Because in the short-term in the first second --

MS. AKKURT: Yeah.

MEMBER MARCH-LEUBA: -- lots of things change.

MS. AKKURT: That is not covered under this presentation. That is a completely different thing. We are talking about, you know, longer term beyond one year because that's usually how it is divided. For example, Regulatory Guide 354 is valid for 1 to 100 years, right? And below that is covered under differently.

So this is what we have in terms of published data, the decay heat range, you know, going up to 50 and 50, but there are not many points and also their measurement uncertainty is an issue. And at the time, in terms of cooling time, it was from 2-1/2 years to 27.

` So the existing measurements that were included in the decay heat PIRT were coming from three sources, CLAB, GE-Morris and HEDL. CLAB is the light blue. And yes, you can see they are more focused on lower decay heat, but they have really lower measurement after.

Why are they focused on lower decay heat? CLAB is in Sweden. They don't have dry storage. They mainly rely on, you know, centralized wet storage.

They were doing, you know, decay heat measurements. These are full assembly decay heat measurements using calorimeter for their repository. Because for their repository, decay heat is used in the meter.

And GE-Morris have measurements that are extended, but for some of them the measurement quality is not very good. And in the PIRT report, we recommended explaining those from the validation sets.

Now when it comes to high decay heat, you only have six measurements from HEDL. But the measurement uncertainty is large. I want to say it is about 10 percent. It is because, you know, no uncertainty analysis was done. And then they have published a documents that says it is 10 percent. And since there is not much documentation it is not like you can go back and re-evaluate this.

And during PIRT, we became aware of some unpublished CLAB decay heat measurements. They have measured ones that are on the left. They are the measurements that were conducted in 2003 and 2004, and published in a document that has been used by the global industry, including the regulators, you know.

We will find out about the unpublished decay heat measurements given the fact that the measurement uncertainties are much lower. Those are uncertain events for 2 sigma as opposed to 1 sigma on the left. We wanted to see if we can get a collaborative work initiated to get those published and make them available for the global industry for validation purposes. So, you know, those measurements on the left, with low measurements, the time measurement uncertainty can be excluded from future validation sets.

And we were able to reach an agreement. And we signed an agreement in December 2020, and the agreement includes three tasks. In the previous slide, I said, you know, at the time we became aware of six measurements after signing the agreement, CLAB, ESCP sends us over 150 unpublished decay heat measurements. And, you know, it was really right before Christmas is when they delivered, a gift basically.

And after we signed the agreement, we performed additional measurements. Basically, while performing additional measurements, we targeted the higher burnup shorter cooling time and also we targeted the GE-14 fuel because for any U+, you know, that's one of the candidates. And CLAB has many fuel types. It's like, you know, I mean, in terms of, you know, in terms of, you know, the number of assemblies that are available for your measurement. But for some reason, they didn't do any GE-14 measurements prior to that.

And so these unpublished measurements, you know, will be extending the key heat range substantially and with lower measurement uncertainties. Cooling down time is going to be extended substantially, but to be able to do measurements with, you know, higher decay heat or shorter cooling time, we basically hit the limit for the existing calorimeter CLAB.

So basically a part of our agreement is the calorimeter is now being upgraded so it can enable, you know, affirming measurements. Right now the original calorimeter for which was used to do these measurements, we can go to up to 2 kilowatt. Right now I think we are targeting 4 kilowatt.

And using the calorimeter as part of it, documentation and making the report available, you know, in a publicly available published EPRI report is part of the agreement. And we in parallel doing validation calculations. And we are using two tools, ORIGEN and also specifics SNF code. And those will be publicly available too.

So for those who are not available, again

Sweden relies on centralized wet storage. After the discharge, it stores -- fuel is stored at the site for nine months and then it comes to CLAB. They don't accept any fuel before nine months basically. And they have over 33,000 fuel assemblies already to choose from. Your BWRs, PWRs and, you know, some of the even molten fuel is there, but, you know, they don't have much information on that.

MEMBER MARCH-LEUBA: So you think the assemblies, they transfer fuel bundles on the active cooling? I mean, you put them in a track with active cooling and send them to the facility?

MS. AKKURT: They do the dry storage. And they actually --

MEMBER MARCH-LEUBA: In nine months, they can do dry storage?

MS. AKKURT: Yeah.

MEMBER MARCH-LEUBA: So what's the problem with .2 if you can do it in half a core?

MS. AKKURT: Big transport. Okay. So now this is, you know, giving you a snapshot of, you know, what is included in terms of cooling time versus measured decay heat. And our report when it is published, it is going to include the original SKBs, you know, those decay heats and then it is unpublished measurements, this 166.

And this maybe will be extending, you know, the, you know, hazardous burn range, burnup range cooling time. And the actuals published, that is the first SKB report and unpublished is going to be part of the, you know -- both of them will be part of the EPRI report basically.

And here we are, you know, basically saying that when we start the measurement campaign, when the calorimeter is ready, you know, this is the distribution of number of measurements as a function of cooling time or decay heat. But we will try to, you know, fill some of the gaps because in this middle middle, you know, for longer times, 10 to 24, we have a lot of measurements for PWR and BWR. And we will be targeting more, you know, on shorter cooling time and higher decay heats.

So in terms of the uncertainties, you know uncertainty in the original report was evaluated using the repeat measurements and components but uncertainty is being re-evaluated -- actually, Bob Hall took the uncertainty evaluation portion. To do that, we have to look at the components of it. As I said, these are full assembly measurements.

So, you know, basically, you have the

heated measurements and looking at the heat upgrade, for which you know the power and then you do the full assembly measurements and look at the heat upgrade but you know that.

So for the uncertainty, you have different components, and the key ones are your calibration measurements using the heated and then also gamma leakage. And ETL uncertainty valuation has been done. And these are the different components of the uncertainty, whether it is your heater and your fuel assembly. Heater and fuel assembly have, you know, some of the components, you know, some of the same components, but also some of the difference. Obviously, when you do the fuel assembly measurements, you are also having common energy loss and so on.

All of those are taken into account and shown that the measurement uncertainty is actually lower than what was quoted in the original document for 05, which is really commonly used around the world. And this measurement uncertainty evaluation has been independently reviewed by some experts and will be reviewed as part of the extended storage collaboration group in OECD-NEI working group. But, as I said, another way of checking, we have a number of repeat and a number of system emission ones. They are on the semantic locations, have similar histories, burnups and so on.

So when we look at the uncertainty ones and, you know, how the repeated system measurements are responding to that, they are all -- it's in the bands with two exceptions. That's in your statistics anyway. So I don't have, you know, statistically significant outliers basically.

So coming back to, the last is what we have in the published domain right now. When we have the EPRI report fully published, it will be used for validation. In the EPRI report, we will be making some recommendations for the validation set, and we are going to recommend removing HEDL measurements, which have very large uncertainties, from all the future validation sets for the GE-Morris, you know, some of the probability ones were already recommended to be removed. But, you know, if -- I think for the lower decay heat, they have lower measurements. They can be kept or if you want to use the entire set like it's done in cross-counting, you can use some if there is uncertainty debating or something, you know.

And this will, you know, obviously, it was not done for any U+ field, but I think by extending the validation range, you know, and with better quality measurements it will benefit because those uncertainty bands if you go to higher decay heat, they will be increasing significantly.

So in summary, the one benefit of PIRT was, besides, you know, identification of, you know, gaps in some, becoming aware of these CLAB measurements and the follow-up EPRI and ESCP cooperation, and we will be publishing the measurement report this year and validation report this year.

Calorimeter upgrade is ongoing. Due to supply chain issues, we had some delays. At the moment, we are planning on starting the new measurement campaign next year. And those new measurements will also be published in a publicly available form.

However, before publishing those, through ESCP decay heat task force, we are planning some blank benchmarking. With that, questions?

MEMBER MARCH-LEUBA: The issue I have asked at the beginning of the presentation, what is it that we want to do? What's our goal?

Let me give you multiple choice. A, validate ORIGEN so I can use ORIGEN to calculate the decay heat of my particular fuel element, B, validate a correlation similar to ANSI, validate the ANSI decay heat correlation so I can continue to use it or C, make a better correlation that maybe includes more parameters that I can use for making decisions.

Having a calorimeter is good, but you should know what you want to use it for, right?

MS. AKKURT: Well, you know, now when they do the cask loading, right, in the past Regulatory Guide 354, Rev. 1, was being used, right, which is very conservative. And, you know, in fact one of our members were challenged by the regulators saying that we are using task force, which is using extended Regulatory Guide 354 and how do you know it is conservative?

At that time, it is a survey comparing to ORIGEN, you know, basically showing that it is conservative. But you also find out that it is significant because in a sense that it was all estimating 10 to 55 percent depending on the closing parameters?

MEMBER MARCH-LEUBA: How much?

MS. AKKURT: 10 to 55 percent, Regulatory Guide 5.

MEMBER MARCH-LEUBA: 65?

MS. AKKURT: 10 to 55 percent.

MEMBER MARCH-LEUBA: In some cases, what

you resolve by a factor of 55 percent.

MS. AKKURT: Yeah. So, but now --

MEMBER MARCH-LEUBA: It is ORIGEN calculation?

MS. AKKURT: No, no. This is Regulatory Guide 5 --

MEMBER MARCH-LEUBA: Oh.

MS. AKKURT: -- for origin of ORIGEN. So now task force, which is also -- it is usually used by utilities according to loading campaign, it uses ORIGEN, right? So now, you know, in terms of calculation component, you are using a better tool that, you know, predicts decay heat more accurately, right?

But even if you do the calculation, you know, you still need to take into account measurement uncertainty for your final loading. This piece is -- okay, in terms of calculation, yeah, we have better tools. You can use them, and we can predict this, you know, within future percentages or better than that. But still we have to take penalty for measurement uncertainty.

MEMBER MARCH-LEUBA: All right. So the goal of this exercise is to calculate or apply or determine which uncertainty you apply to the calculations when you load the dry canister. So your goal was to determine uncertainty. Therefore, using that very robotic experiment that you have in there makes a lot of sense.

MS. AKKURT: Yeah. So now we have measurements with better, you know, load --

MEMBER MARCH-LEUBA: (Simultaneous speaking) uncertainty.

MS. AKKURT: Total uncertainty because it has different components, right, you know? Because if this is my validation set, and my validation set has very -- my measurement uncertainty, I have --

MEMBER MARCH-LEUBA: Your calculation uncertainty cannot be smaller than the measurement certainty. You start with a measurement and then you add. So you are taking the main component of your uncertainty by spending some money, doing some intelligent analysis and collecting more data. That is very applaudable. I mean very good.

MS. AKKURT: Thank you. Questions?

CHAIR BALLINGER: No questions? Again, what's next?

MR. WELLS: So I'm going to say we take a break here.

MS. AKKURT: Take a break, yes.

MR. WELLS: There were three more higher level summary presentations that we had prepared -- well, there were three more that were higher level summary type. Maybe we can talk, David, on the break on which one of those, or any of them. We have gone through all the very detailed technical presentations at this point.

CHAIR BALLINGER: I'm sure I speak for the other members, and they can speak for themselves. But, again, the presentations that we have had were to my mind very instructive.

MEMBER MARCH-LEUBA: And very detailed.

CHAIR BALLINGER: And very detailed. And will become useful for us, which is what was one of our goals going forward, especially on the burnup, increased enrichment burnup in the XLPR, that kind of --

MEMBER MARCH-LEUBA: Especially --

CHAIR BALLINGER: -- discussion because we hadn't seen that before, and we expect that we will.

MEMBER MARCH-LEUBA: May I suggest that we take public comments now in case we convince Dave to quit early after you are gone?

CHAIR BALLINGER: I mean, I'm easily convinced. Okay. So are there any members of the public that would like to make a comment? If there are, can you state your name and make your comment, please? Okay. Hearing none, we can take a break and then you can negotiate --

MEMBER REMPE: After you leave.

CHAIR BALLINGER: -- what you do.

(Whereupon, the above-entitled matter went off the record at 3:37 p.m. and resumed at 3:55 p.m.)

MEMBER PETTI: Okay. We're back. This is Dave Petti who is filling in for Member Ballinger. And we are going to hear one more presentation on the Collaborative Research on Advanced Fuel Technologies known as CRAFT.

MR. MUFTUOGLU: Yes.

MEMBER PETTI: Go ahead.

MR. MUFTUOGLU: Yes. My name is Kurshad Muftuoglu. I am technical executive with the Fuel Reliability Program at EPRI. And today I am going to talk about the CRAFT. And it stands for, as you said, Collaborative Research on Advanced Fuel Technologies for LWRs, PWRs and BWRs.

Next slide. And then we can move one more. So CRAFT takes its mandate from the NEI working group and the task force. The purpose of the CRAFT is to foster a research and collaboration environment while we bring the various subject matter experts from stakeholder organizations together and collect the resources on research and optimize the research and development efforts by bringing everybody on the same table.

It emulates the Extended Storage Collaboration Program, ESCP. Today we had a presentation on that one, however, due to time restrictions, we will not get on that one. But it has a steering committee. And under that, we have technical expert groups that collaborate on various topics.

We can move to the next slide. So basically the objectives, the main objective is to bring subject matter experts from particular U.S. organizations and as appropriate, international organizations together, and these are the stakeholders basically working on the high burnup, high enrichment area and advanced fuel technologies areas. And another objective is to identify both short and long-term options and recommendations to support the highest priority associates to licensed methodologies and ultimately put them to use in the U.S.

And to do that we support gap analysis and the PIRT process. And ultimately the work products are communicated by synthesizing the research results from technical basis and distribute them as targeted deliverables.

Basically, we bring to the table under EPRI's coordination DOE National Labs. The NRC is also a participant. The fuel vendors are participant, and the utilities are also participants of CRAFT.

Next slide. So the main technical focus until recently has been on the advanced fuel technology deployment and particularly the fuel fragmentation relocation and the dispersal topic that is an important aspect of moving to higher burnup. It has been the main focus of CRAFT until now.

And in order to address those under the steering committee, two technical expert groups were formulated, one of whom is the General Guidance and Analysis Committee Technical Experts Group, GGA and the other one is Fuel Performance and Testing, FPT, Technical Experts Committee.

Move to the next slide. By using an issue tracking matrix, we have identified the important research areas, and they have focused on enabling the research in their organizations and incorporating with others.

So this slide shows the overall CRAFT

sector and the key stakeholder interfaces. As I have indicated, we work under NEI working group priorities, set of priorities, licensing safety analysis. The Safety Benefits Task Force identifies the major priorities for the U.S. industry moving to higher burnup, higher enrichment and advanced fuel technologies. And on as needed basis, we collaborate with the U.S. DOE programs advanced fuel campaign, light-water reactors, sustainable program, nuclear energy analysis and modeling simulation, advanced modeling simulation means program.

And in a true memorandum of understanding, NRC also participates in the CRAFT organization. They have representations both from NRR and the research branch. And on as needed basis, they can weigh in, but typically they can also recuse themselves. And they do not do any regulation during these regulatory activities during these deliberations.

They are an important and a very valuable interface to have, the NRC, so that they can be kept up-to-date with the research that supports the regulatory activities as well.

Under the steering committee, the steering committee has membership from, as I mentioned earlier, from vendors, fuel vendors, fuel suppliers, utilities.

EPRI is leading DOE and National Labs and Oak Ridge and INL. And under that, we have the General Guidance and Analysis and Fuel Performance Testing Technical Expert Groups.

Just recently, we have also formed a new technical expert group that started to work on time and temperature area.

Moving on to the next slide.

MEMBER REMPE: I'm going to stop you there.

MR. MUFTUOGLU: Sure.

MEMBER REMPE: Again, this isn't a safety issue, but I'm just puzzled in knowing what I know about how OECD projects work and how you've got to sign agreements and you can't distribute what you've learned from OECD project to other folks who aren't part of the project, I'm just kind of puzzled about this organizational structure and how that would work.

Were you getting input from the OECD projects or the other international projects? It just seems a little different than what I've seen in other things. And who is in charge of this -- is this an EPRI run program?

MR. MUFTUOGLU: So EPRI coordinates it. We are running the program. And the stakeholders that

I have mentioned, industry stakeholders, DOE, NRC and fuel vendors, they are participating on their willingness voluntarily.

So we have a charter that sets up the rules of how we are going to run things, how we are going to generate consensus. And it is all research, technical research oriented. It's all on technical issues. There is nothing proprietary. Everybody understands that.

MEMBER REMPE: So give me an example like this. A representative from OECD-NEA who may be involved in other OECD projects --

MR. MUFTUOGLU: Right.

MEMBER REMPE: -- comes and they just review the things? Are they giving input from those other projects to it?

MR. MUFTUOGLU: That interface with the international participants is there, but we don't have any international focus so far. It has been only in the U.S. focus. And we did not have any conflict or any of these questions never came up until now.

MEMBER REMPE: Okay. So no one from OECD is coming in?

MS. AKKURT: Can I just jump in here? So for, you know, those programs, OECD programs, they have their own membership, like agreements and limitations and so on as you are saying. And then, you know, EPRI has similar things as well that are available with our members and everything.

So we get those representatives to give an overview of their programs in our meetings. So that the whole community attending the CRAFT is aware of those OECD programs. They are not necessarily sharing the reports or details.

MEMBER REMPE: Right.

MS. AKKURT: But they provide those overviews so that -- and then we kind of compare it to what else is going on. Like we have a -- I mean, this is kind of a collaboration on what is going on in that particular issue everywhere in the world basically.

MEMBER REMPE: Okay.

MS. AKKURT: But also it will depend on, you know, how the coordination and activities and roles and responsibilities are divided as Erich mentioned, you know, on the storage collaboration program.

(Operator speaking.)

MS. AKKURT: We don't do --

(Operator speaking.)

MS. AKKURT: -- a national focus. You

have participation from 23 countries. But the rule of the game is if you want to get results, you need to provide results. What this IP is your design and so on. But for the results to be combined and shared or shared, you know, reviewed, everyone needs to bring something to the table.

MEMBER REMPE: I get it if it was a particular country. But I'm just thinking about these memos of agreement are assigned by folks for OECD projects. And I'm wondering how that can happen. But if it's at a high level, there are certain things that are approved by the management board of the individual PRG of the individual projects. So I can understand how this (simultaneous speaking).

MS. AKKURT: In OECD, NEI, and many other, all those kinds of organizations participate, you know, and those we have in other countries. Yeah. ESCP is more intended -- it becomes more international focus. We started in the U.S. but now we have representation from 23 countries, from various social organizations and utilities, vendors, regulators, you know, or others, as such.

MEMBER REMPE: Thank you.

MR. MUFTUOGLU: So the idea is to generate and provide a collaborative research environment.

When we hold our yearly meetings they also come and present their high level studies.

Next slide, thank you. And on this one, as I have indicated earlier -- just ignore that? Okay.

The focus has been on the fuel fragmentation and relocation and the dispersal and the issue side of the matrix item stands for that identified particular aspects of it, particular tasks underneath that needed to be looked at, and these were to be divided up between the two technical expert groups, including the EPTA and GGA technical expert group.

Fuel fragmentation portion, which takes more testing into account, had more activities under FPT tag. Dispersal, however, on the other hand, is more -- the approach is more ineligible. And so the GGA tag focused on that one. The fuel location is shared between the two of them.

I'm not going to go into the details of the activities itself so we can move to the next slide. Some of the recent developments, last year we had spoken to the development of the DOE-AFC advanced fuel campaign LOCA plan that was developed by the National Labs and DOE. It is an important piece of work, which provides a combined integral and semi-integral look at this plan. And it covers many aspects of the fuel fragmentation and the dispersal topic and also identified the research gaps under that. And CRAFT provided comments on the report and facilitated the comment resolution. And it was issued last September and now the work by the National Labs are initiated according to this plan.

Next slide. Another development, as I have mentioned earlier, is the time and temperature criteria. I will get back to that. Another area we have focused was after the NRC's 2021 13 reel was issued. A look at the data assessment and the data needs on the technical panel assessment was performed by EPRI. CRAFT took that report and performed an official review of that published white paper. Now we are working on advancing the comments generated by CRAFT community on the white paper.

The time and temperature criteria, it is a new technical expert group that has been generated.

Next slide. So it is going to be a focus area for this year particularly. Just briefly speaking, time and temperature is a post-BRB and post dry-out conditions where fuel can survive getting the past critical heat flux values. So it has potential for power-up rates as well as the fuel utilization benefits were recognized by the industry and prioritized.

And we have put together a new technical expert group to develop a material testing plan. And that testing plan will include the definition of the testing facility, define the testing protocol and identify the materials, both irradiated and fresh fuel coated, uncoated, et cetera.

That brings me to the last slide, the summary of CRAFT's overall provider forum for various stakeholders by bringing them on the same table into research collaborative environment. And we focus on the issues that are relevant to the deployment of the advanced fuel technologies and also particularly focus on the plant safety and operational flexible improvements including power upgrades and extended cycles and basically we try to keep aligned with the industry needs, working closely with the NEI safety benefits and the working group on the ATF. And that is my last slide. Thank you.

MEMBER PETTI: Questions, members, consultants online? Okay. I don't see any. I want to thank you for coming. Very informative. We covered most of the waterfront, I think, a whole swath of things. So that's a pretty typical place to ask so it is good to hear from you guys on that so.

MEMBER REMPE: Even though we had public comments a few minutes ago, there has been another presentation. Maybe you ought to give it a run again.

MEMBER PETTI: Okay. Any member of the public that has a comment, unmute yourself, state your name and your comment. I'm not hearing anything then I --

MEMBER MARCH-LEUBA: There may be somebody on the chat.

MEMBER REMPE: It is probably is from a while ago or something.

MEMBER PETTI: Yeah. Okay. Then with that, I call the meeting to a close. We are recessed. Thank you.

MEMBER DIMITRIJEVIC: Thank you. Safe travels.

(Whereupon, the above-entitled matter went off the record at 4:14 p.m.)

Fuels and Chemistry

Updates on Topics of Interest to ACRS

EPRI Fuels and Chemistry Department Staff
Dan WELLS, PhD – Director, Fuels and Chemistry

Advisory Committee on Reactor Safeguards: Fuels, Materials, and Structures Subcommittee Meeting
18 May 2023
Washington, DC, USA

EPRI Nuclear Fuels and Chemistry

Research and development supporting maintenance of the primary containment boundaries, optimized and efficient operation, cost effective waste disposal and protection of workers and the public

Chemistry & Radiation Safety

Operating Nuclear Fuels

Used Fuel / High Level Waste

Decommissioning (Supp)

Fuels & Chemistry Organization Chart

Dan WELLS
Director
Fuels & Chemistry

Fuels & Chemistry Technical

Lisa EDWARDS
Rodolfo VAGHETTO

Fuels & Chemistry Support

Kathy DANICH
Amanda HISLOP
Merlyn JOHN

Aylin KUCUK
Program Manager
Nuclear Fuels

Kurshad MUFTUOGLU
Fred SMITH
Suresh YAGNIK, NFIR Lead

Reliability

Dennis HUSSEY
John BEALE

Advanced Fuels

Ruwan RATNAYAKE
Prin. Project Manager

Mohammed ABDOELATEF
Brenden MERVIN
Ken YUEH

Lisa EDWARDS (acting)
Program Manager
Used Fuel & High-Level Waste

Hatice AKKURT
Shannon CHU
Bob HALL
Laura McMANNIMAN, *starting 7/31/2023*
Keith WALDROP

Michelle MURA
Program Manager
Chemistry, Decommissioning & Rad. Safety

Paul FRATTINI
Keith FRUZZETTI
Dave PERKINS

Radiation Safety

Karen KIM-STEVENS
Prin. Project Manager

Borja BRAVO
Darcy CAMPBELL
Nicole LYNCH
Rich McGRATH
Lane WADDELL

Chemistry

Michelle MURA (acting)
Prin. Project Manager

Matt BRANHAM
Iain DUNCANSON
Jennifer JARVIS
Joel McELRATH

Decommissioning

Martin BRANDAUER
Rick REID

Advisory Committee on Reactor Safeguards – Agenda (1/2)

Fuels, Materials, and Structures Subcommittee

18 May 2023 (AM)

Item	Topic	Presenter(s)	Time (ET)
1	Opening Remarks and Objectives	Prof. Ballinger, ACRS	8:30 – 8:35 a.m.
2	EPRI Opening Remarks	Dan Wells, EPRI	8:35 – 8:40 a.m.
3	KOH for PWR RCS pH Control: Radiological Impacts	David Perkins, EPRI	8:40 – 9:05 a.m.
4	EPRI Fuel Reliability Program Overview	Aylin Kucuk, EPRI	9:05 – 9:35 a.m.
5	xLPR Methodology – Probabilistic Fracture Mechanics for PWR Piping	Nathan Glunt, EPRI Marcus Burkardt, Dominion Engineering	9:40 – 10:20 a.m.
	Break		10:20 – 10:35 a.m.
6	EPRI Alternative Licencing Strategy – New Approach to Address FFRD	Fred Smith, EPRI	10:35 – 11:10 a.m.
7	Fuel Fragmentation Threshold	Suresh Yagnik, EPRI	11:10 – 11:35 a.m.
8	Atomistic Modeling of Cladding Coating Behavior	Erich Wimmer, MDI	11:35 – 12:00 p.m.
	Lunch		12:00 – 1:30 p.m.

Advisory Committee on Reactor Safeguards – Agenda (2/2)

Fuels, Materials, and Structures Subcommittee

18 May 2023

Item	Topic	Presenter(s)	Time (ET)
9	Collaborative Research on Advanced Fuel Technologies (CRAFT)	Kurshad Muftuoglu, EPRI	1:30 – 1:45 p.m.
10	EPRI Used Fuel High Level Waste Program Overview	Bob Hall, EPRI	1:45 – 2:00 p.m.
11	SFP Criticality for ATF/HE/HBU: Depletion Uncertainty and Criticality Code Validation	Hatice Akkurt and Bob Hall, EPRI	2:00 – 2:35 p.m.
12	Decay Heat: EPRI-SKB Collaboration for Extending Validation Range	Hatice Akkurt, EPRI	2:35 – 3:00 p.m.
13	Scoping Analysis for Decay Heat and Radiation Dose for ATF/HE/HBU	Bob Hall, EPRI	3:00 – 3:30 p.m.
14	Extended Storage Collaboration Program (ESCP)	Hatice Akkurt, EPRI	3:30 – 3:45 p.m.
	Break		3:45 – 4:00 p.m.
15	Open Discussion	All	4:00 – 4:30 p.m.
16	Committee Discussions	Prof. Ballinger	4:30 – 4:40 p.m.
17	**Adjourn**	All	4:40 p.m.

KOH Update

Radiation Protection, and Radioactive Waste Update

David Perkins
Technical Executive, Senior

KOH Project Manager
Keith Fruzzetti, PhD
Technical Executive, Senior
May 2023

Potassium Hydroxide: Why?

- **<u>Eliminate</u> vulnerability to enriched Li-7 supply**
  - Limited production (China and Russia)
  - Increased demand (flexible operation, new PWRs, molten salt reactors)
- **Significantly reduced operational cost**
  - Estimated savings per year of ~$100k/unit (2016 estimate)
- **May be more beneficial for fuel**
  - Potentially lower corrosion
  - Potential mitigation strategy for Crud Induced Power Shift (CIPS)

House Committee on
Science, Space, & Technology
Eddie Bernice Johnson
Ranking Member

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Press Releases
In The News
Letters

GAO Raises Questions about Adequate Supply of Lithium-7 for Nuclear Power Reactors

October 09, 2013

Nuclear Energy

Supply Chain Deep Dive Assessment

February 24, 2022

U.S. Department of Energy Response to Executive Order 14017, "America's Supply Chains"

<u>Most Important Supply Chain Issues</u>

Current Large Reactors: Enriched Lithium (#2)

Advanced Reactors: Lithium/Enriched Lithium (#4/#5)

Significant value with KOH. Successfully used in VVERs for Decades.

EPRI

KOH for Western-design PWRs: Generic Testing & Assessments

SCOPE

Materials	Fuels	Chemistry	Radiation Safety
• Initiation & CGR Testing • Non-irradiated • Stainless Steel (SS), Alloy 600, Alloy A-286 and Alloy 182 (CGR) • Irradiated • Stainless Steel	• Vendor assessments • Experimental Loop testing • Experimental Autoclave testing	• System review and impacts • High temperature chemistry (MULTEQ) • Purity specifications • Multiple alkali modeling and control	• Activation species and dose pathways • Impact on plant radiation fields • Effluent and radioactive waste handling

STATUS

- All testing completed with one exception.
- SS in crevice chemistry **expected to complete in second-half of 2023**.

- Vendor assessments and both planned testing programs completed.
- Further WALT Loop testing on-going.

- Assessments completed
- NSSS Vendor reviews completed
- VVER experience leveraged
- *Station is working through the 10 CFR 50.59 review process to support implementation*

Needed Testing and Assessments Expected to Complete in Second-half of 2023

KOH For Western Style Pressurized Water Reactors

Chemistry and Radiation Safety Overview

Needed materials and fuels testing and assessments are expected to complete in the second half of 2023

Chemistry

- System review and impacts
- High temperature chemistry (MULTEQ)
- Purity specifications
- Multiple alkali modeling and control

Radiation Safety

- Activation species and dose pathways
- Impact on plant radiation fields
- Effluent and radioactive waste handling

Chemistry and Radiation Safety Big Picture

- Assessments completed
- NSSS Vendor reviews completed
- VVER experience leveraged

Chemistry and Radiation Safety Scope

Chemistry Control — CVCS Bed Operation with K and Li

- Completed. Entirely feasible and the VVER experience was very helpful. (3002010650)

Radio-isotopic Generation — Evaluate KOH Chemical and Sodium Impurity

- Completed. No significant issues. (3002015902)

NSSS Vendor Reviews — Westinghouse and Framatome Assessment on Potential Primary System Impacts

- Phase 1 completed. (3002018427 and 3002018429)
- Phase 2 with Westinghouse completed, to address identified gaps from Phase 1. (3002020959)

MULTEQ CW Tools — High Temperature Chemistry Thermodynamics

- Several important potassium species added in V9 (e.g., KOH, $KB(OH)_4$, KCl). Additional species to be added in V10.
- pH Calculator updated in ChemWorks Tools v4.3 (3002016775)

VVER Experience — Literature Data and Operating Experience

- Significant literature data gathered and assessed.
- VVER operating experience assessed. Supports monitoring plan for demonstration.

Plant Demonstration — Support and Assessment (3 cycles)

- Identify and work with KOH demonstration unit.

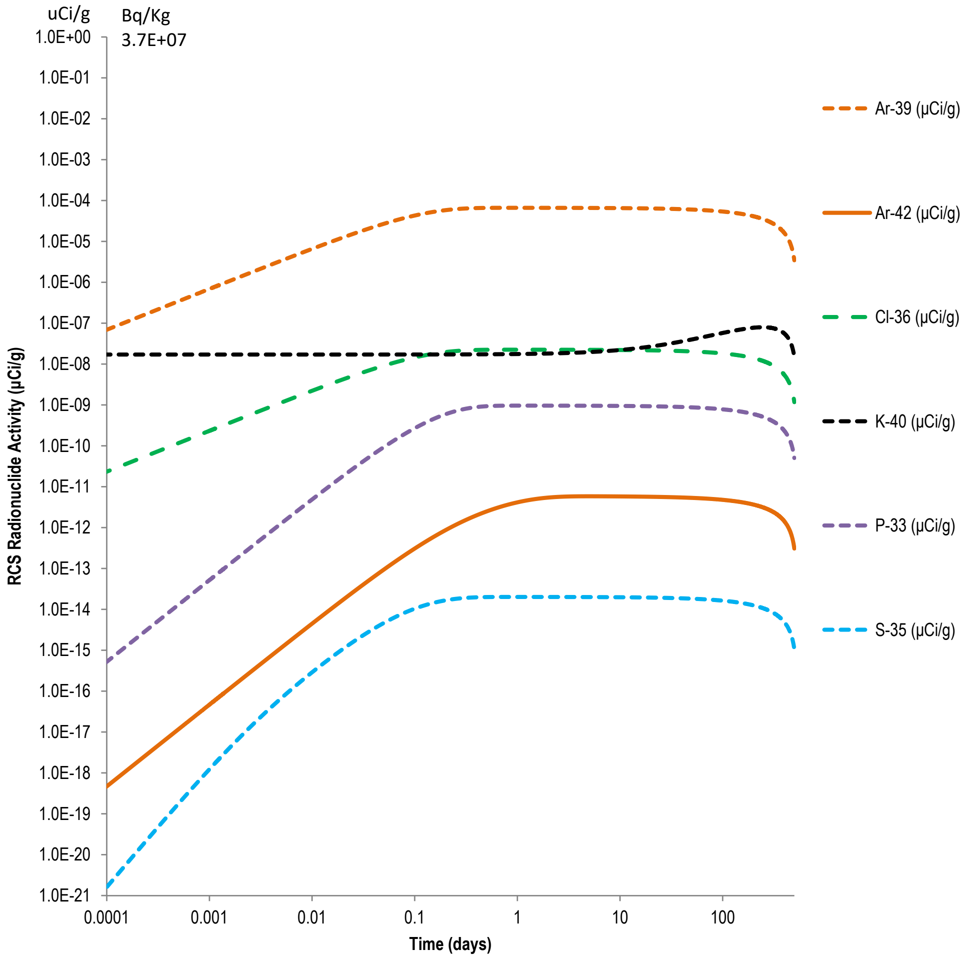
Chemistry and Radiation Safety results indicates no significant challenges.

Chemistry Control

- Mixed alkali chemistry (K and Li) has comparable pH_T values to Li-only on an equivalent molal basis (based on analysis using MULTEQ)
  - In the limit of K-only control, the maximum pH_T deviation (i.e., to same molal Li) is 0.03 units
  - Although K binds more strongly than Li to cation resin, exchange is one-for-one (ion-to-ion)—having an equivalent effect on pH_T
  - Use an "equivalent lithium" approach for pH Control
- Additional chemistry monitoring
  - K, Na
- Chemical Volume and Control System (CVCS) resin management being addressed
- Room temperature conductivity is higher with potassium present
  - Important if using conductivity to monitor for impurities

Chemistry Control with K and Li is Equivalent to Li-only

KOH and Radionuclide Generation



- Six have half-lives (t½) greater than 24 hours (Ar-39, Ar-42, Cl-36, K-40, P-33, and S-35)
  - Maximum total coolant activity from these is predicted to be < 3.7 Bq/g (1E-04 μCi/g)
  - Total estimated activity of these six radionuclides is less than 0.1% of the total RCS Activity.
- From the Radiation Protection perspective, K-40 can present unique challenges.
  - Considered a natural isotope
  - Potential impacts / issues raised:
    - Whole body counting and other release monitors
    - Effluents
    - Waste disposal
    - Dose impacts

KOH Radiation Safety White Paper

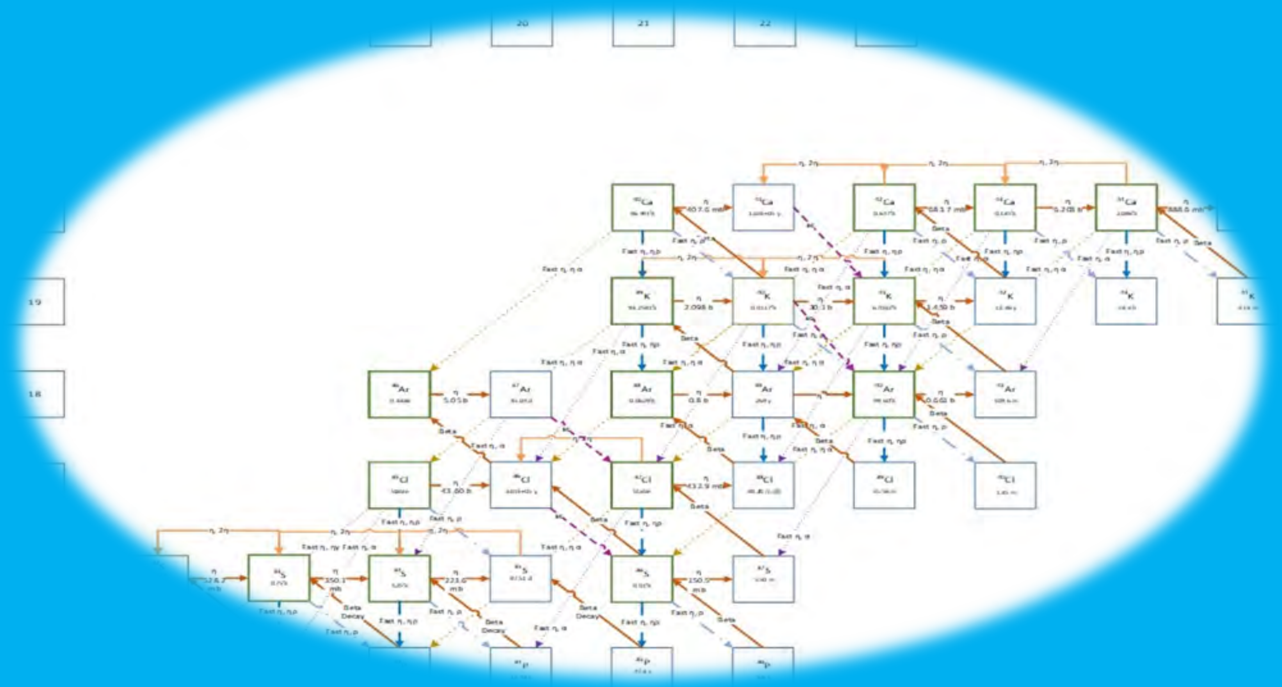
EPRI R&D performed a comprehensive review and assessment of chemistry and Radiation Safety factors (3002015902)
Plant-specific application items identified by the demonstration Plant Radiation Protection staff are being addressed with EPRI support.

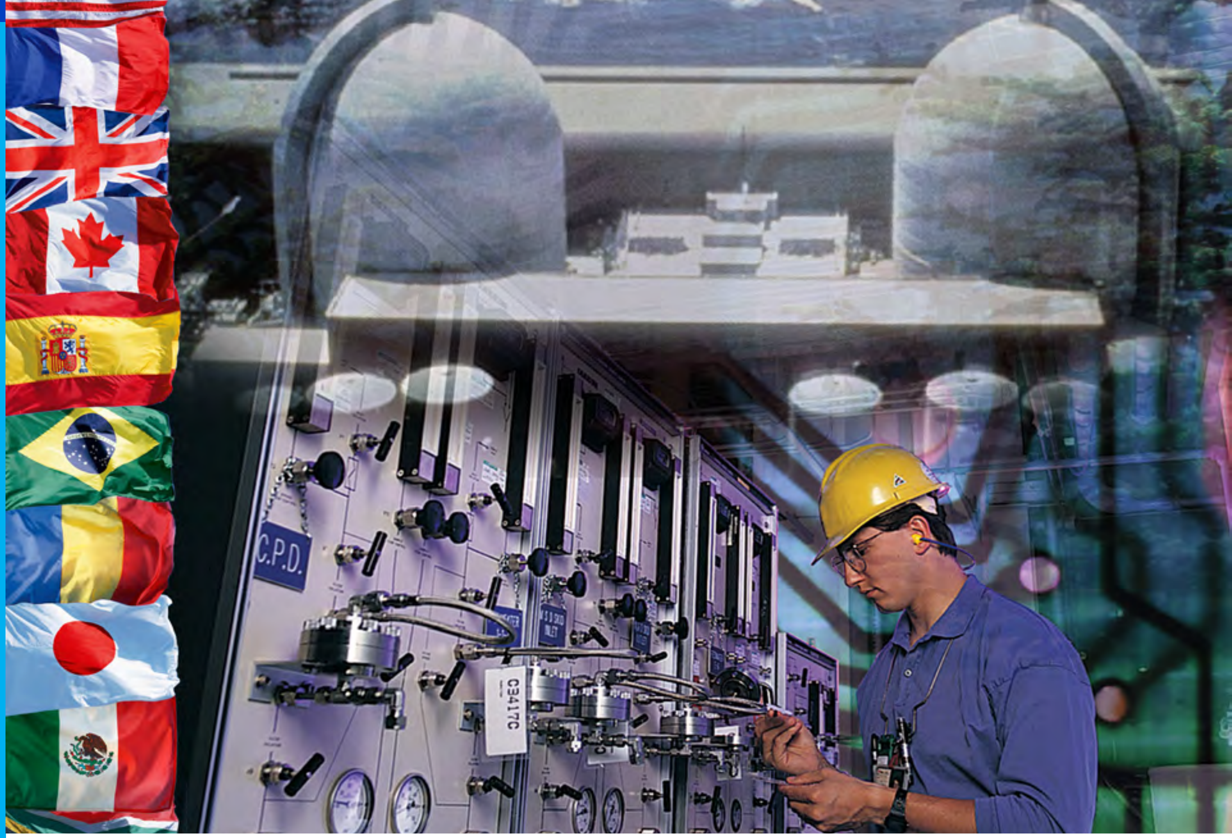
- Nuclides of Concern (Completed)
- Passive Monitoring (In progress)
- Declared Worker
- Radioactive Waste
- Dose Rate Impacts (In progress)
- Submersion Dose (In progress)
- Instrument Responses (In progress)
- Contamination Surveys
- Staffing Concerns
- Personnel Monitors (In progress)
- Air Sampling (In progress)
- Equipment Monitor
- Whole Body Counting (In progress)
- Lens of Eye
- Skin Dose (In progress)

Completed   In progress

KOH Radiation Safety White Paper

Radionuclides of Concern Review

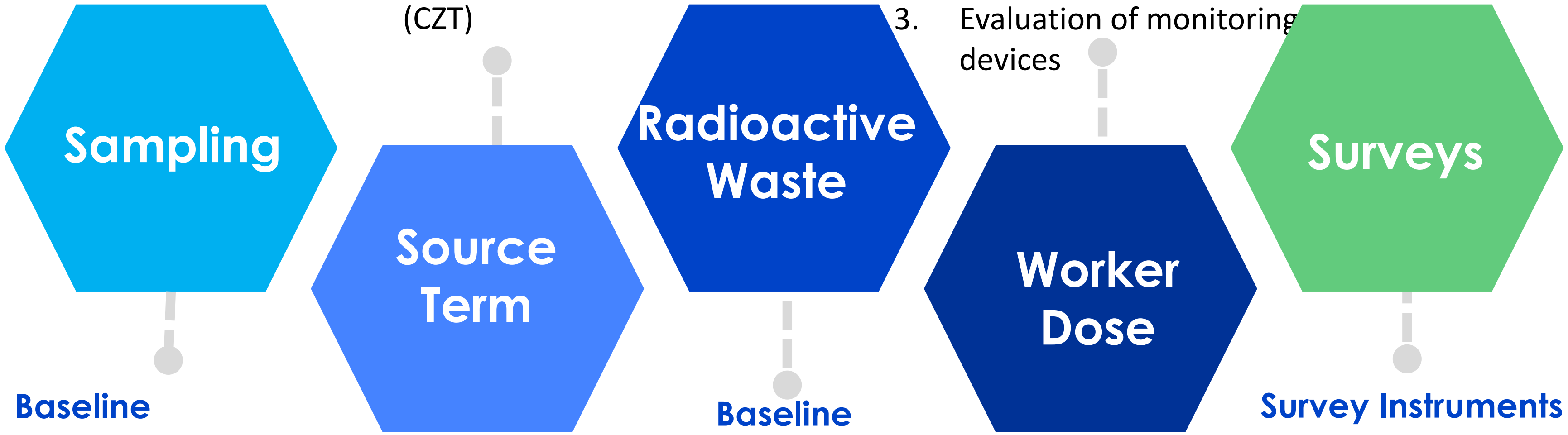




- Verify that gamma spectroscopy systems libraries have the appropriate radioisotopes
- Prior to KOH addition, establish baseline levels of these radionuclides in reactor coolant and support systems, reactor cleanup resins and filters, effluents, and radioactive waste.
- Analyze reactor coolant, cleanup systems, and the waste and effluent streams that are generated in the beginning, middle, and end of the cycles.

KOH Radiation Safety White Paper Update

Monitoring – baseline and demonstration



Baseline

1. Plant sample program
2. CMA
3. Gamma Isotopic
4. Hard to detects

Baseline

1. Baseline characterization (CZT)

Baseline

1. 10 CFR Part 61 baseline
2. Dry active waste

Dose

1. Personal contamination events,
2. Radionuclide distribution, including internal contamination events, if any
3. Evaluation of monitoring devices

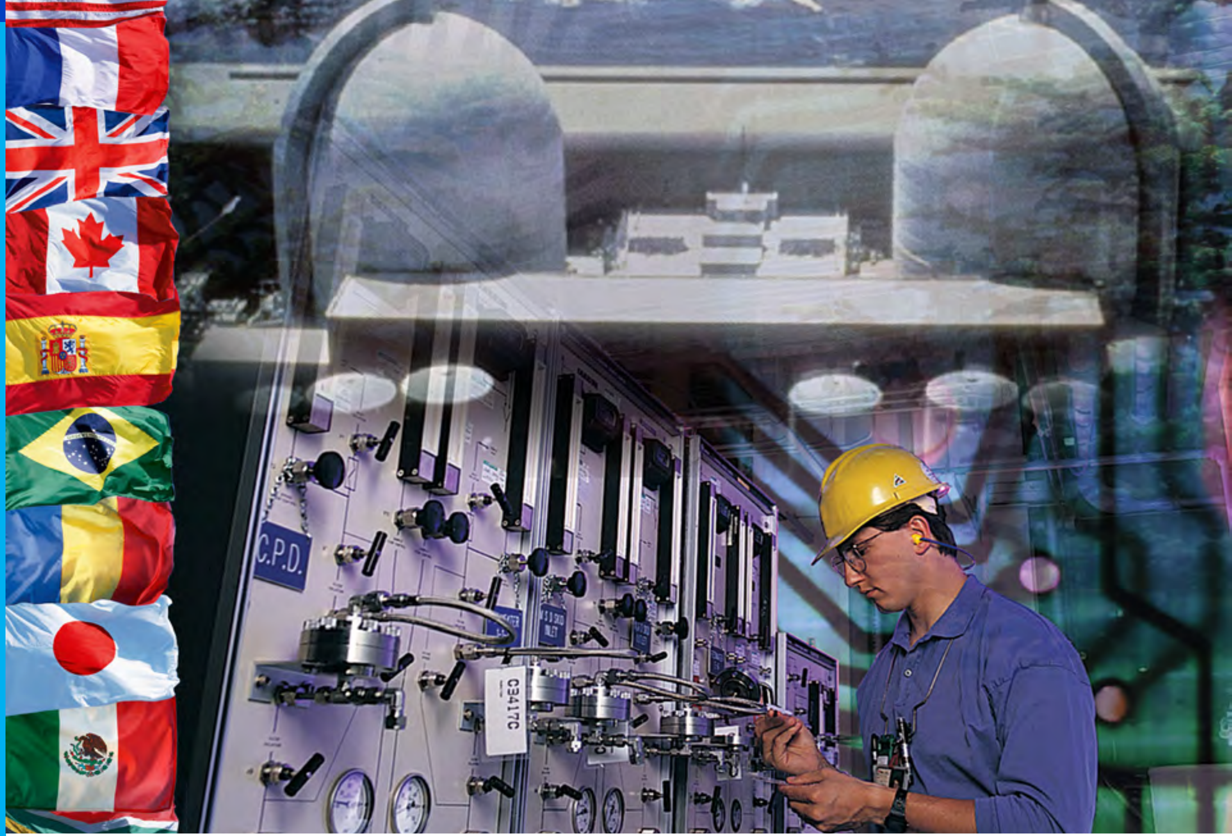
Survey Instruments

1. Review and evaluation of instrumentation sensitivities and responses
2. Review of areas before and after initiation of KOH injection

*LILW: Low and Intermediate Level Waste

KOH Radiation Safety White Paper

Instrument Verification and Preparations



- Determine plant systems or areas that may be subject to having only pure beta emitters and establish process controls for these areas to ensure appropriate monitoring is conducted.
- Establish procedure controls to use a GM detector or other large area proportional counter to perform a survey of items from these areas for unconditional release.
- Evaluate if the tool equipment monitor setpoints should be adjusted to account for non-gamma emitting isotopes.

Station Radiation Safety KOH White Paper Summary

Sequoyah station radiation protection staff have ongoing work activities and addressing the KOH White Paper activities and moving through the different area.

Gamma Spectroscopy System Updates

- Whole body counter updates completed
- Gamma spectroscopy system update in progress

Dose and Effluent System

- DAC values are reviewed and updated for the dose management system.
- HIS-20 updates in progress

Radiation Fields

- CZT data collected from outage and under review
- Baseline chemistry data collection in progress

Bioassay Program

Working with vendor to ensure offsite bioassay program is aligned with the potential new radionuclides

Plant Specific Activities with TVA/Sequoyah

Timeline

2021 x **2022** **2023** x **2024** x **2025**

Engineering Scoping Document

10CFR50.59

Engineering Change Package

Final fuels and materials testing to be provided

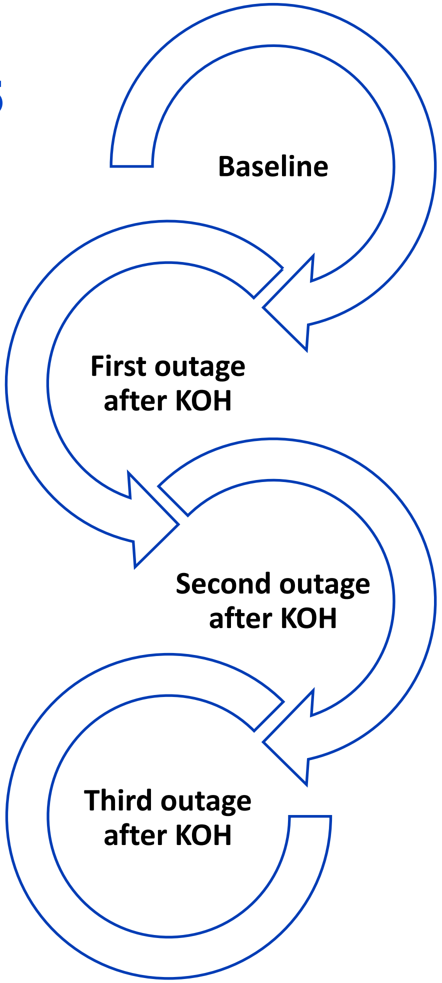
Have been holding regularly scheduled webcasts with TVA staff, with in-person meetings beginning May 2023

Chemistry

- CVCS bed operation and management
- RCS chemistry monitoring and analysis gaps

Radiation Protection

- Generation of new radionuclides
  - Monitoring, assessment and analysis gaps
  - Review / update of procedures



Fall 2024

Spring 2026

Fall 2027

Spring 2029

‡ Fuel duty bounds ~ 50–60% of fleet

‡ Mixed core started Spring 2023 (**U2**)

Together…Shaping the Future of Energy®

EPRI Fuel Reliability Program

Overview

Aylin Kucuk
Program Manager, Nuclear Fuels, EPRI

U.S. NRC ACRS Fuels, Materials, and Structures Subcommittee Meeting
Bethesda, MD
May 18, 2023

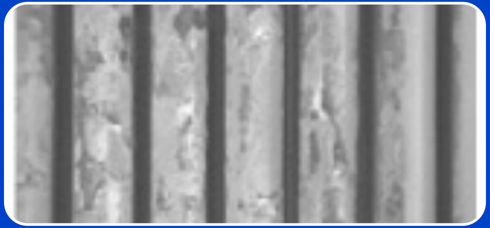
EPRI Fuel Reliability Program

- Support current operating fleet to minimize and avoid fuel failures and fuel performance issues
  - Develop and update Fuel Reliability Guidelines, Tools, and Handbooks - technical basis and operating experience
  - Inform industry to support change management

- Perform research to capture cost and operational efficiencies
  - ATF
  - HBU/LEU+
  - KOH
  - Cycle Length Extension
  - Power Uprates
  - Time-at-Temperature
  - Flexible Power Operation
  - Control Rod/Blade
  - NDE

- Develop technical basis for regulatory and safety issues
  - ATF
  - HBU/LEU+
  - FFRD

FRP Research Focus Areas

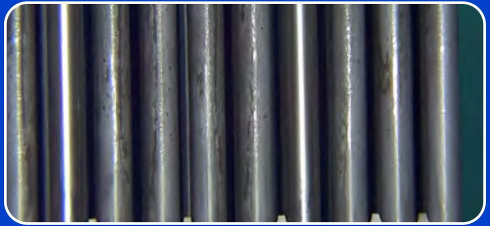


PWR Crud and Corrosion (PWR C/C)

- CIPS and CILC Risk Management, BOA CIPS/CILC Risk Assessment Tool
- KOH, PWR Fuel Cladding Corrosion and Crud Guidelines

BWR Crud and Corrosion (BWR C/C)

- BWR Fuel Cladding Corrosion and Crud Guidelines, CORAL Crud Risk Assessment Tool
- Water Chemistry Changes (i.e. Early/Continuous OLNC)



Advanced Fuel Technologies (AFT)

- Evaluate fuel reliability and performance benefits of ATF/HBU/LEU+, updates EPRI guidelines, tools, and handbooks for implementation of ATF/HBU/LEU+,
- Perform research to enable safety and economic benefits of ATF/HBU/LEU+

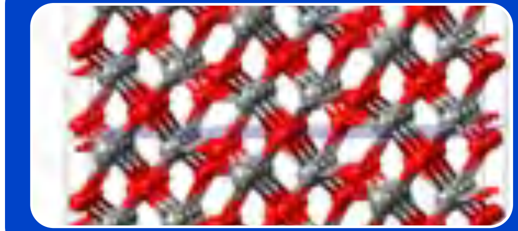
Debris-Induced Failure Mitigation (DFM)

- Guidance and Training on FME Control
- Research that enables debris-resistant fuel cladding

LEU+ is 5-8%

EPRI

Con't FRP Research Focus Areas

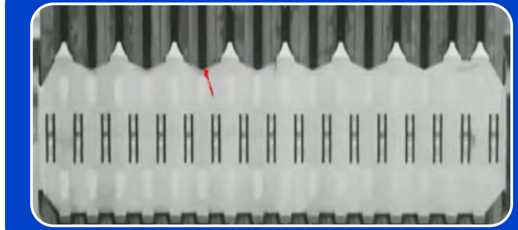


Study of Hydrogen Impacts in Zirconium (SHIZAM)

- HPU Measurement Data and Margin Assessments, Scientific Understanding of HPU and Hydrogen Impact Mechanisms
- Design and Operation Guidance Handbook

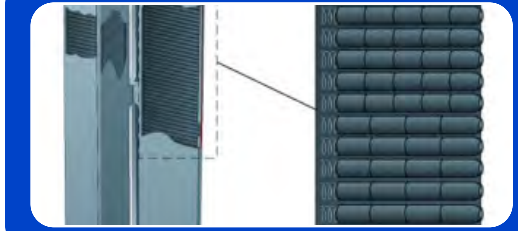
Guidance Methods and Tools (GMT)

- Fuel Surveillance and Inspection Guidelines, PCI Guidelines, FRED Database
- Fuel Failure Monitoring and Evaluation Handbook, Fuel Design Handbook, Falcon – Fuel Performance Tool



Non-destructive Evaluation (NDE)

- Failed Fuel Identification, Anomaly Identification and Characterization
- Fuel Cladding Corrosion, Crud, and Hydrogen Content Measurements in Poolside (F-SECT Oxide and F-SECT Hydrogen)



Control and Structural Component Integrity (CCI/SCI)

- BWR CRB Leakage and Lifetime Prediction Improvements, PWR Control Rod Wear Modeling
- Additive Manufacturing of Fuel Components (i.e. debris filters)

ATF Concepts in US – Near-term and Long-term Plan

Framatome	General Electric	Westinghouse
Cr-coated M5 Cladding (PWRs) Proprietary Coated Cladding (BWRs)	Coated Cladding (ARMOR™) (BWR)	Cr-coated ZIRLO cladding (PWR)
Doped UO_2 for improved thermal conductivity and fuel performance	FeCrAl Cladding (IronClad™) (BWR)	Doped UO_2 (ADOPT™) and high-density fuels with improved thermal conductivity (UN)
High Burnup (75 GWD/MTU)/LEU+	High Burnup (75 GWD/MTU)/LEU+	Interim Burnup (68 GWD/MTU) and High Burnup (75 GWD/MTU)/LEU+
Long Term: SiC Cladding	Long Term: Oxide Dispersion-strengthened (ODS) Variants of FeCrAl for improved strength, Advanced Ceramic Fuels (next generation dopants)	Long Term: SiC Cladding

LEU+ = 5-8%

Each vendor is developing near and long-term ATF cladding concepts

ATF/HBU/LEU+ Lead Fuel Assembly Programs in US

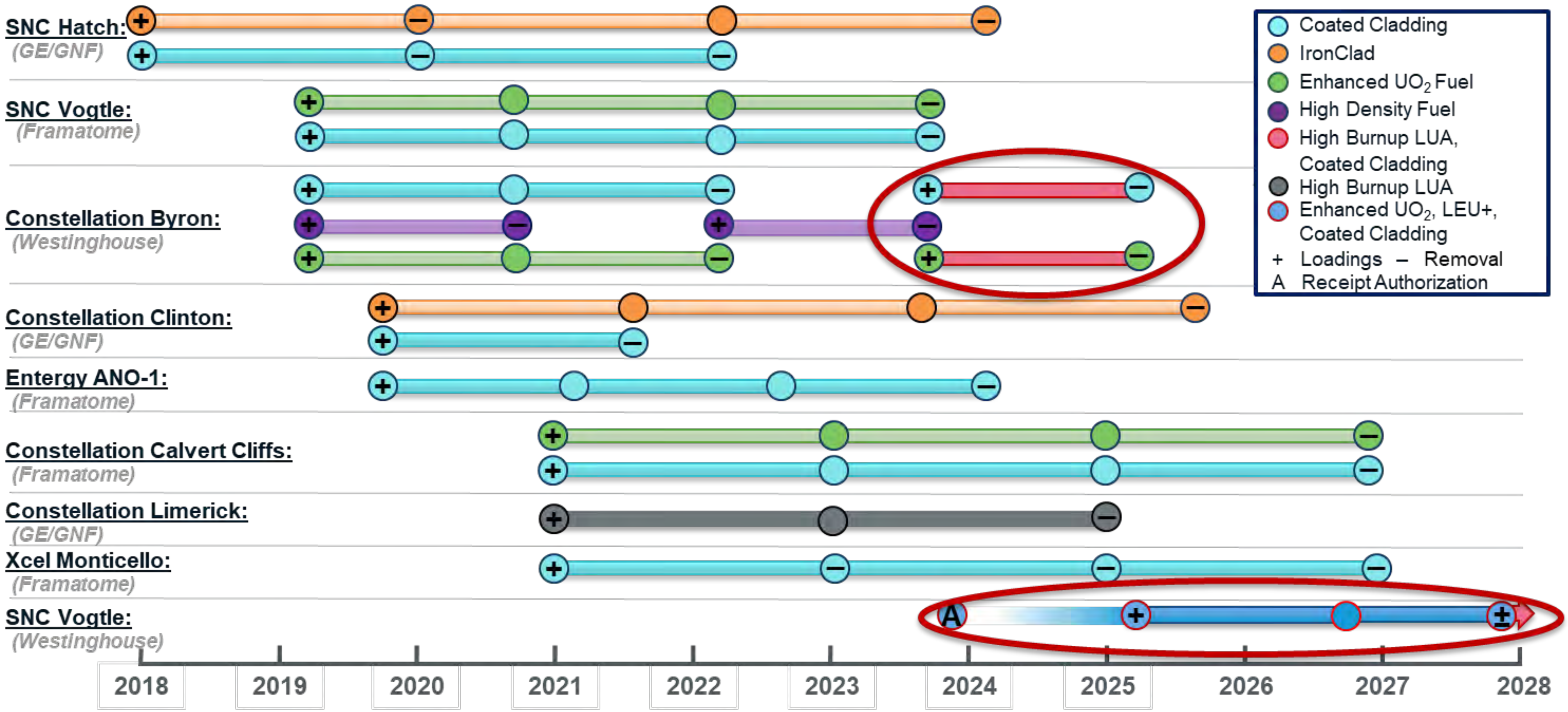
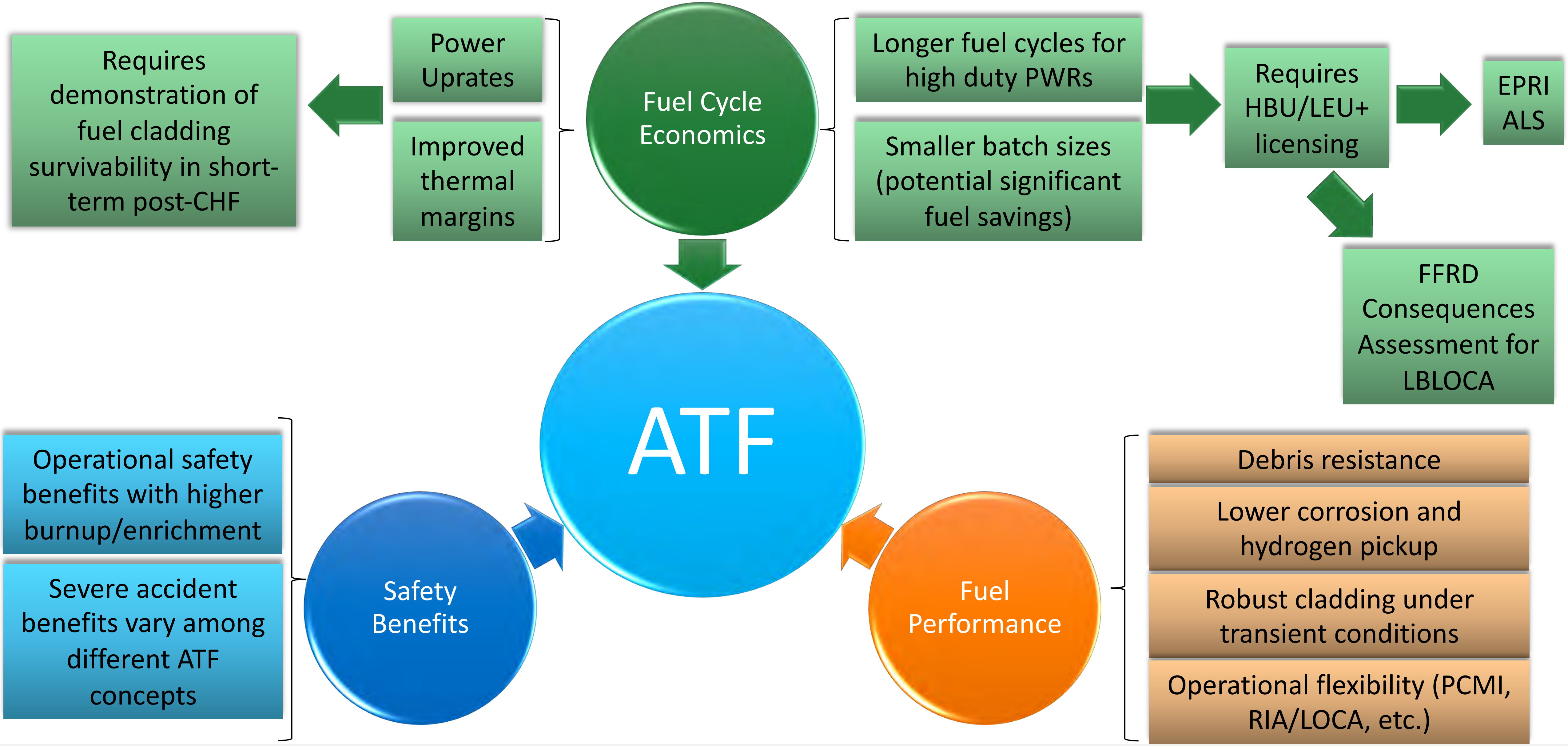


Chart courtesy of NEI

U.S. ATF Program Overview and Activity in the Industry

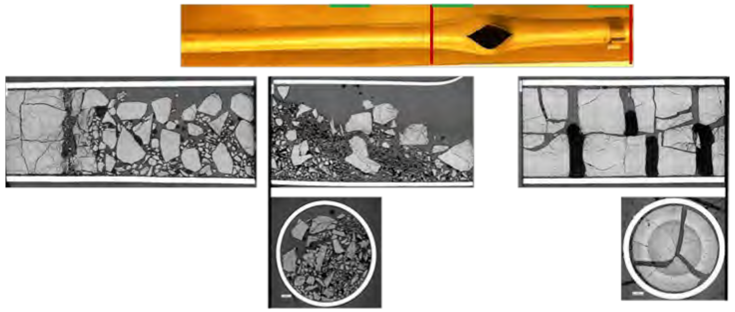


Fuel Cycle Economics – HBU/LEU+

Goal: Develop technical and licensing bases for resolving FFRD issue and open the path for industry to extend burnup up to 75 GWd/MTU

Alternative Licensing Strategy

- Approach to address FFRD in high burnup PWR fuel
  - Perform small break and intermediate break LOCA analysis to demonstrate no clad rupture and acceptable fuel relocation
  - Realistic treatment of large break LOCAs based on xLPR (Extremely Low Probability of Rupture) calculated event propagation and T/S required plant shutdown for Leak Before Break (LBB) qualified piping



CRAFT - Collaborative Research on Advanced Fuel Technologies

FFRD Consequences Assessments for LBLOCA

- Fuel suppliers are developing methods to assess FFRD consequences
- EPRI is performing analysis and testing to investigate the FFRD consequences and develop technical basis data
  - Fuel Fragment Fragility
  - Fragment Terminal Velocity in Steam
  - Mobility in water (Containment Impact)
  - Fuel Release
  - Fragment Dispersal in RCS/Containment
- Industry wide coordination of FFRD issue through CRAFT

Note: EPRI's NFIR Program performed a series of separate effect tests to understand the fuel fragmentation phenomenon and developed a threshold – detailed presentation later this morning

Fuel Cycle Economics - Power Uprates and Thermal Margin Improvements

- Relaxed TaT T/H criteria has significant economic benefits to all plants
  - IRA Production Tax Credit drives utilities considering power uprates
    - Many plants are DNB/MCPR limited
    - Credit TaT for select transients
    - Establish cladding performance limits
  - Coated cladding may provide additional margin for cladding survivability at post-CHF condition

- CRAFT TaT Technical Experts Group (TEG)
  - Develop a material testing plan including identification of testing facility, defining the test protocol, selection of materials, and determination of the funding source
  - Fully vetted research plan by all industry experts and stakeholders through CRAFT

CRAFT - Collaborative Research on Advanced Fuel Technologies, TaT – Time-at-Temperature, IRA – Inflation Reduction Act

ATF Fuel Performance Assessment and Implementation

Goal: Demonstrate no harm to plant operation and assess fuel reliability margins – focus is full reload implementation, not product development

Coating Behavior Assessment

- BWR ATF Performance in Transient Water Chemistry
- PWR Cr-51 Assessment due to Cr-coating Dissolution
- Irradiated Coated PWR Cladding - Corrosion and Hydrogen Pickup Measurements
- Atomistic Modeling of Coating Behavior

NDE Technology Development

- F-SECT Qualification for Coating Thickness Measurements
- F-SECT Poolside Demo for Coating Thickness Measurements

Fretting Wear Assessment

- Fretting Wear Testing and Modeling of ATF Concepts - Qualitative Margin Assessment for Debris-Induced Failure Mitigation

HBU Fuel Assessment

- BWR/PWR HBU Fuel Characterization - Thermo-Mechanical Model Improvements
- HBU Fuel PCI Risk Assessment

Safety Benefits

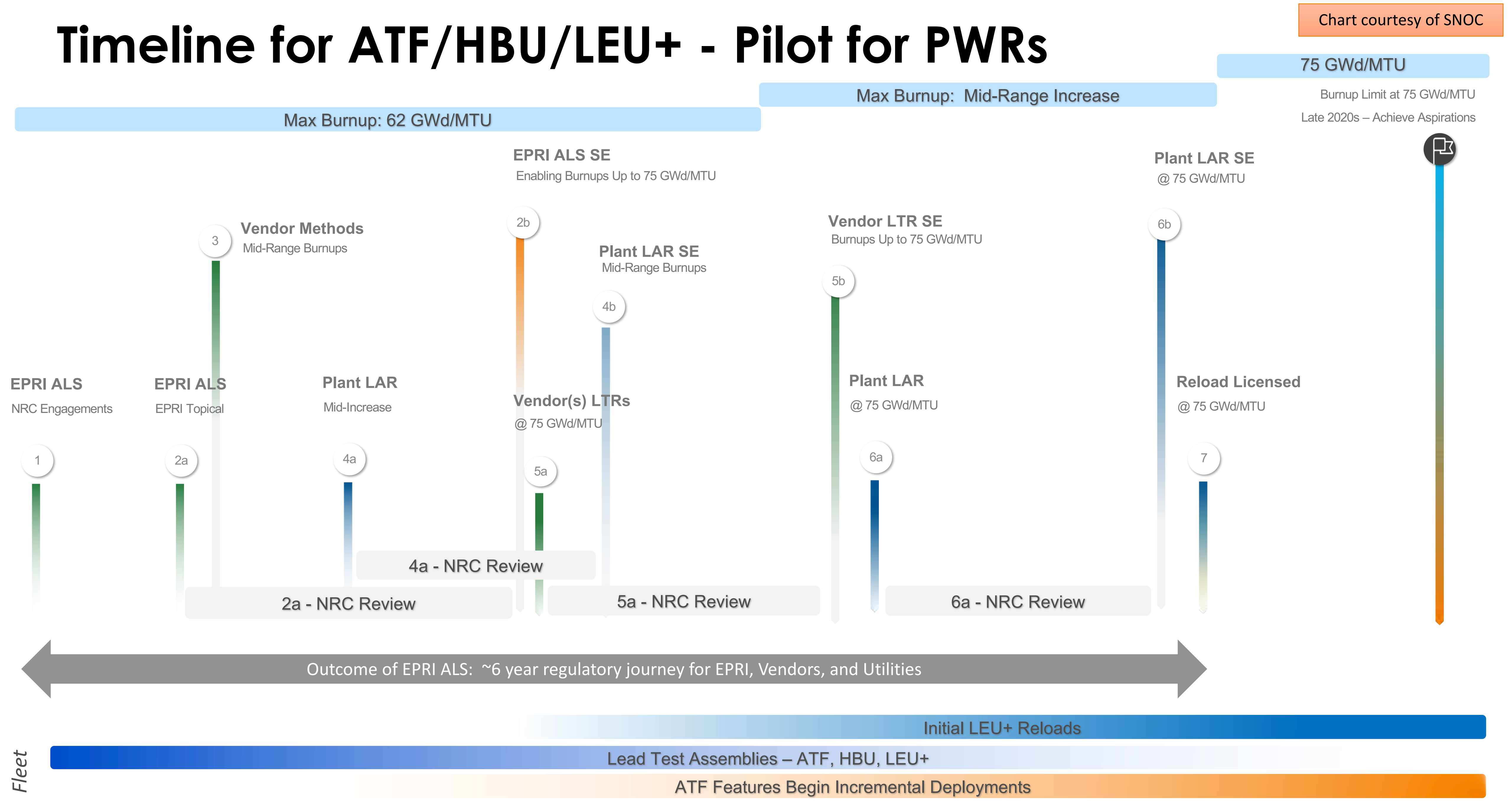
HBU Safety Benefits

- Fuel cycle
  - Reducing reload batch sizes
  - Reducing fuel handling accident risk
  - Producing less high-level rad waste
  - Reduced dose for the whole fuel cycle
- Back-end impacts
  - Decay heat
  - Criticality
  - Radiation Dose

ATF Severe Accident Safety Benefits

- Safety benefits vary depending on the ATF cladding concepts and accident scenarios
  - Each concept provides additional coping time
- Plan to reassess safety benefits
  - Upgrades to Modular Accident Analysis Program (MAAP) code for ATF cladding, higher burnup, and increased enrichment with new models and material properties
  - Participating in OECD-NEA QUENCH-ATF Project and severe accident code benchmark exercise
- Accident Source Term impacts
  - NOT driven by fuel burnup increase

Timeline for ATF/HBU/LEU+ - Pilot for PWRs

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xLPR Methodology

Probabilistic Fracture Mechanics for PWR Piping

Craig Harrington and Nate Glunt
EPRI Materials Reliability Program (MRP)

Markus Burkardt and Gideon Schmidt
Dominion Engineering, Inc. (DEI)

U.S. NRC ACRS Fuel, Materials, and Structures Subcommittee Meeting
May 18, 2023

Outline

- xLPR Overview
- xLPR Work Scope
- Summary of xLPR Analysis Cases
- Key Results
  - LOCA frequency compared to NUREG-1829
  - Time between detectable leakage and rupture
  - Investigating limiting cases
- Conclusions

List of Acronyms

ALS	Alternative licensing strategy
CE	Combustion Engineering
CL	Cold leg
DMW	Dissimilar metal weld
DN	Diametre nominal
FFRD	Fuel fragmentation, relocation and dispersal
HL	Hot leg
ISI	In-service inspection
LBB	Leak-before-break
LRD	Leak rate detection
LOCA	Loss-of-coolant accident
MDM	Materials Degradation Matrix
NPS	Nominal pipe size
NRC TLR	US Nuclear Regulatory Commission Technical Letter Report
PWR	Pressurized water reactor
PWSCC	Primary water stress corrosion cracking
PZR	Pressurizer
RCP	Reactor coolant pump
RCS	Reactor coolant system
RVIN	Reactor vessel inlet nozzle
RVON	Reactor vessel outlet nozzle
SGIN	Steam generator inlet nozzle
SGON	Steam generator outlet nozzle
WRS	Weld residual stress
xLPR	Extremely Low Probability of Rupture

xLPR Overview

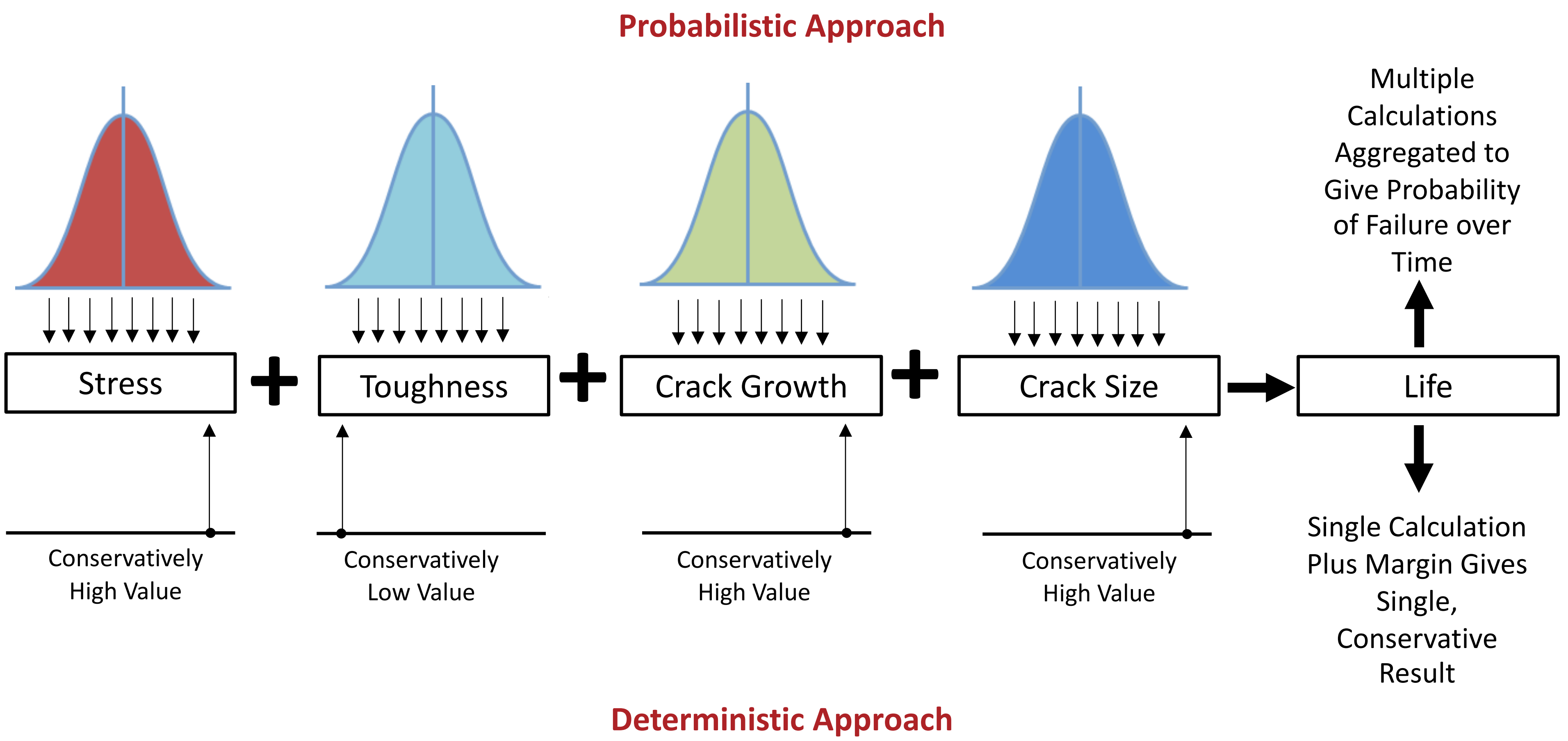
xLPR Probabilistic Fracture Mechanics Code

- xLPR is a state-of-the-art **probabilistic fracture mechanics code** jointly developed by the NRC's Office of Nuclear Regulatory Research and the Electric Power Research Institute (EPRI)

- Provides new quantitative capabilities to analyze the risks (e.g., leakage or rupture) associated with nuclear power plant piping systems subject to active degradation mechanisms

- Core capabilities include modeling fatigue, stress corrosion cracking, inservice inspection, chemical and mechanical mitigation, leak rates, and seismic effects

PROBABILISTIC VS. DETERMINISTIC



xLPR Work Scope

xLPR Work Scope & the Fuels Alternative Licensing Strategy

- Use xLPR probabilistic fracture mechanics analyses to provide validation of the expert elicitation-based LOCA frequency estimates within NUREG-1829, Vol. 1, “Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process”

- Gain insights from xLPR analyses about the time between detectable leakage and rupture

- Key xLPR outputs investigated through this report, which are inputs for the fuels alternative licensing strategy (ALS) for fuel fragmentation, relocation, and dispersal (FFRD), are:
  - Probability of LOCAs (e.g., pipe ruptures) as a function of line size
  - Probability that leakage as a precursor to a LOCA will be detected in sufficient time to allow for reactor shutdown and reduce decay heat levels before a reactor coolant system (RCS) piping rupture occurs

Line Size Considerations

- NUREG-1829 gives estimates of LOCA frequencies based on expert elicitation (Table 1)
- The expert elicitation considered LOCA-sensitive piping systems and associated degradation mechanisms (Table 3.5)

Table 1 Total BWR and PWR LOCA Frequencies (After Overconfidence Adjustment using Error-Factor Scheme)

Plant Type	LOCA Size (gpm)	Eff. Break Size (inch)	Current-day Estimate (per cal. yr) (25 yr fleet average operation)				End-of-Plant-License Estimate (per cal. yr) (40 yr fleet average operation)			
			5th Per.	Median	Mean	95th Per.	5th Per.	Median	Mean	95th Per.
BWR	>100	½	3.3E-05	3.0E-04	6.5E-04	2.3E-03	2.8E-05	2.6E-04	6.2E-04	2.2E-03
	>1,500	1 7/8	3.0E-06	5.0E-05	1.3E-04	4.8E-04	2.5E-06	4.5E-05	1.2E-04	4.8E-04
	>5,000	3 ¼	6.0E-07	9.7E-06	2.9E-05	1.1E-04	5.4E-07	9.8E-06	3.2E-05	1.3E-04
	>25K	7	8.6E-08	2.2E-06	7.3E-06	2.9E-05	7.8E-08	2.3E-06	9.4E-06	3.7E-05
	>100K	18	7.7E-09	2.9E-07	1.5E-06	5.9E-06	6.8E-09	3.1E-07	2.1E-06	7.9E-06
	>500K	41	6.3E-12	2.9E-10	6.3E-09	1.8E-08	7.5E-12	4.0E-10	1.0E-08	2.8E-08
PWR	>100	½	6.9E-04	3.9E-03	7.3E-03	2.3E-02	4.0E-04	2.6E-03	5.2E-03	1.8E-02
	>1,500	1 5/8	7.6E-06	1.4E-04	6.4E-04	2.4E-03	8.3E-06	1.6E-04	7.8E-04	2.9E-03
	>5,000	3	2.1E-07	3.4E-06	1.6E-05	6.1E-05	4.8E-07	7.6E-06	3.6E-05	1.4E-04
	>25K	7	1.4E-08	3.1E-07	1.6E-06	6.1E-06	2.8E-08	6.6E-07	3.6E-06	1.4E-05
	>100K	14	4.1E-10	1.2E-08	2.0E-07	5.8E-07	1.0E-09	2.8E-08	4.8E-07	1.4E-06
	>500K	31	3.5E-11	1.2E-09	2.9E-08	8.1E-08	8.7E-11	2.9E-09	7.5E-08	2.1E-07

Table 3.5 PWR LOCA-Sensitive Piping Systems

System	Piping Matls.	Piping Size (in)	Safe End Matls.	Welds	Sig. Degrad. Mechs.	Sig. Loads.	Mitigation/ Maint.
RCP: Hot Leg	304 SS, 316 SS, C-SS, SSC-CS CS – SW	30 - 44	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
RCP: Cold Leg/Crossover Leg	304 SS, 316 SS, C- SS, SSC-CS, CS – SW	22 - 34	A600, 304 SS, 316 SS, CS	A82 304 SS, 316 SS, CS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, SUP	ISI w TSL, REM
Surge line	304 SS, 316 SS, C-SS	10 - 14	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FDR, UA	P, S, T, RS, DW, O, TFL, TS	TSMIT, ISI w TSL, REM
SIS: ACCUM	304 SS, 316 SS, C-SS	10 - 12	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
SIS: DVI	304 SS, 316 SS	2 – 6	A600, 304 SS, 316 SS,	A82 304 SS, 316 SS	TF, SCC, MA, FS, FDR, UA (FAC)	P, S, T, RS, DW, O	ISI w TSL, REM
Drain line	304 SS, 316 SS, CS	< 2"			MF, TF, GC, LC, FDR, UA	P, S, T, RS, DW, O, V, TFL	ISI w TSL, REM
CVCS	304 SS, 316 SS	2 – 8	A600 (B&W and	A82	SCC, TF, MF, FDR, UA	P, S, T, RS, DW, O, V	ISI w TSL, REM

The goal of the current study is to analyze piping welds > NPS 14 (> DN 350) in support of alternative licensing strategy (ALS) for FFRD

Investigation of Other Degradation Mechanisms

- NUREG-1829 considers additional material degradation mechanisms not included in xLPR
  - A review of the Materials Degradation Matrix (MDM) was performed Mechanisms relevant to 300 series stainless steels and Alloy 82/182 welds in PWR primary system piping are rigorously identified
  - Identified degradation mechanisms are either evaluated herein, addressed by other industry guidance, or are not anticipated (per the MDM) to be degradation modes of concern
- Consequently, results from xLPR considering primary water stress corrosion cracking (PWSCC) and fatigue provide valuable information regarding conservatism or non-conservatism of the NUREG-1829 LOCA frequencies

Summary of xLPR Analysis Cases

Analysis Cases

- xLPR analysis cases were developed applying PWSCC and/or fatigue *(driven by plant transients and not local thermal fluctuations or vibration)* as the material degradation mechanisms
- Analysis cases either modeled flaws as present at the start of the simulation or used initiation models to calculate the time to flaw initiation
  - All flaws at initiation were modeled as flaws of engineering scale.
- Sensitivity studies were performed to determine the impact of changes to analysis inputs
  - Sensitivity studies modeled alternate inputs for parameters such as geometry, loading, weld residual stress profiles, or initial flaw sizes

Output Quantities of Interest

- Results Directly Output by xLPR
  - Probability of rupture
    - Used to calculate rupture frequencies
    - Option of conservatively not crediting in-service inspection (ISI) or leak rate detection (LRD)
    - Results utilizing initial flaw of engineering scale are conditional on crack initiation
  - Probability of crack initiation
  - Leak rate
- Post-Processed Results
  - Time between 1 gpm detectable leakage and rupture ("lapse time")
  - $P(\text{Rupture}|\text{Initiation}) \approx P(\text{Rupture}|\text{Initial Flaw}) \times P(\text{Initiation})$
  - Average 80-year rupture (LOCA) frequency = P(Rupture) / 80 yrs

Previous xLPR Studies

- xLPR analyses have recently been published by the US NRC in the context of LBB analyses for A82/182 dissimilar metal butt welds in PWR piping systems:
  - **TLR-RES/DE/REB-2021-09 (ML21217A088)**
    - Referred to herein as "xLPR piping system analysis"
    - Documented xLPR analysis of representative reactor vessel outlet and inlet nozzle welds in a Westinghouse four-loop PWR
    - Includes extensive set of sensitivity studies
  - **TLR-RES/DE/REB-2021-14 R1 (ML22088A006)**
    - Referred to herein as "xLPR generalization study"
    - Documented xLPR analysis of other piping systems containing Alloy 82/182 dissimilar metal piping butt welds which had received prior LBB approvals from the NRC staff
    - Includes reduced set of sensitivity studies per analyzed component, as informed by "xLPR piping system analysis"

The results of these analyses are used where possible and supplemented with additional xLPR analysis cases as needed

Summary of Base Cases

Study	Piping System Analysis		Generalization Study			
NUREG-1829 Line/System	Reactor Coolant Piping: Hot Leg	Reactor Coolant Piping: Cold Leg	Reactor Coolant Piping: Hot Leg	Reactor Coolant Piping: Cold Leg	Surge Line	Safety Injection (Accumulator)
Weld Analyzed	RVON	RVIN	RVON, SGIN	RCP Inlet/Outlet, SGON	PZR Surge, CE Hot Leg Branch Line DMW	CE Cold Leg Branch Line DMW
Fatigue Crack Growth	No	No	No	No	No	No
PWSCC Crack Growth	Yes	Yes	Yes	Yes	Yes	Yes
Initial Flaws	No	No	No	No	No	No
Axial/Circ Flaws	Circ	Circ	Both	Both	Both	Both
Seismic Occurrences	No	No	No (4-loop RVON); Yes (others)	Yes	Yes	Yes
Mitigation	No	No	No (RVON); Yes (SGIN)	No	No	No
ISI/LRD	Optional in outputs	Optional in outputs	Optional in outputs	Optional in outputs	Optional in outputs	Optional in outputs

Focus of ALS

Summary of Sensitivity Studies

Legend	
	Sensitivity study included

Study	Piping System Analysis		Generalization Study			
NUREG-1829 Line/System	Reactor Coolant Piping: Hot Leg	Reactor Coolant Piping: Cold Leg	Reactor Coolant Piping: Hot Leg	Reactor Coolant Piping: Cold Leg	Surge Line	Safety Injection (Accumulator)
Weld Analyzed	RVON	RVIN	RVON, SGIN	RCP Inlet/Outlet, SGON	PZR Surge, CE HL Branch Line DMW	CE CL Branch Line DMW
Initiation						
WRS						
Earthquake						
Normal Operating Thermal Loads						
LRD/ISI						
Mitigation						
Fatigue						
Initial Flaw Size						
Geometry						
Other	axial cracks, hydrogen, temperature					

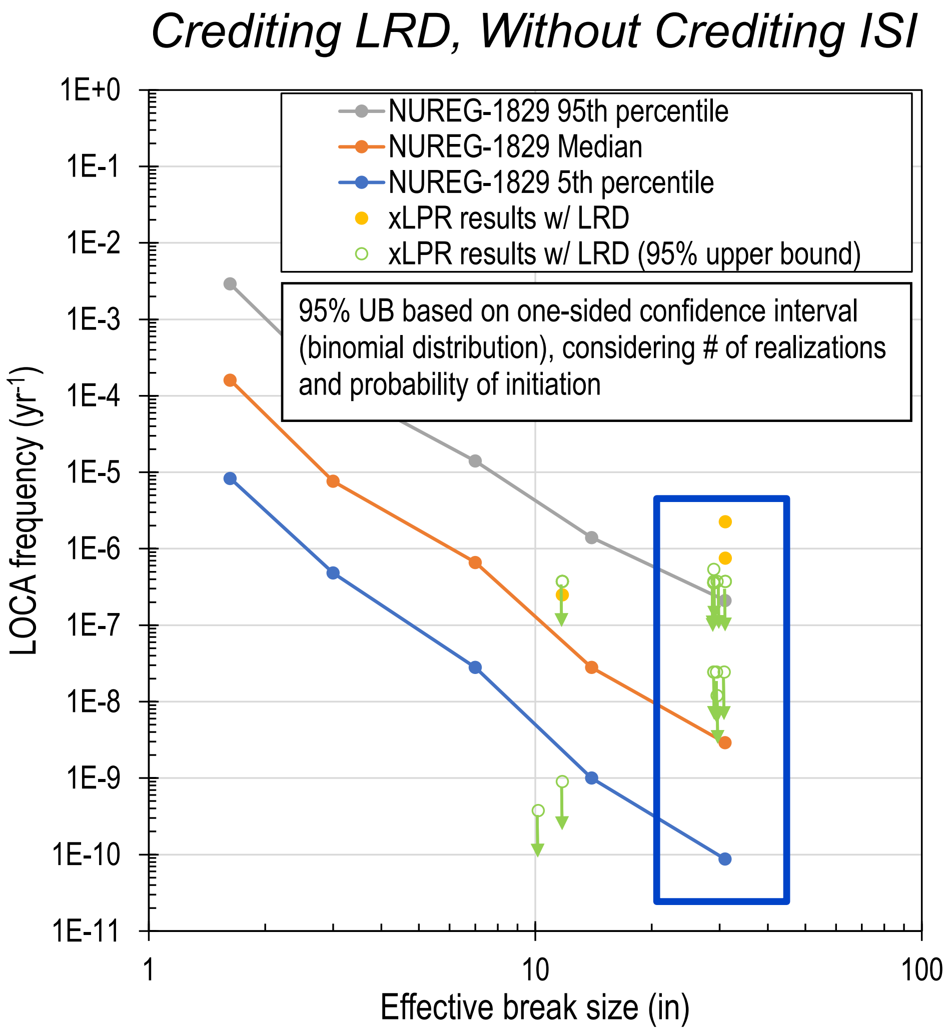
Focus of ALS

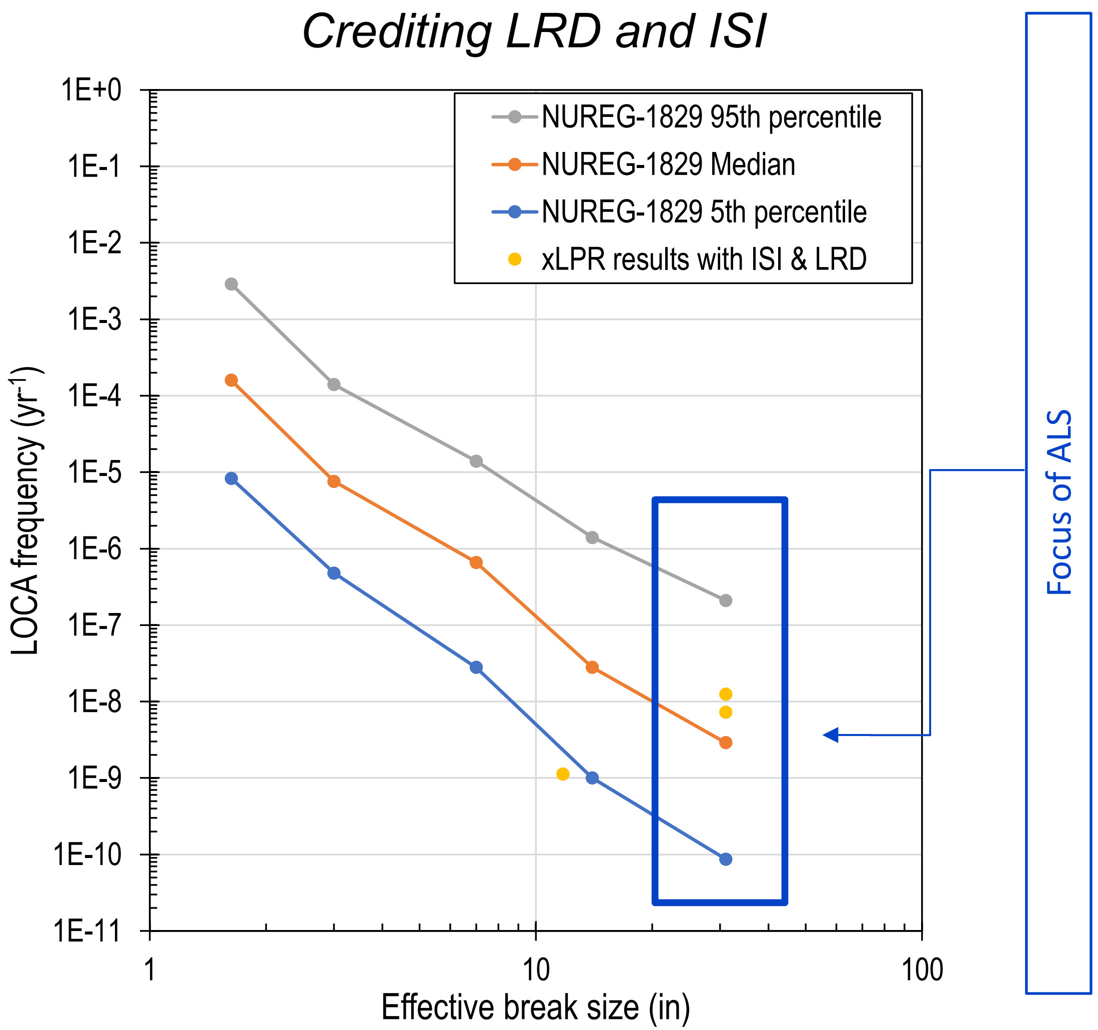
LOCA Frequency Compared to NUREG-1829

LOCA Frequency Compared to NUREG-1829 Table 1

- NUREG-1829 LOCA frequencies used for comparison are:
  - Based on expert elicitation
  - From Table 1
    - Median, 5th percentile, and 95th percentile
    - Total PWR LOCA frequencies after overconfidence adjustment using error-factor scheme
    - 40 yr fleet average values
    - Consider typical ISI with LRD resolution as required by tech spec limits
  - Results are presented on a per plant basis, for each distinct LOCA category
  - Considers piping and non-piping passive system contributions

LOCA Frequency Compared to NUREG-1829 Table 1





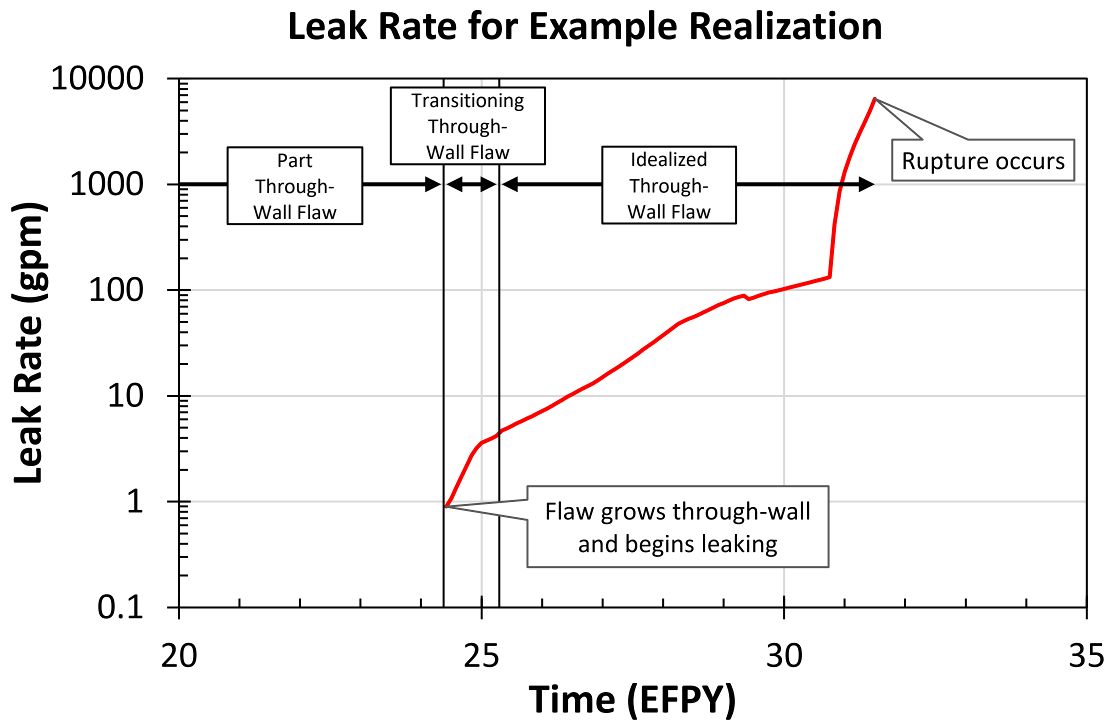
When considering ISI and LRD, LOCA frequencies estimated from xLPR are on a similar order of magnitude as median NUREG-1829 LOCA frequency estimates

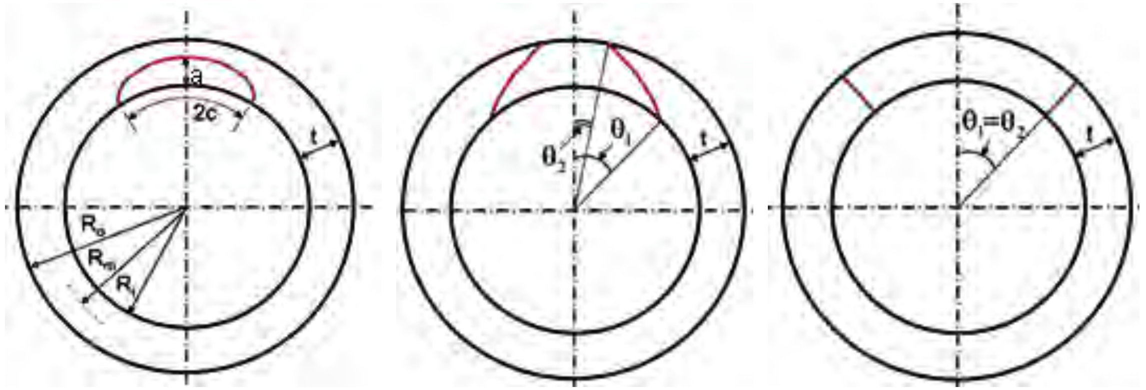
Time Between Detectable Leakage and Rupture

Time from Detectable Leakage to Rupture

For a Single xLPR Analysis Case Realization

- Results shown depict example leak rate time history for one realization modeled in xLPR
  - Component modeled: Unmitigated Alloy 82/182 reactor vessel outlet nozzle dissimilar metal weld
  - Key modeling options selected:
    - Initial flaw model (i.e., initiation at time = 0)
    - PWSCC growth only
    - One circumferential crack
    - No inservice inspection, leak rate detection, mitigation, or seismic effects





Part Through-Wall

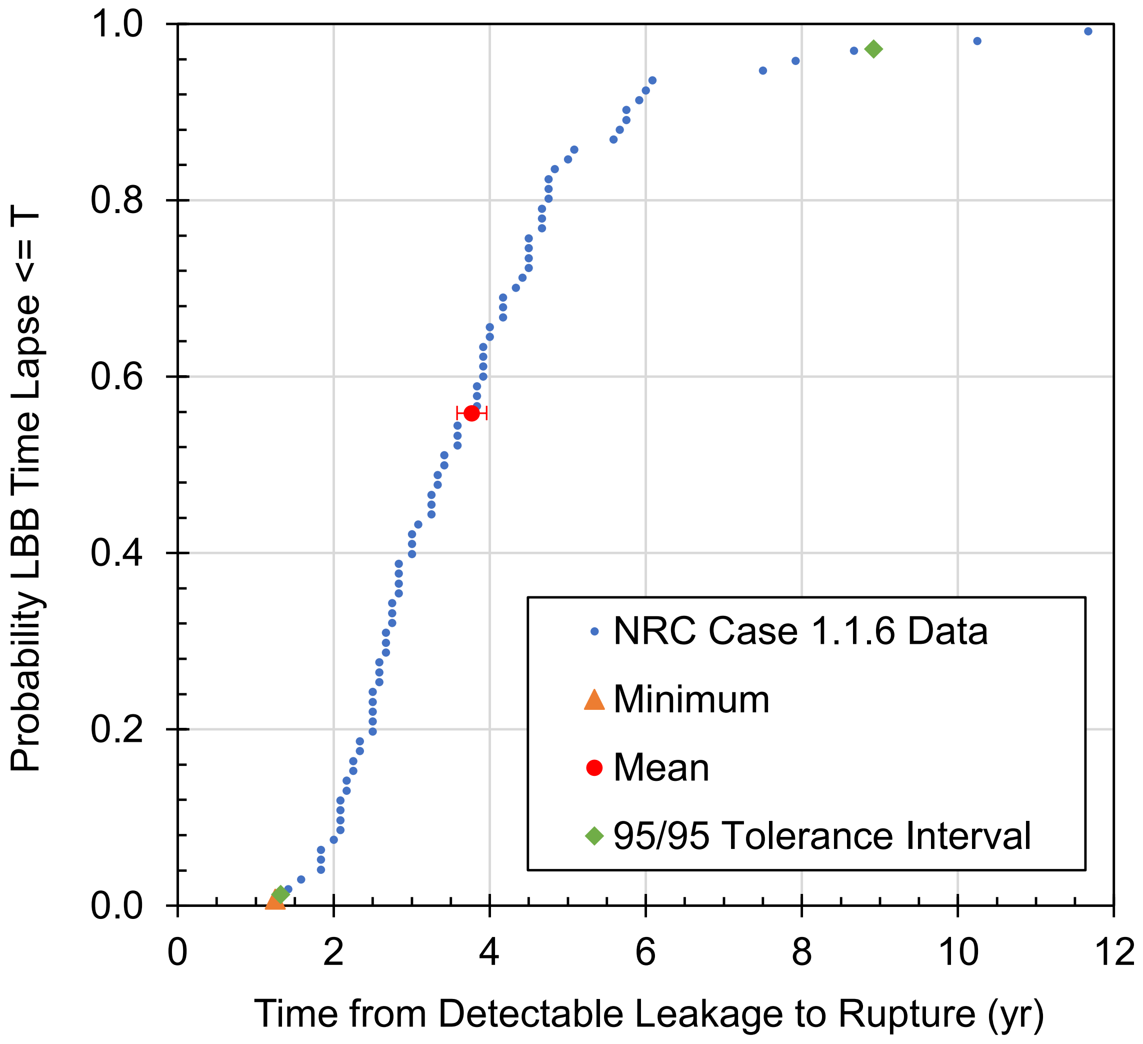
Transitioning Through-Wall

Idealized Through-Wall

Distributions of Time from Detectable Leakage to Rupture

For a Single xLPR Analysis Case

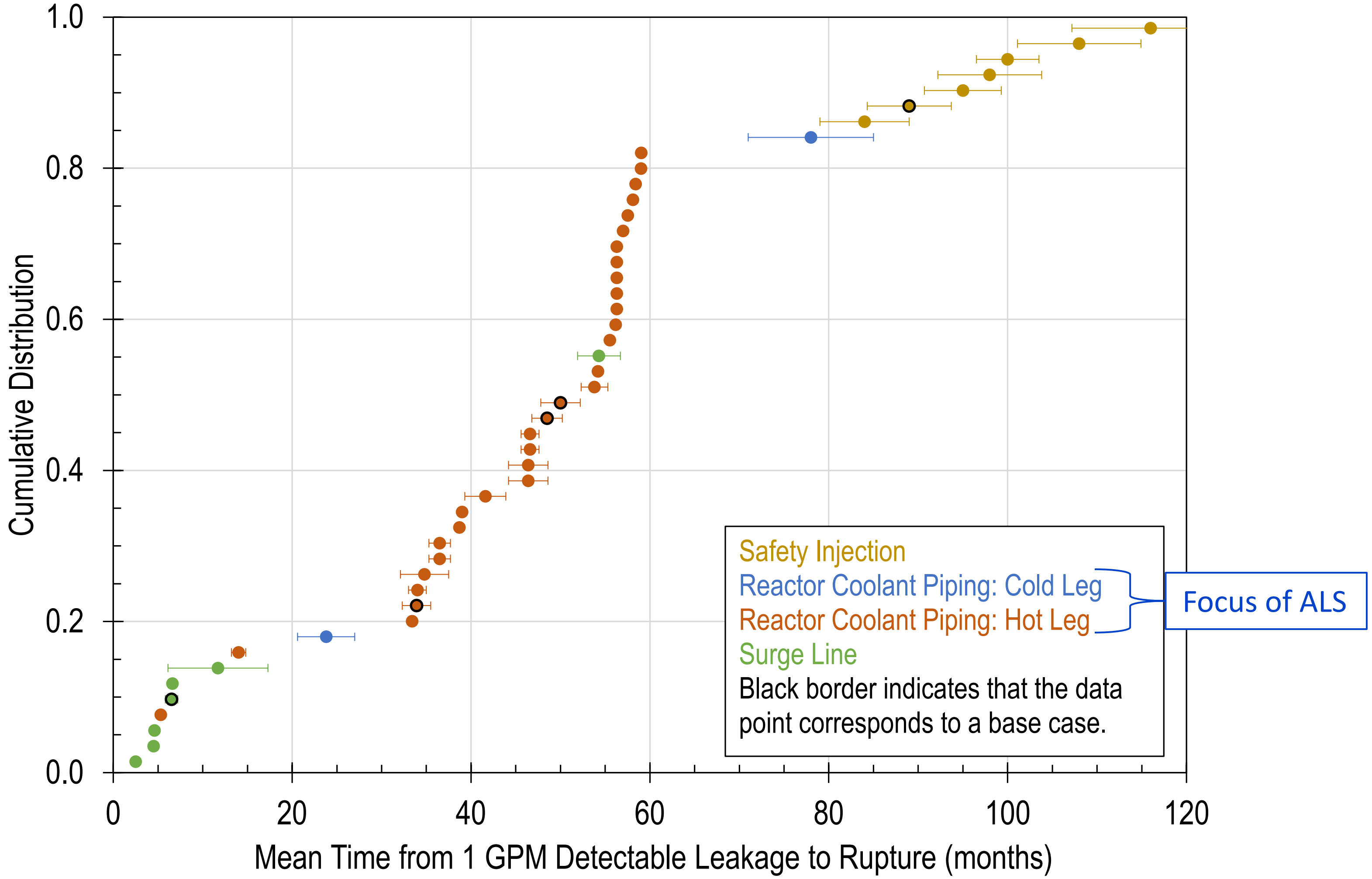
- Results for one xLPR analysis case produce a distribution of lapse times
- Each data point corresponds to one realization which resulted in rupture (without crediting ISI or LRD)
  - Note that lapse time results greater than 12 years are truncated in NRC TLRs
- Also shown:
  - Minimum
  - 95/95 tolerance interval (lognormal)
  - Mean
  - Standard error (error bars on mean)



Mean Time from Detectable Leakage to Rupture

For all xLPR Analysis Cases

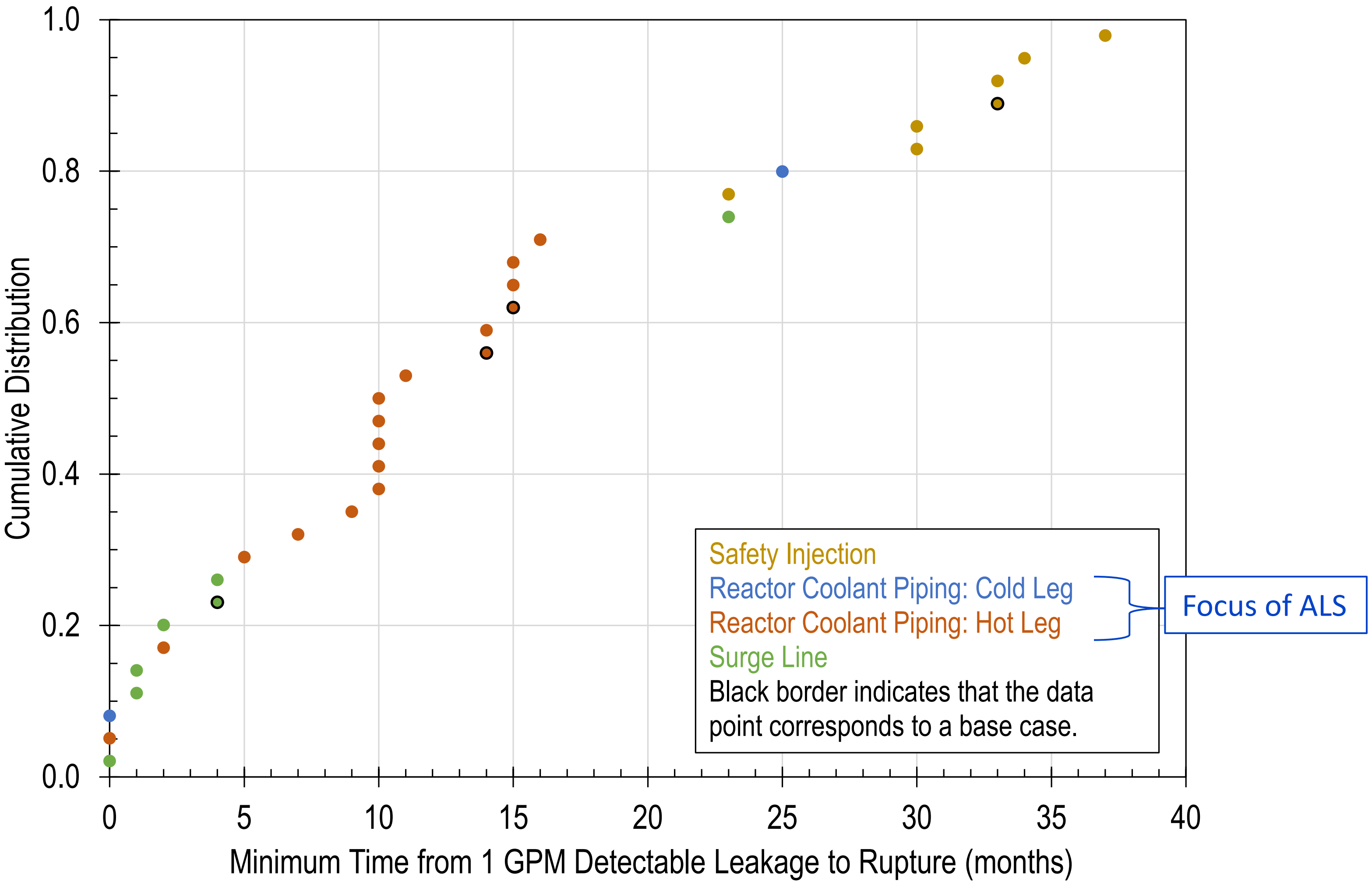
- Mean times from detectable leakage to rupture are reviewed for additional context
  - Shown with error bars equal to standard error
  - Times from detectable leakage to rupture listed as N/A in the NRC TLRs are not shown



Minimum Time from Detectable Leakage to Rupture

For all xLPR Analysis Cases

- Minimum times from detectable leakage to rupture are reviewed as a screening exercise
- Cases with minimum time from detectable leakage to rupture under 3 months are investigated in further detail
  - These cases
    - Considered unmitigated welds subject to PWSCC growth at hot leg or pressurizer temperatures or included modeling not representative of plant conditions and operations
    - Are all sensitivity studies



Investigating Limiting Cases

Time Between Detectable Leakage and Rupture

Summary of Investigation of Limiting Sensitivity Studies

- Performed further investigation for limiting cases exhibiting either:
  - Minimum time between detectable leakage and rupture < 3 months
  - Nonzero occurrence of rupture with LRD
- Some of these limiting cases were then re-run with:
  - Refined time-stepping
  - Updated input model parameters
- After dispositioning, for cases relevant to the ALS, the minimum time from detectable leakage to rupture is:
  - 14 months for the most limiting of the base cases evaluated
  - 0.8 months for the most limiting of the sensitivity studies evaluated

Time Between Detectable Leakage and Rupture

Lower Bound Times Using 95/95 Tolerance Interval

- For the cases with limiting minimum times, the 95/95 tolerance interval was computed using a lognormal distribution
  - A 95/95 tolerance interval is defined such that "there is a 95% probability that the constructed limits contain 95% of the population of interest for the surveillance interval selected"
- The 95/95 lower bound of the most limiting sensitivity study representative of the US PWR fleet is 3.8 months
- The sensitivity studies were
  - Defined to inform understanding of the base case results by investigating inputs known to have influence on xLPR results
  - Less constrained by maintaining fidelity to realistic plant conditions

Conclusions

Conclusions

- When crediting ISI and LRD, occurrence of rupture results are on a similar order of magnitude as NUREG-1829 LOCA frequency estimates
  - The only nonzero results were for cases including modeling not representative of plant conditions and operations
  - For cases with zero ruptures w/ LRD, a 95% upper bound based on a one-sided confidence interval is considered for comparison
- For all base cases and most sensitivity studies considered, the minimum observed times from 1 gpm (3.8 lpm) detectable leakage to rupture exceeded three months
  - The 95/95 tolerance intervals show that for the most limiting sensitivity study representative of the US PWR fleet, the lower bound is 3.8 months

Together…Shaping the Future of Energy®

Alternative Licensing Strategy (ALS)

LOCA induced Fuel Fragmentation, Relocation and Dispersal Topical Report

Fred Smith
Sr. Technical Executive, EPRI

May 18, 2023

U.S. NRC ACRS Fuels, Materials, and Structures Subcommittee Meeting
Bethesda, MD

ALS – Deterministic and Risk Informed Approach

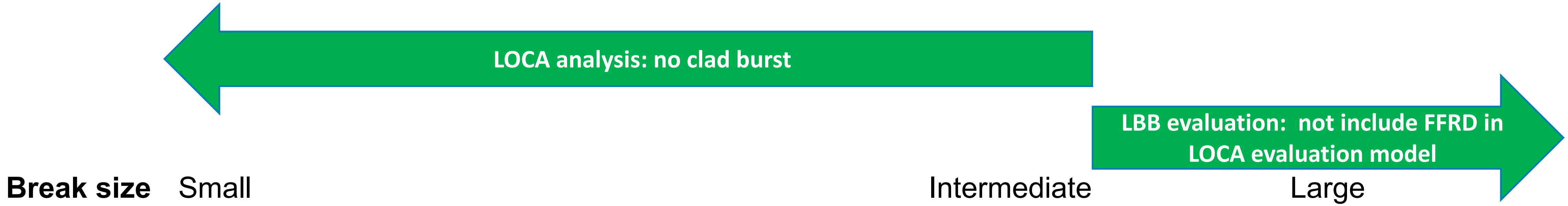
- Obtain NRC approval of generic method to address PWR LOCA induced FFRD in an expeditious manner
  - Avoid reliance on additional LOCA testing for FFRD
  - Limit licensing complexity and risk
    - Use previously approved methods and licensing strategy to the extent possible
    - Update as needed to address high burnup phenomena
  - Minimize the plant specific implementation activities
    - Confirm applicability requirements apply to specific plant

ALS Approach

- **Approach to address FFRD in high burnup PWR fuel:**
  - Small and intermediate break LOCA analysis to show acceptable fuel cooling and no clad rupture
  - Large break LOCA acceptability based on realistic xLPR calculated event propagation
    - Leak Before Break (LBB) qualified piping
    - Ample time to detect precursor leakage prior to rupture
    - T/S required plant shutdown and significant reduction in decay heat
- **Rationale**:
    - Full LOCA analysis maintained to show ECCS performance meets 10 CFR 50.46
    - LBB to exclude FFRD from evaluation model, as already done for various local phenomena
    - External to RPV (jet impingement, asymmetric vessel loading, failure of ECCS cross-connect valve)
    - Internal to RPV (control rod scram, fuel mechanical loads)

ALS Approach

- **Implementation**:
  - Apply xLPR analysis to coolant piping of reactor coolant system (RCS)
    - LBB already approved for large-diameter piping
    - Demonstrate time available from reaching detectable leak rate until potential pipe rupture is sufficient to justify crediting operator detection and reaching cold shutdown
      - LBLOCA will not induce FFRD in Mode 5
  - Non-LBB piping analyzed with design bases LOCA methods
      - SBLOCA and IBLOCAs will not cause clad burst, thereby precluding fuel dispersal
  - Utility license amendment requests (LARs) reference ALS Topical Report
      - Confirm the analysis range of applicability applies to the plant
      - Reduces repetitive NRC staff review effort



LOCA Analysis Approach for SB-LOCA & IB-LOCA

- Develop analysis applicable to Westinghouse plants
  - Bounding PWR/ECCS models
    - Nuclear design envelope
    - Fuel rod design data
  - Include conservatisms to ensure plant-specific operation falls within analysis envelope
  - Address ECCS design differences for various class of plants
- Address high burnup fuel rod phenomena
  - Transient fission gas release
  - Pre-burst axial fuel relocation
  - Cladding and fuel materials intended for high burnup operation
- Execute cladding rupture calculation for high burnup fuel population
- Define analysis applicability for utility LAR submittals

LBLOCA with LBB

Leakage Technical Specifications

- Limiting Condition for Operation (LCO)
  - No Reactor Coolant Pressure Boundary leakage
  - Unidentified Leakage < 1 gpm

Required Action:

Mode 3 within 6 hours

Mode 5 within 36 hours

- Periodic surveillance to verify LCO met
- Most appropriate operating domain for LB-LOCA induced FFRD analysis is Mode 5

RCS Operational LEAKAGE
3.4.13

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE, and

d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. OR Pressure boundary LEAKAGE exists. OR Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

Westinghouse STS 3.4.13-1 Rev. 4.0

LB-LOCA Evaluation

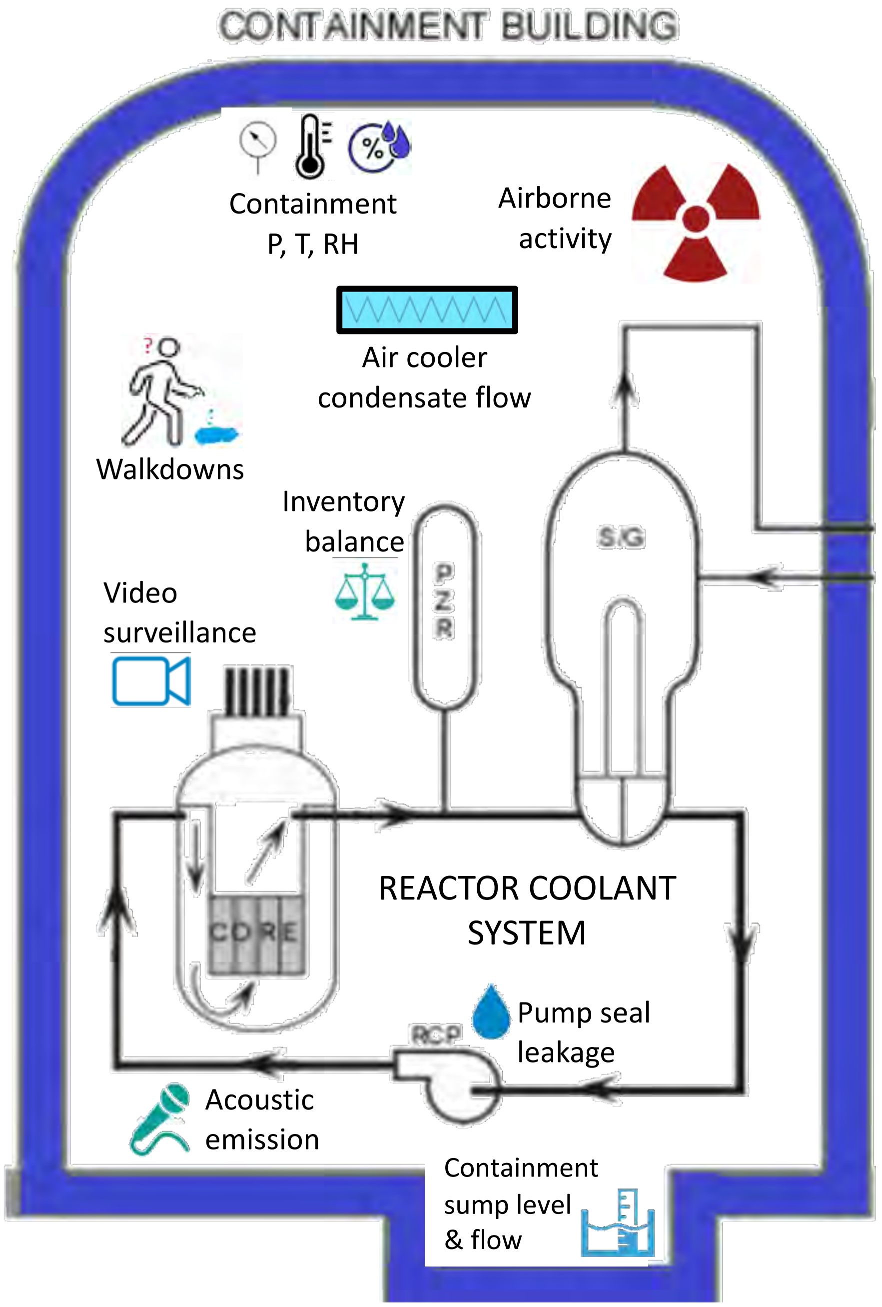
- All PWRs have LBB approved for RCS piping
  - xLPR analysis of main coolant piping demonstrates
    - 3.8 months (95/95 lower limit) between exceeding Technical Specification leakage limit and pipe rupture
  - There is no motive force to cause a pipe rupture in Mode 5
    - Even if a rupture could occur, fuel cladding burst would not occur
      - Negligible cladding heat up: decay heat will have dropped to 1/50th of full-power value
      - Core pressure reduced
      - No full system blowdown
  - Configuration is essentially the same as post LOCA long term cooling

Defense in Depth Considerations for LB-LOCA

- For LBB evaluated piping systems
  - A substantial range of pipe crack sizes are stable for an extended period of time
  - Provides detectable leaks
  - The probability of piping rupture is extremely low ($<10^{-6}$)
- Defense in depth considerations for LBB analysis are embedded in fracture mechanics analysis process
  - Main coolant pipe rupture is not assumed as an additional defense in depth consideration
- xLPR addresses uncertainties using modern uncertainty propagation to provide upper/lower limit analysis results

Defense in Depth Considerations

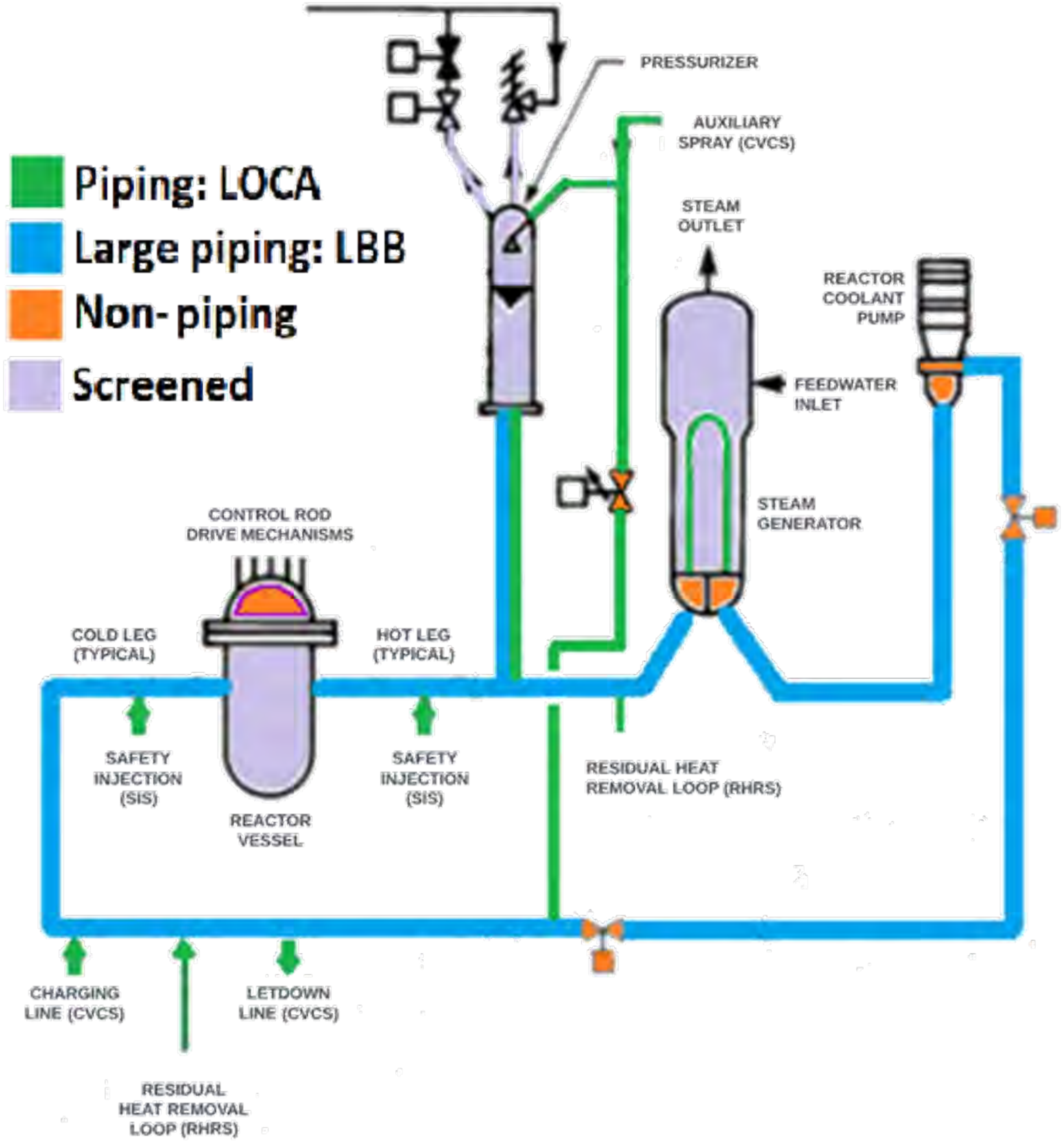
- Barriers to prevent LB-LOCA FFRD are:
  - Low frequency of LB-LOCA
  - Unidentified Leakage Technical Specification
  - Extended time between leakage detection and pipe rupture
- Defense in Depth Considerations to protect these barriers
  - Plant Operations performing T/S surveillance
    - Significant time margin before adverse consequences could develop
    - Highly qualified/trained operations staff
    - Highly proceduralized process
    - Independently review determination
    - Multiple independent indications (RG 1.45)
  - Rigorous treatment of uncertainties in xLPR ensure time to rupture is appropriately conservative



Non-piping LOCA

Evaluation of Non-piping LOCA

- Non-piping
  - Screened
    - Beyond design basis (e.g., RPV failure)
    - Bounded by LOCA with larger flow rate
  - Bolted
    - Failure mechanisms
    - Evaluation of LBB-type behavior
    - Margin to failure
  - Component bodies
    - ASME design limits allowable stress
    - Intervening flow resistance prevents flow rate high enough to cause clad burst
    - Supports/restraints make large opening implausible
  - Active component failures



Other Related Information

Sample of Various Engagements with NRC

- Alternative Licensing Approaches for Higher Burnup Fuel, July 2020, EPRI 3002018457 (Public)
- xLPR NRC Public Meeting Briefing June 14, 2022, ML22166A345
- ALS Presentation at NRC High Burnup Workshop August 24, 2022, ML22235A740
- ALS Pre-submittal meeting August 30, 2022, ML2241A133
- High Burnup Alternative Licensing Strategy (ALS) Update, CRAFT meeting November 3, 2022
- xLPR NRC Public Meeting Briefing January 19, 2023, ML 22363A572

Safety and Environmental Benefits of High Burnup fuel and ALS

- Reduced risk of transportation accidents across entire fuel cycle due to reduced volume
- Reduced risk of fuel handling accidents within a plant due to smaller reload batch sizes
- Reduced high level waste to store on site, load into dry cask containers and eventually transport and store in a repository
- Improved economic performance for nuclear sites reduce the risk of early shutdown; thereby supporting US and international environmental goals of reduced greenhouse gases emissions
- Improved core design efficiency reduces Uranium environmental and radiological impacts during mining
- Higher burnup core designs support longer fuel cycles, fewer outages and lower risk of outage related safety challenges
- Higher burnup core designs support longer cycles which results in less plant personnel dose accumulation due to fewer refueling outages
- More effective use of limited NRC and industry resources by avoiding modeling and analysis of fuel dispersal consequences

ALS enhances benefits due to faster review/approval

EPRI Docketed Topical Report

Topical Report Table of Contents

Summary of Supporting Reports
Integration of Analysis Results
- Regulatory Framework
- Exceptions to LBB Requirements
  - SRB 3.6.3
  - NRC Policy on LBB application to ECCS
- Justification for Crediting Operator Actions
- LBLOCA induced FFRD Risk Determination (Qualitative or Quantitative)

Evaluation of Non-piping failures
- Passive Components
- Active Systems

Defense-in-Depth
Applicability Requirements

Utility LAR references EPRI TR for LOCA Induced FFRD

Supporting Reports

Vendor Specific Small/Intermediate LOCA Analysis (P and NP) and Affidavit

Cladding Rupture Results

Relocation Evaluation

X-LPR Fracture Mechanics Analysis

EPRI - Materials and Advanced Nuclear

Vendor Specific LOCA Methodology Updates
- Vendor Submitted Prior to or in Parallel to ALS
- Burnup Extension Effects
- Margin Improvement Items

ALS Scope/Schedule

- **FFRD LOCA analysis**
  - Deterministic treatment of SBLOCA and IBLOCA
  - Application of LBB to LBLOCA Consistent with LBB applications
    - Limiting branch lines are the Accumulator Line Break (Cold Leg) and Pressurizer Surge Line (Hot Leg)
    - Justification of xLPR results and LBB Technical Specifications to preclude LBLOCA induced FFRD
- **Address non-piping LOCA**
  - Non-Piping Breaks – manways, component bodies, nozzles, heater sleeves
  - Active System Failures – Stuck open valves, pump seals
- **Schedule**
  - Submittal 4th quarter 2023 to 1st quarter 2024


Together…Shaping the Future of Energy®

Fuel Fragmentation Threshold

Suresh Yagnik
Senior Technical Executive, EPRI

May 18, 2023

U.S. NRC ACRS Fuels, Materials, and Structures Subcommittee Meeting

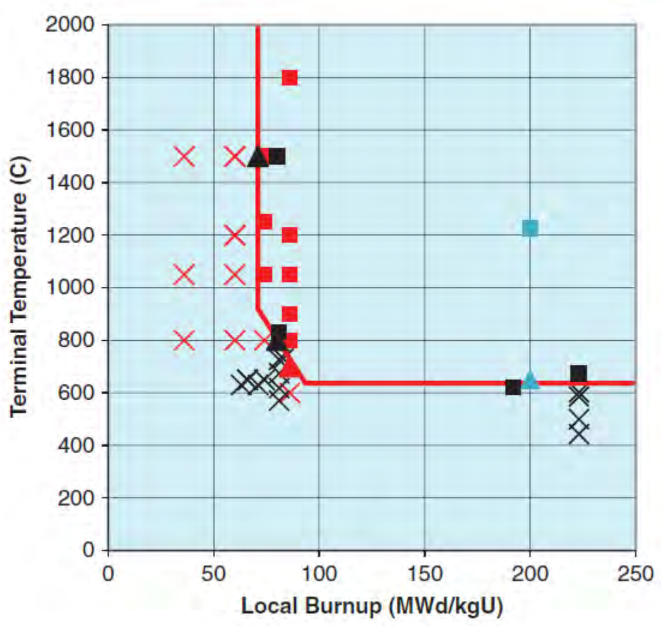
Bethesda, MD

Outline

- Best estimate Fuel Fragmentation (FF) threshold
- Scoping tests
- Mechanistic studies on IFA-649 fuel disc samples
- Summary and future outlook

Best estimate fuel fragmentation threshold

- Separate effects, Halden (IFA-650 series), SCIP-III, and open literature data were evaluated and a best estimate threshold for FF recommended in 2015
  - Local temp and local burn-up dependent; easy to implement in fuel performance codes; published in open literature and used in DOE's BISON code and elsewhere
  - Can help assess mass of fuel that could be subject to fragmentation in a fuel rod

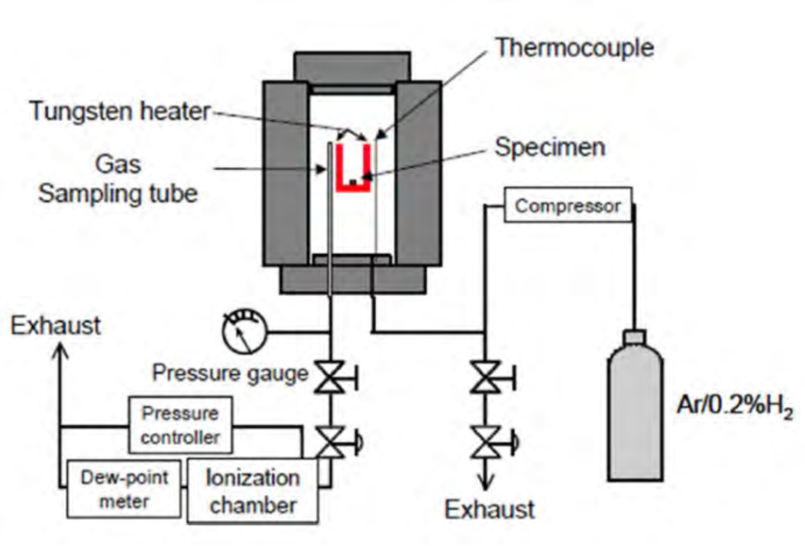


- Fragmentation limit (fuel local burn-up + terminal test temperature)

NUCLEAR SCIENCE AND ENGINEERING: 179, 477–485 (2015)

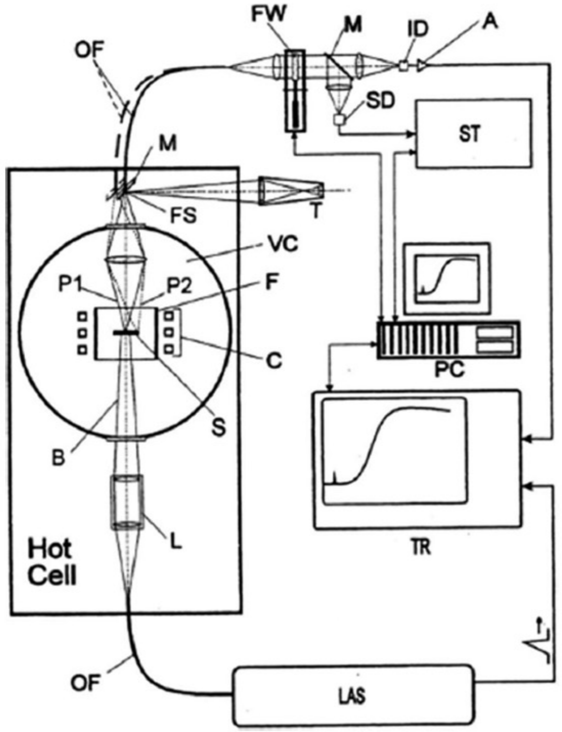
Initial scoping studies using LWR irradiated fuels

M. Hirai et al.



- Very small samples
- Fast temperature ramps
- Effect of hydrostatic restraint pressure

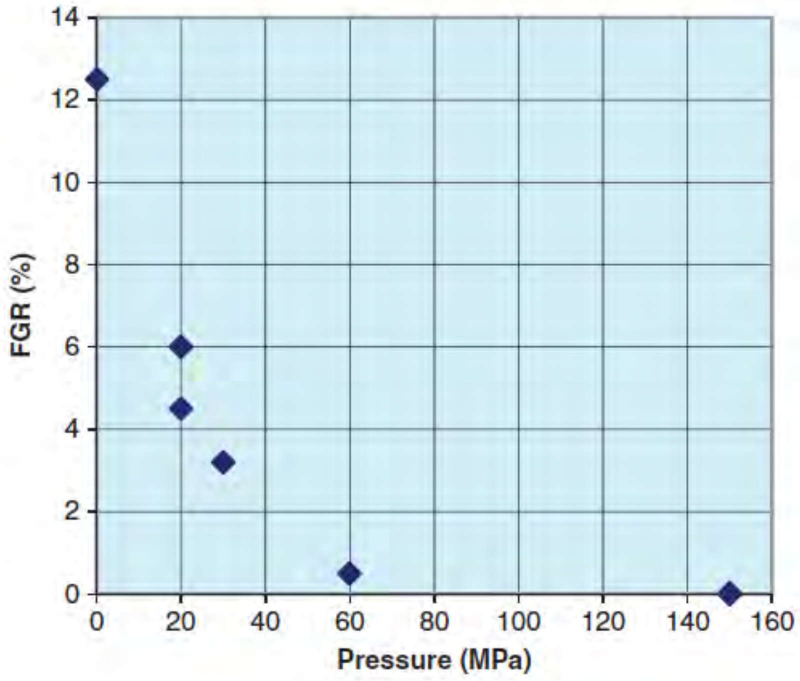
D. Staicu et al.



- Induction + multiple laser pulse heating
- Fragmentation temperature investigation
- Optical images after laser pulses

Effect of external restraint

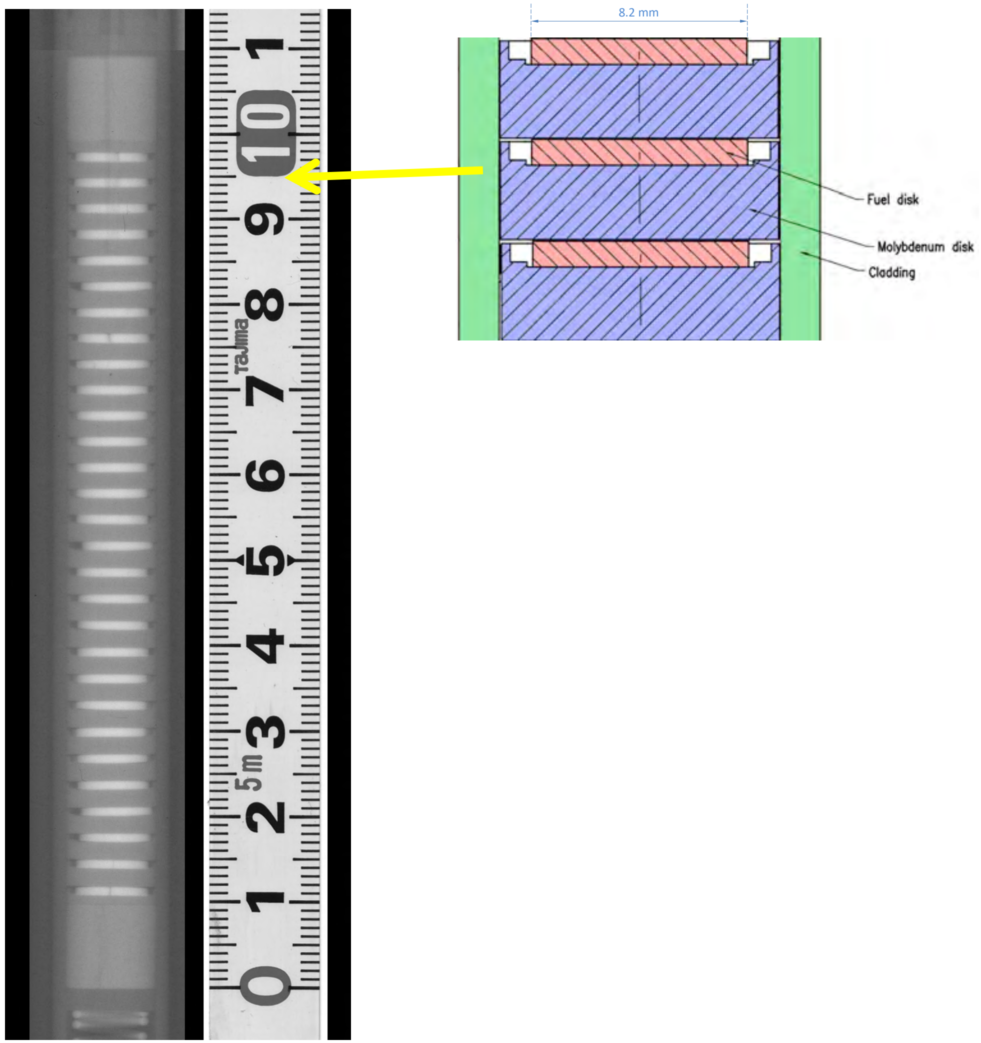
- Scoping tests clearly confirmed that PCMI restraint will reduce FF
- This was further confirmed on IFA-649 fuel disc samples at the LECA lab (CEA, Cadarache)



- High hydrostatic restraint pressures prevented fission gas release and thus fragmentation

NUCLEAR SCIENCE AND ENGINEERING: 179, 477–485 (2015)

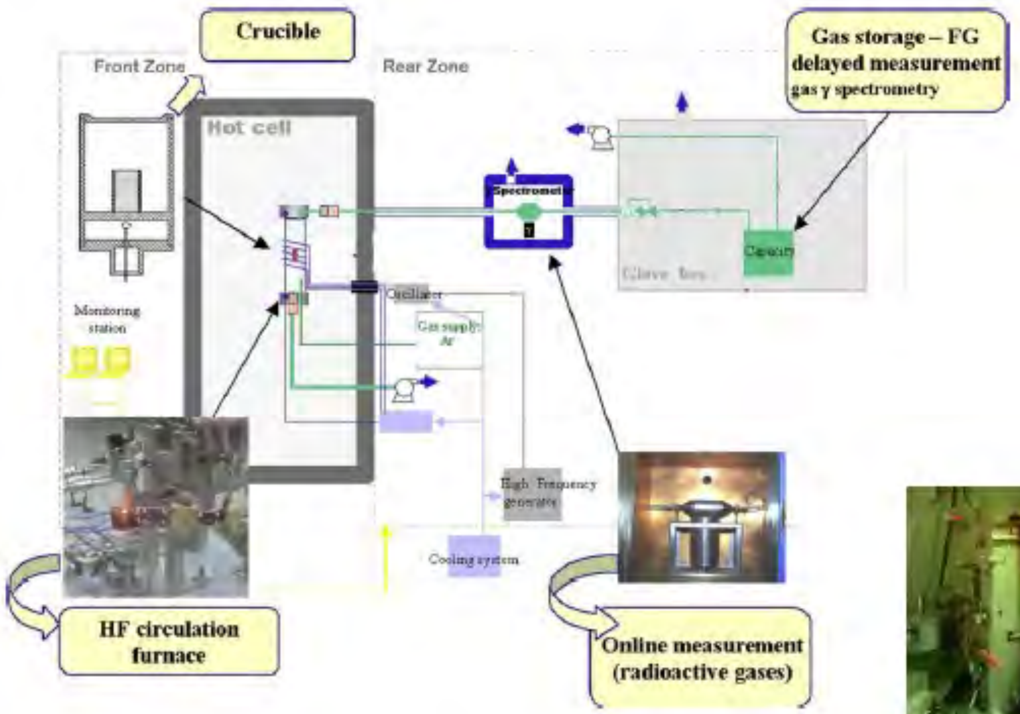
Irradiation in IFA-649: UO_2 based fuel discs



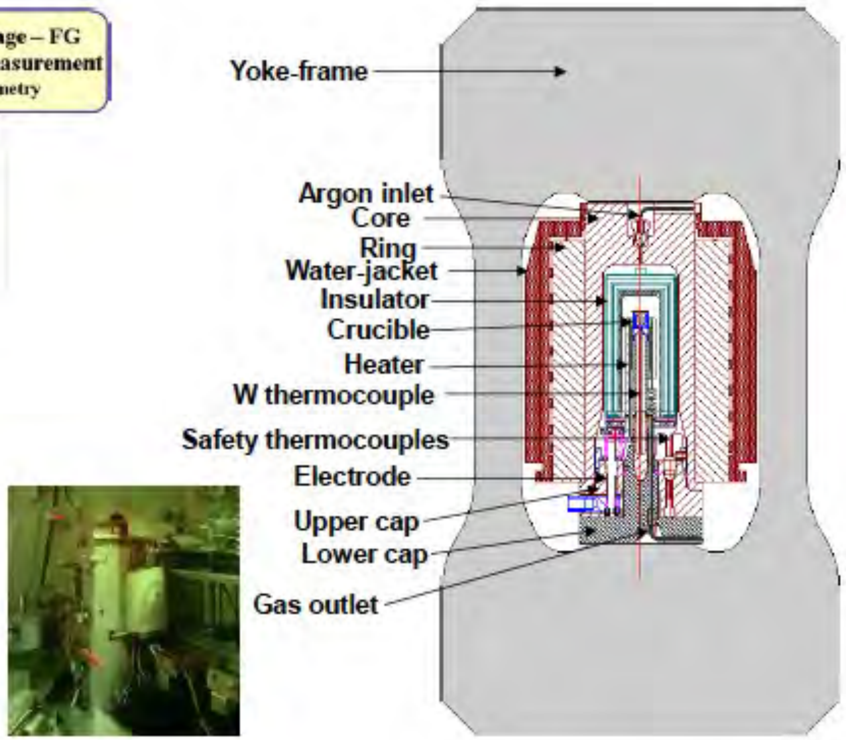
- Almost homogeneous burn-up and irradiation temp across fuel discs
- Fuel discs with no HBS, partial HBs, and full HBS microstructure discharged with a reactor scram to preserve 'at temperature' fuel characteristics
- Such unique fuel samples of known pedigree can not be retrieved from LWR rods
- Subsequent annealing performed in two hot cell devices (burn-up, ramp rate, external restraint) with simultaneous Kr-85 release measurements
- Extensive pre- and post- annealing characterizations, including particle size distribution

Fuel sample annealing T (t) using two hot cell devices

Pontillon Y. et al. Hotlab 2005



Hanus E. et al. TopFuel 2016



- MERARG : ~0.1 MPa, max 2800°C

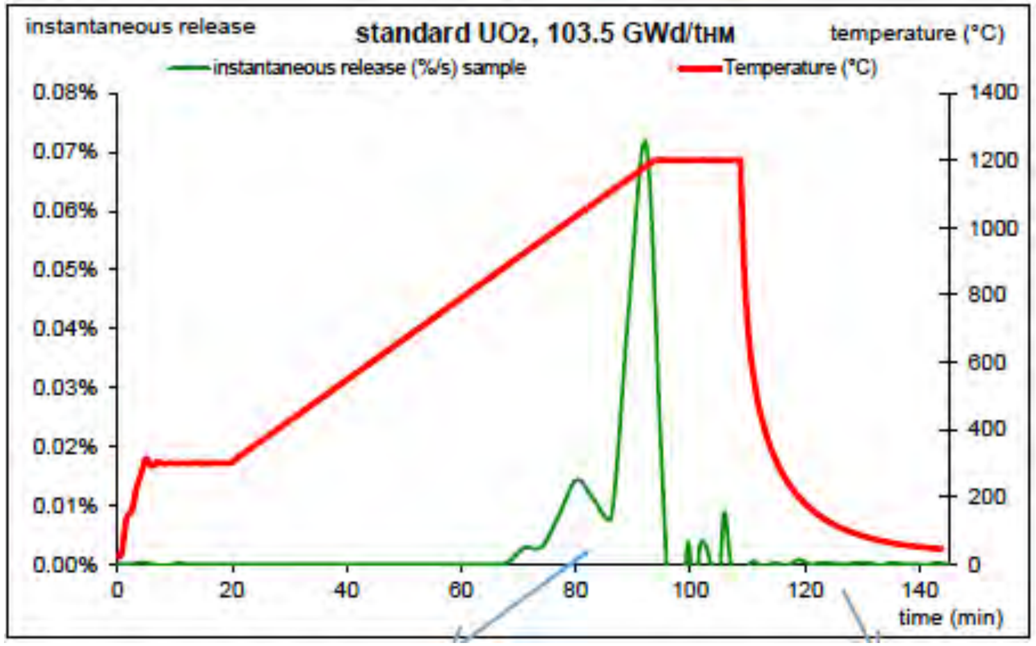
pellet size samples

- MEXIICO : max 160 MPa 1600°C

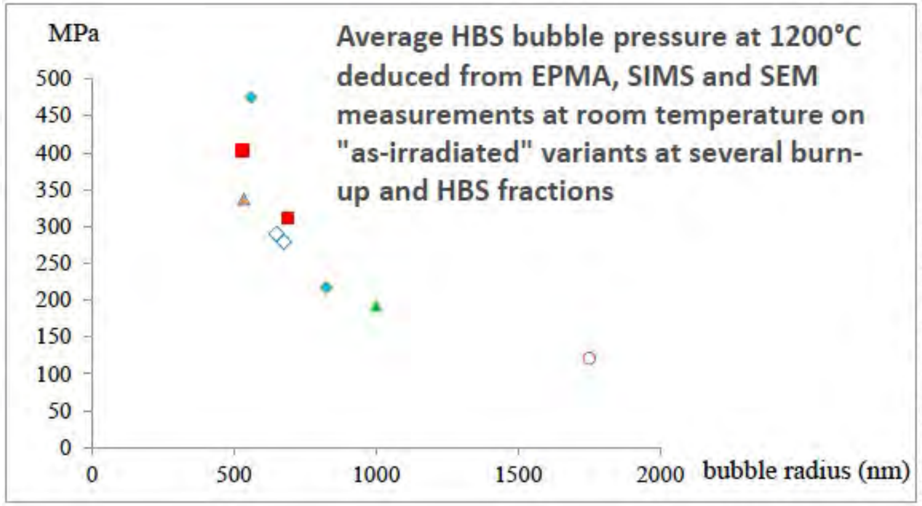
pellet size sample → unique capacities

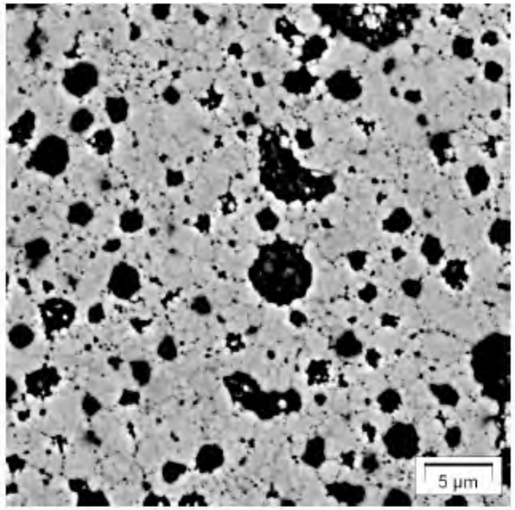
Mechanism and kinetics of gas release and fragmentation (1/4)

- Significant gas release occurs ~ 1200°C in high burn-up fuel
- The onset of fragmentation is related to the pressure reached in the small HBS bubbles



Measuring gas release upon heating

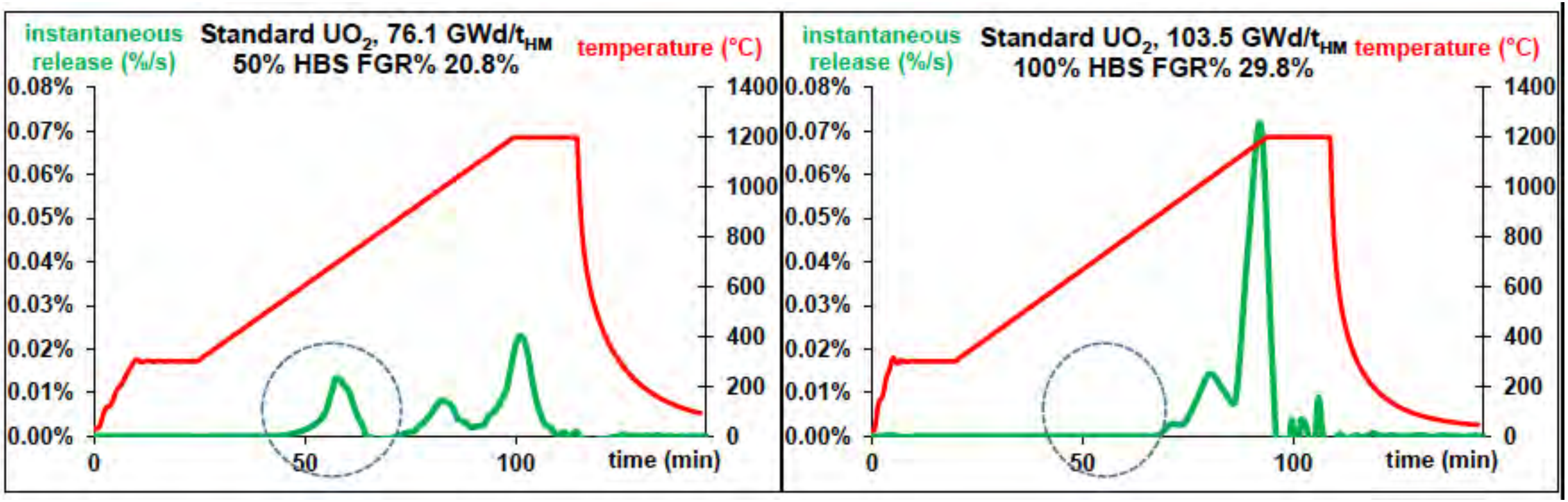




Estimation of pressure inside gas bubbles

Mechanism and kinetics of gas release and fragmentation (2/4)

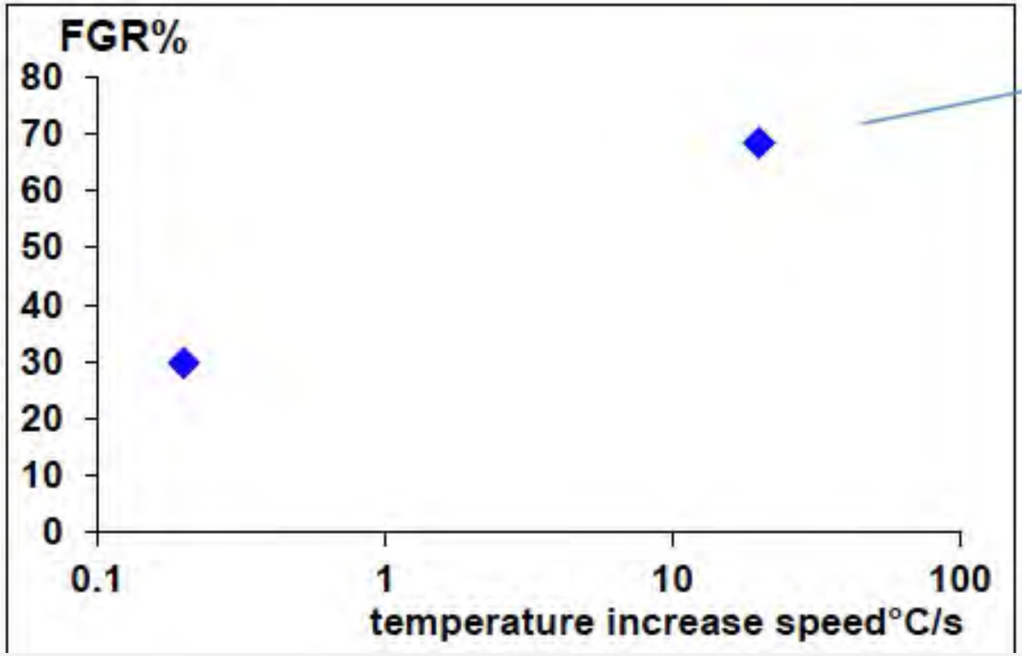
- Gas release begins at lower temperatures for the partially HBS transformed fuel than for fully transformed fuel

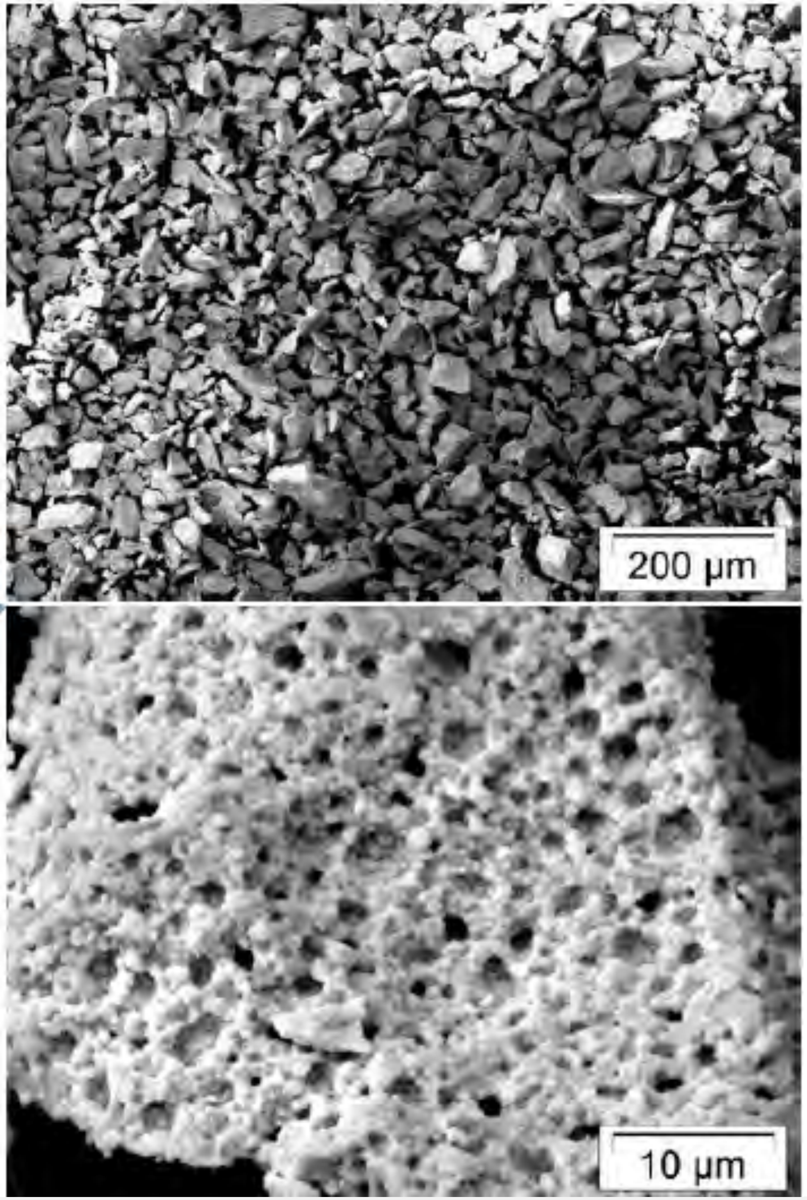


Mechanism and kinetics of gas release and fragmentation (3/4)

- Higher fission gas release and smaller fragments formed for a 20°C/s ramp rate than for a 0.2°C/s ramp rate

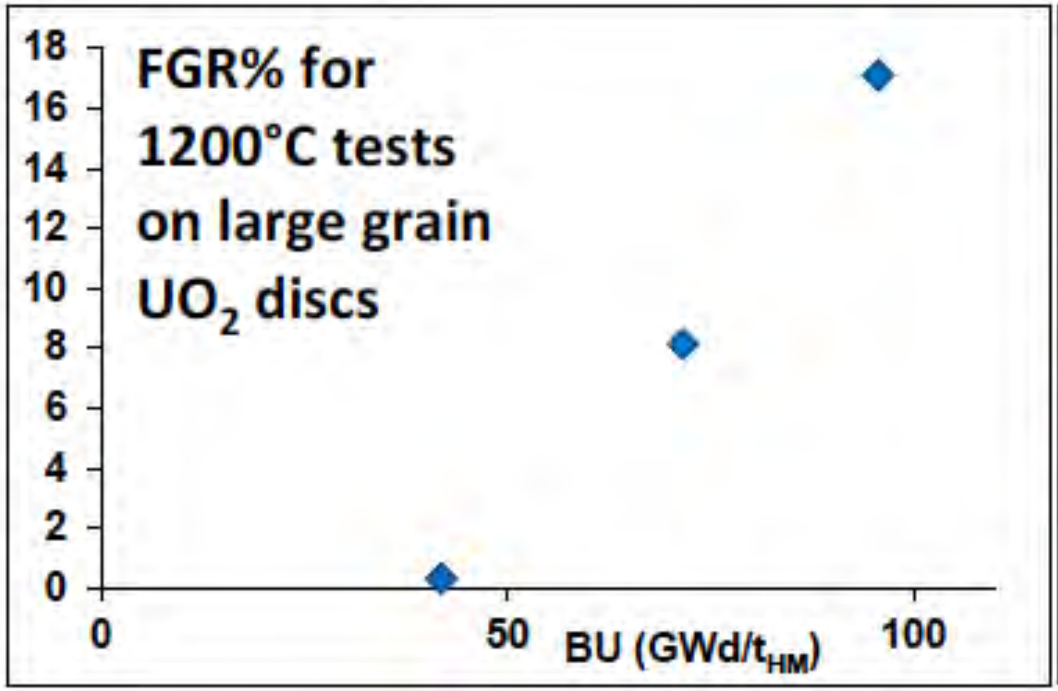
FGR% for 1200°C tests
on UO_2 discs at 0.2°C/s and 20°C/s

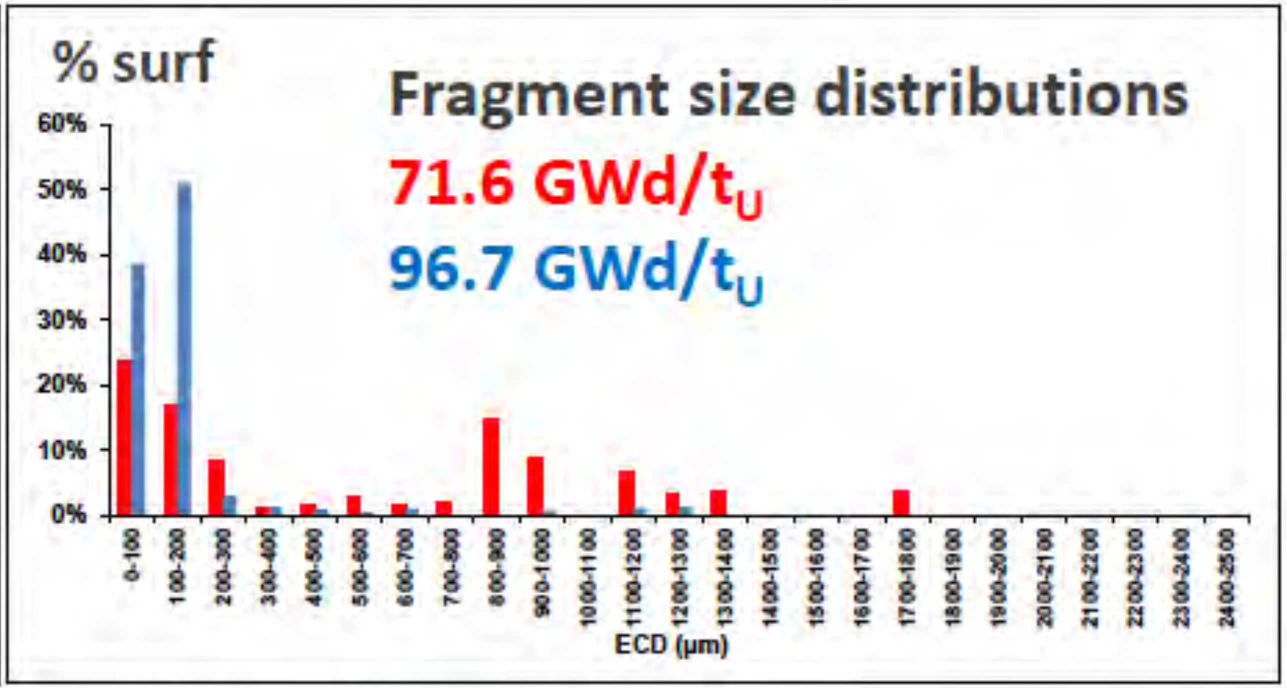




Mechanism and kinetics of gas release and fragmentation (4/4)

- Fuel with no HBS transformation showed no fragmentation in annealing, even up to 1600°C
- Fuel with partial and full HBS conversion fragmented during annealing to 1200°C
- The higher the HBS conversion, the higher the gas release and smaller the fragment size



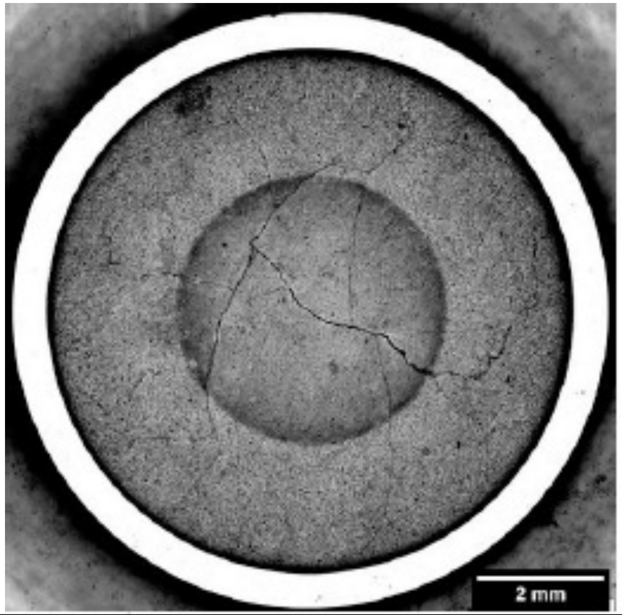


Summary and conclusions

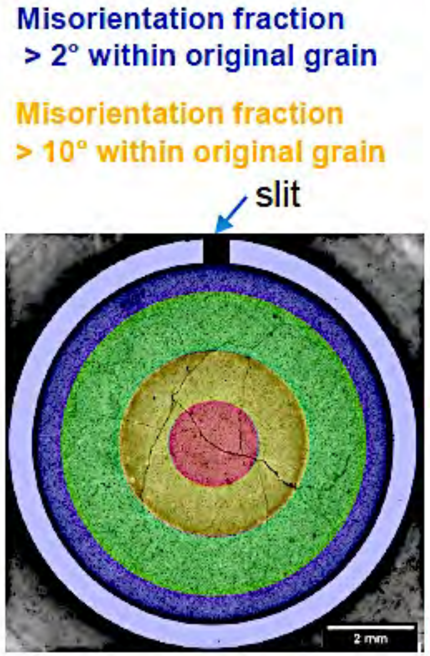
- Careful experimental work on almost homogeneous fuel material of known pedigree has been an efficient and cost-effective way to answer questions on fuel behavior in LOCA situations
- The initial scoping studies and subsequent detailed separate effects investigations on well-characterized fuel has provided fundamental insights into FF phenomenon
  - Effects of external restraint, % HBS transformation, estimated gas bubble pressures, and temp ramp rate on FF
- This work is still ongoing filling gaps in knowledge + PCMI/SCC works + burst release studies
- Plans are also underway to establish equivalence of IFA-649 fuel sample behavior with the behavior of HBS region of LWR fuel pellets
  - By harvesting samples from different radial zones of PWR fuel (next slide)

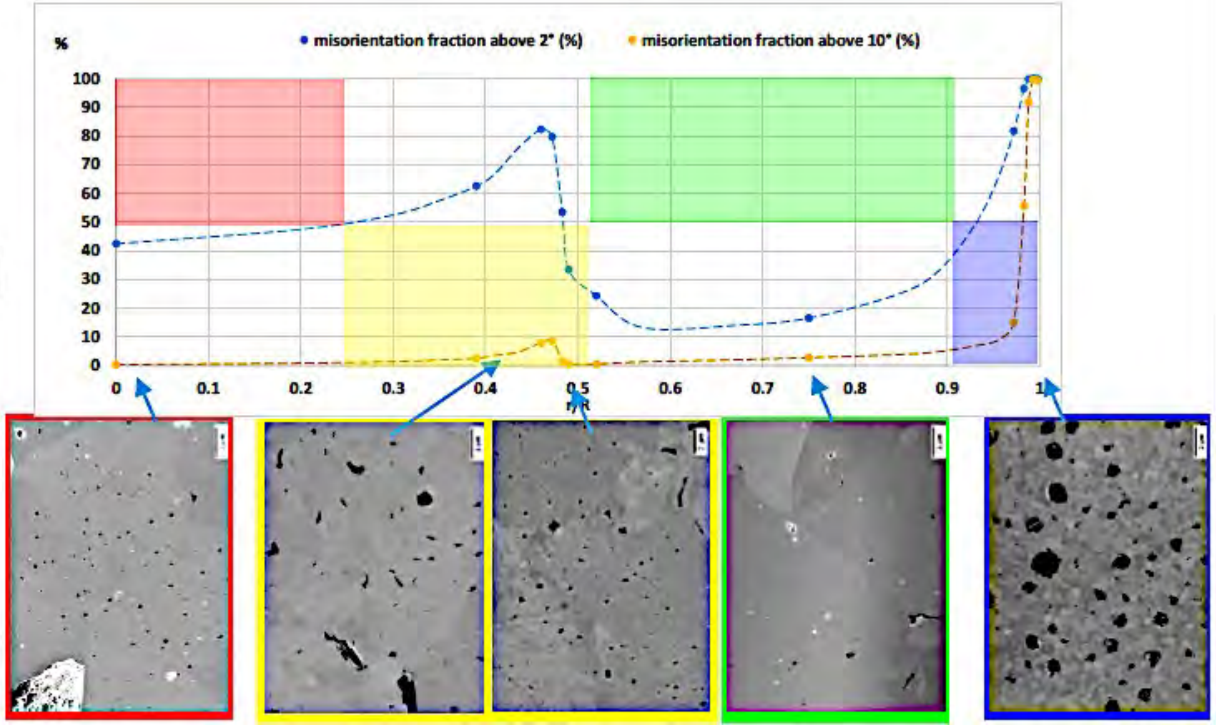
Which part of pellet contributes most to FF?

- Four axial-cores are prepared from two adjacent pellets (PWR fuel rod at high Bu)
  - Microstructure differences in each core characterized
  - Annealing performed separately on each core and fragment size distribution quantified



PWR fuel ceramograph at 7th span cross-section at local Bu 68.8 GWd/t





Microstructure of each zone characterized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Atomistic Modeling of Cladding Coating Behavior

Erich Wimmer and Mikael Christensen
Materials Design, Inc.

May 18, 2023

U.S. NRC ACRS Fuels, Materials, and Structures Subcommittee Meeting
Bethesda, MD

Outline

- Motivation and Objectives
- Key Results
- Modeling Approaches
- Computer Simulations:
  - Hydrogen ingress into Cr coating
  - Adhesion of Cr coating
  - Through-coating defects
  - Resistance of Cr coating to mechanical deformations
- Summary and Engineering Implications

Motivation and Objectives

- This modeling effort has been initiated in response to ATF PIRT in 2018
- To gain a deeper understanding of the effect of Cr coating of Zr cladding on hydrogen pickup
- To find possible weaknesses of Cr coatings of Zr alloys, e.g., enhanced oxidation and HPU caused by through-coating defects.

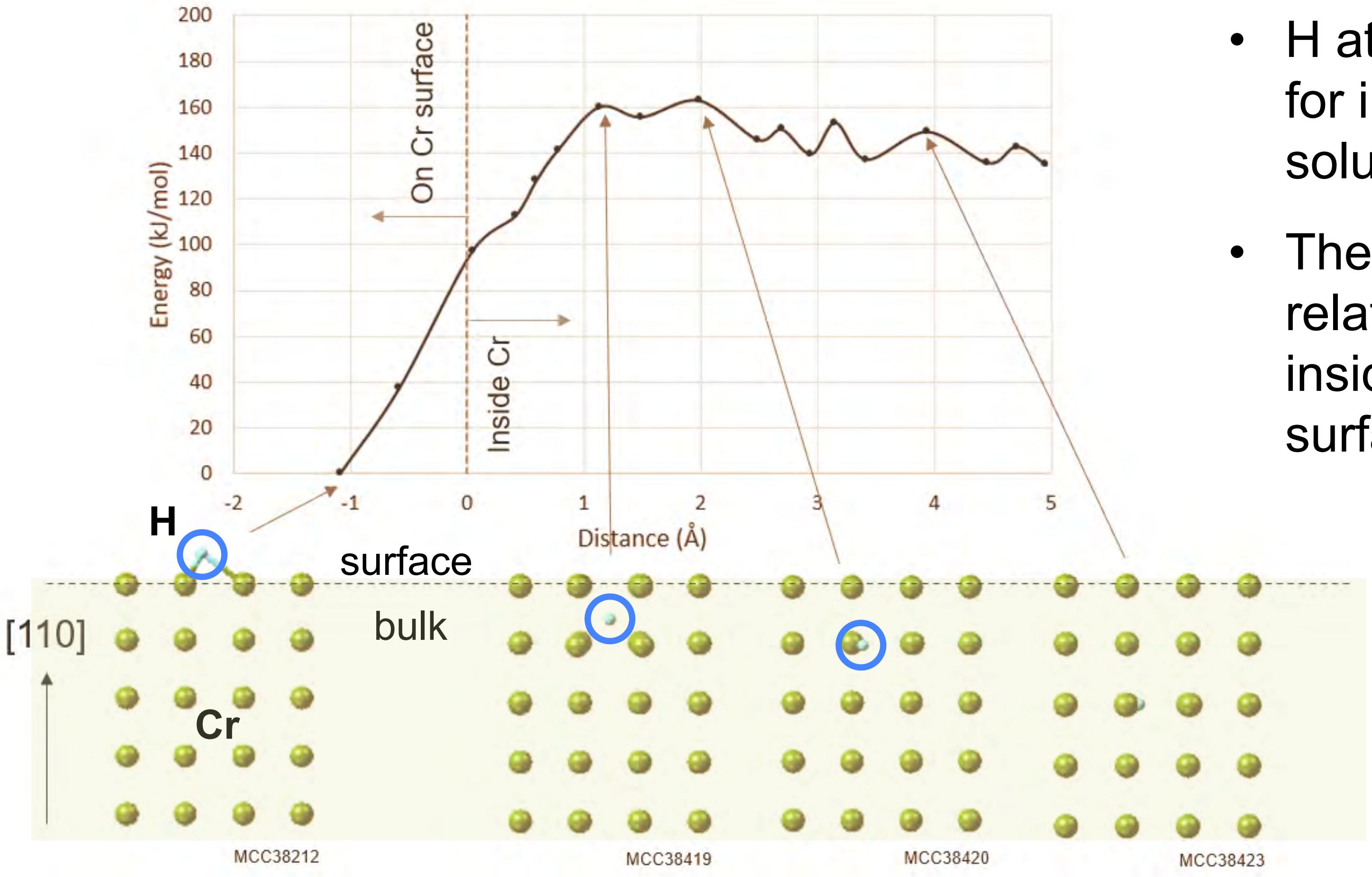
Key Results

- Cr and Cr-oxide (Cr_2O_3) are barriers for H ingress.
- Cr coatings bind very strongly to Zr surfaces and resist large mechanical deformations.
- Oxidation originating from through-coating defects is unlikely to spread much beyond this defect.

Modeling and Simulation Approaches

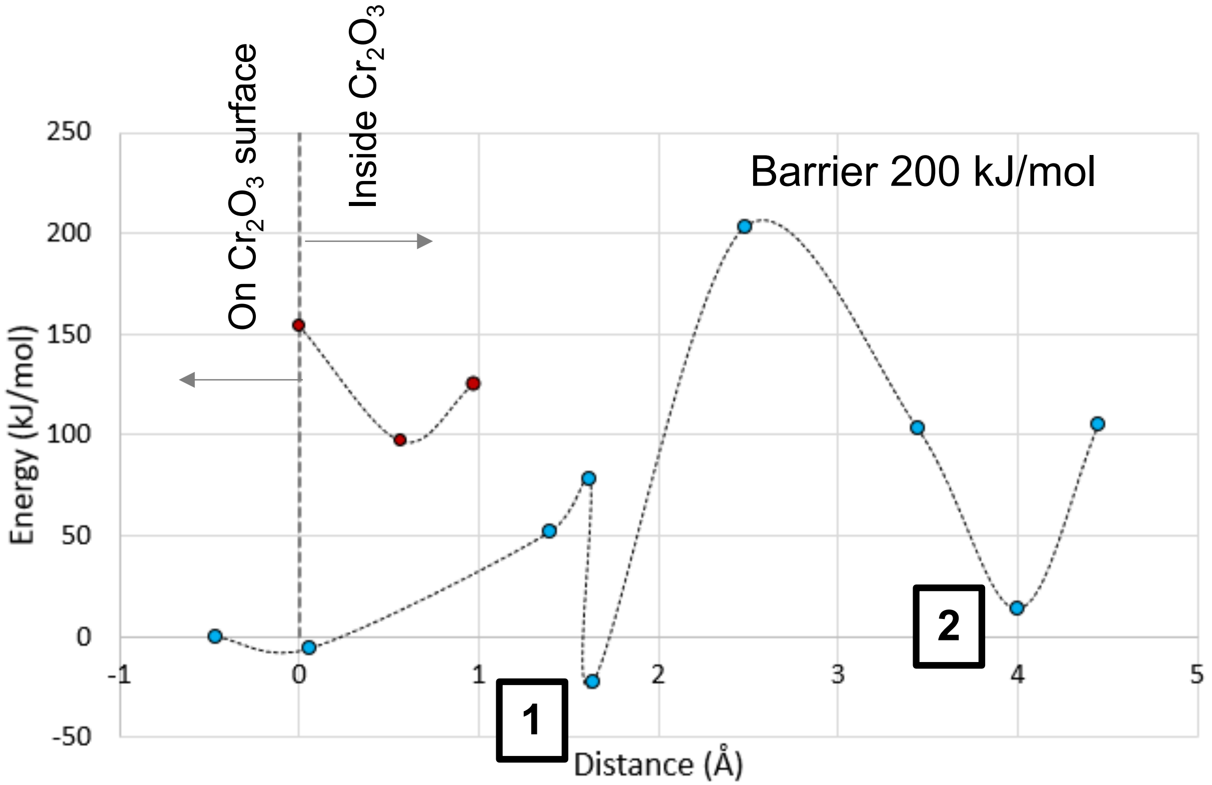
- First-principles quantum mechanical calculations
  - Prediction of thermodynamic, mechanical, and chemical material properties, e.g., interface energies
- Molecular dynamics simulations
  - Prediction of dynamic phenomena, e.g., diffusion, stress-induced plastic deformation, formation of dislocations, voids, and cracks
  - Use interatomic potentials derived from first-principles calculations employing machine learning methods
- Software
  - *MedeA* materials modeling environment with VASP for quantum mechanical computations and LAMMPS for large-scale molecular dynamics simulations.

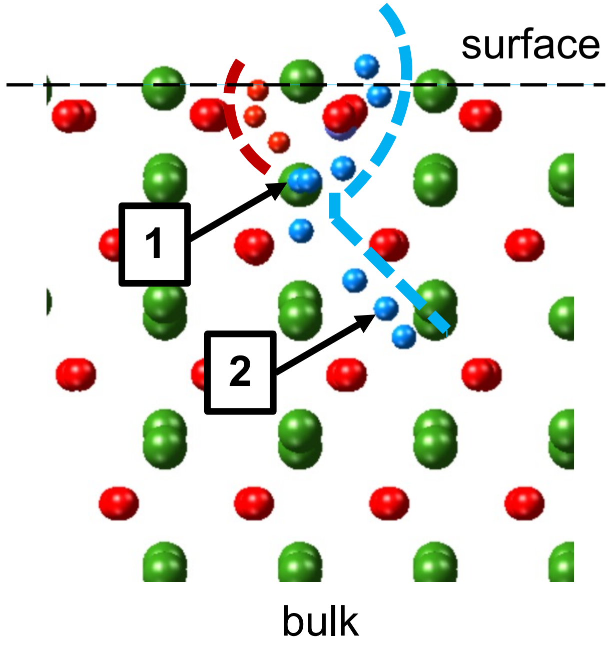
Hydrogen Ingress into Cr Coating



- H atoms face a high barrier of 160 kJ/mol for ingress into metallic Cr, i.e., the solubility of H in bulk Cr is low.
- The diffusion barriers for H inside Cr are relatively low (about 20 kJ/mol). If H were inside Cr, it would rapidly diffuse to surfaces.

Hydrogen Ingress into Cr_2O_3 Scales



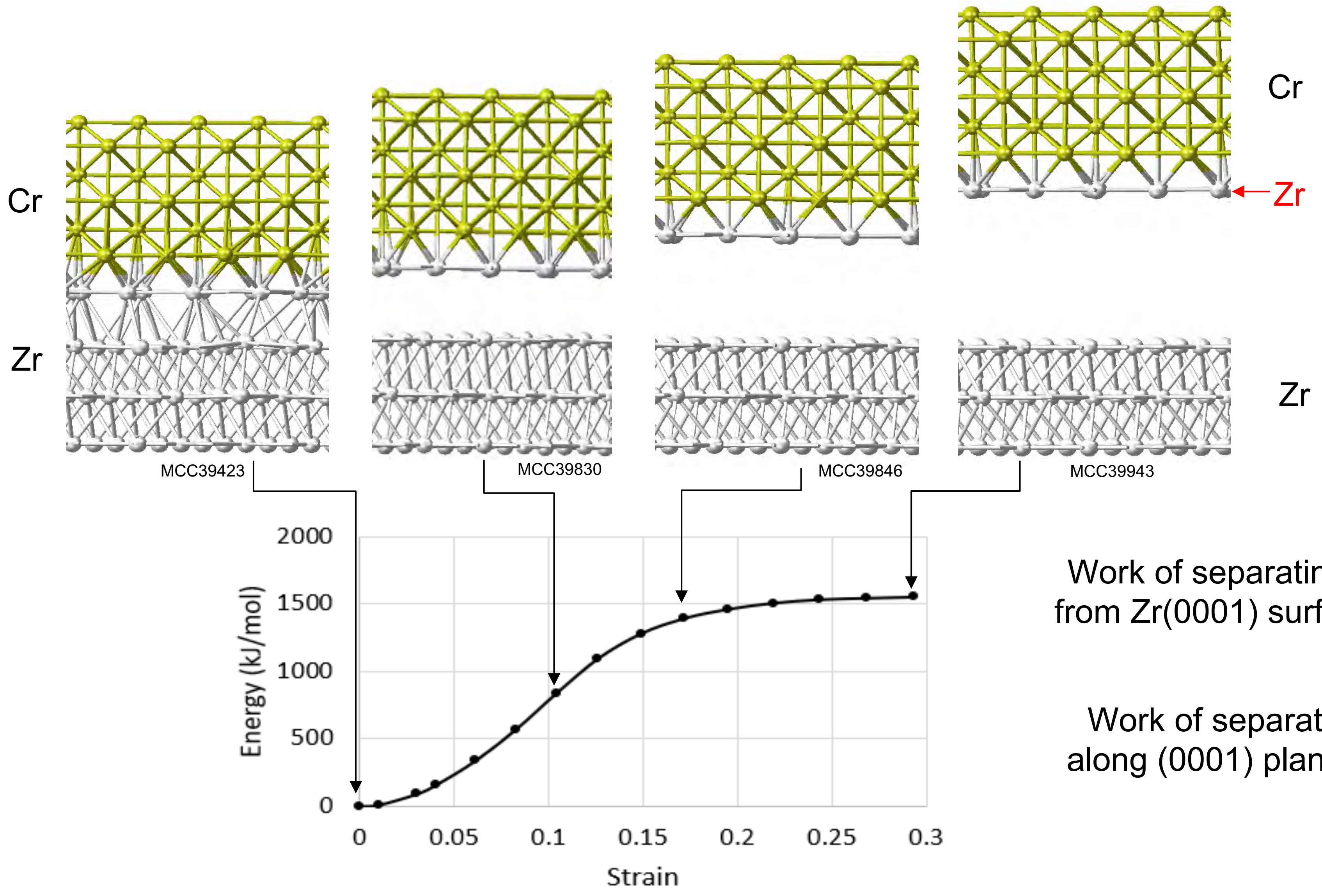


H atoms can be trapped in Cr_2O_3 and have high diffusion barriers (about 200 kJ/mol).

Both Cr metal and Cr_2O_3 are barriers for H ingress.

Due to the protective nature of Cr_2O_3, oxidation of Cr is slower than that of Zr, thus leading to lower H production and lower HPU of Cr coated Zr.

Chromium Coatings Bind Strongly to Zr Surfaces

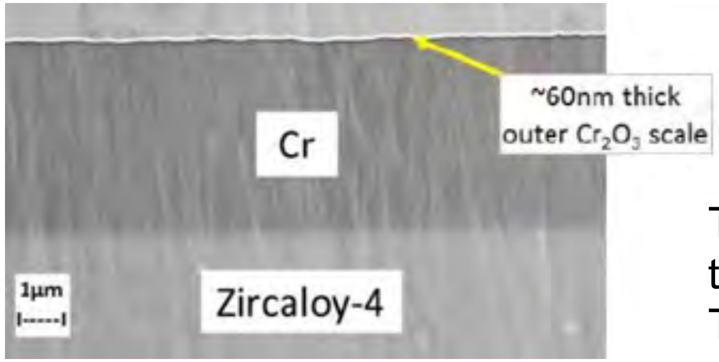


A monolayer of Zr remains on Cr coating, revealing a strong Zr-Cr bond.

Work of separating Cr coating from Zr(0001) surface: **2.6 J/m²**

Work of separating pure Zr along (0001) planes: **3.2 J/m²**

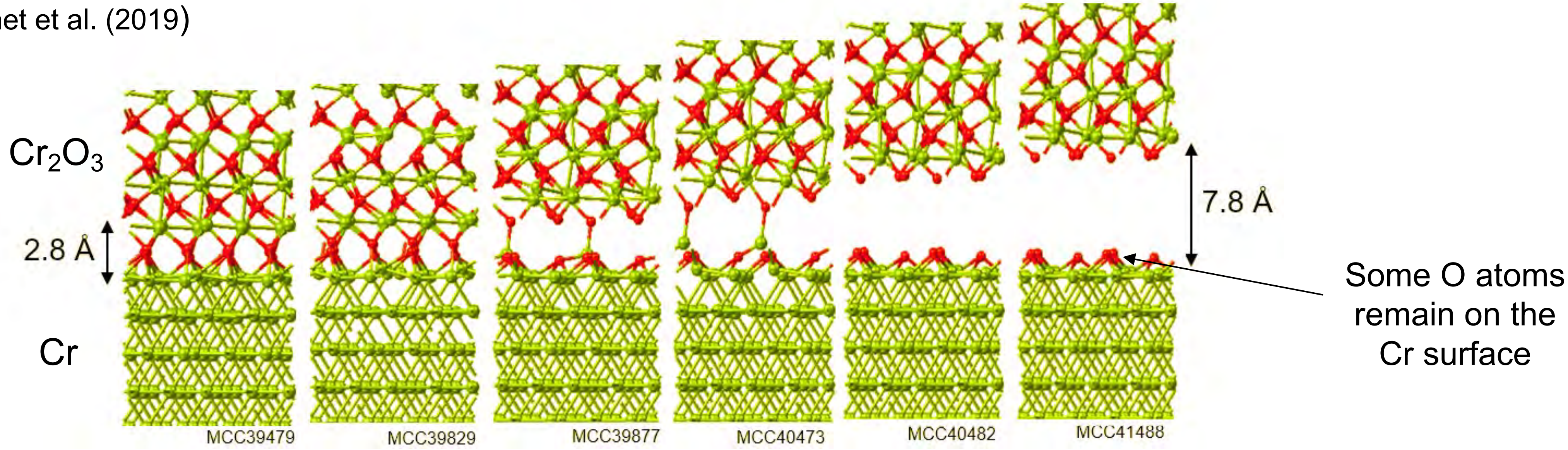
Chromia (Cr_2O_3) Adheres Strongly to Cr Surfaces



Images from Brachet et al. (2019)

$$2Cr + 3H_2O \rightarrow Cr_2O_3 + 6H$$

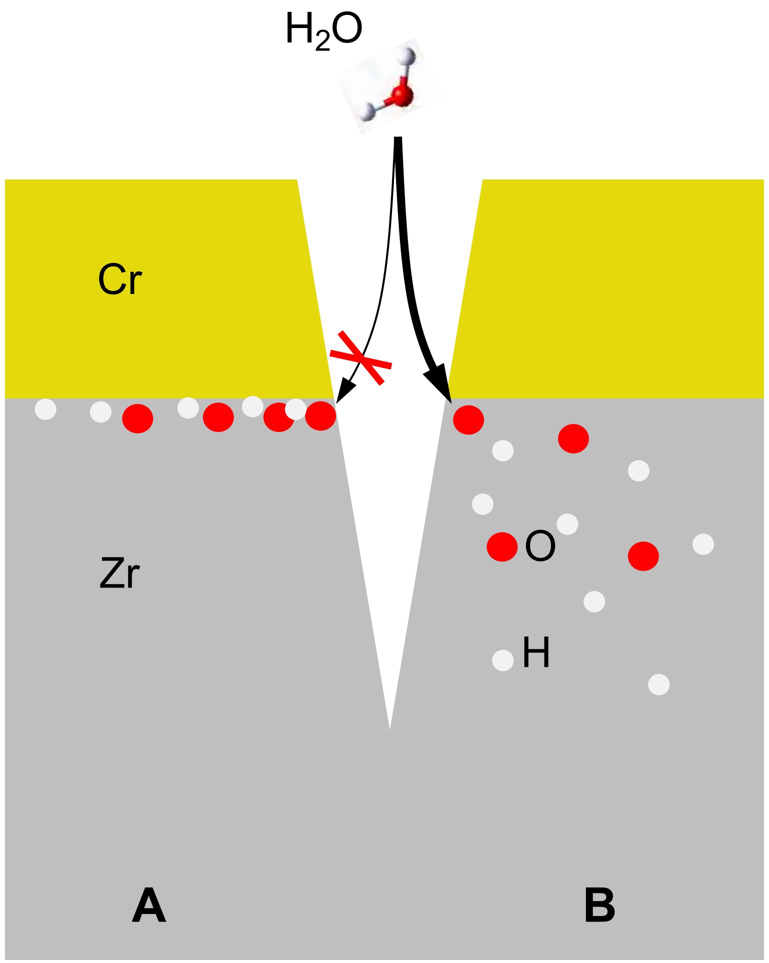
The surface of Cr is oxidized by water and covered with a thin Cr_2O_3 scale which adheres strongly to the Cr coating. The reaction of Cr with water releases H.



Work of separating Cr_2O_3 from Cr surface: **3.1 J/m²**

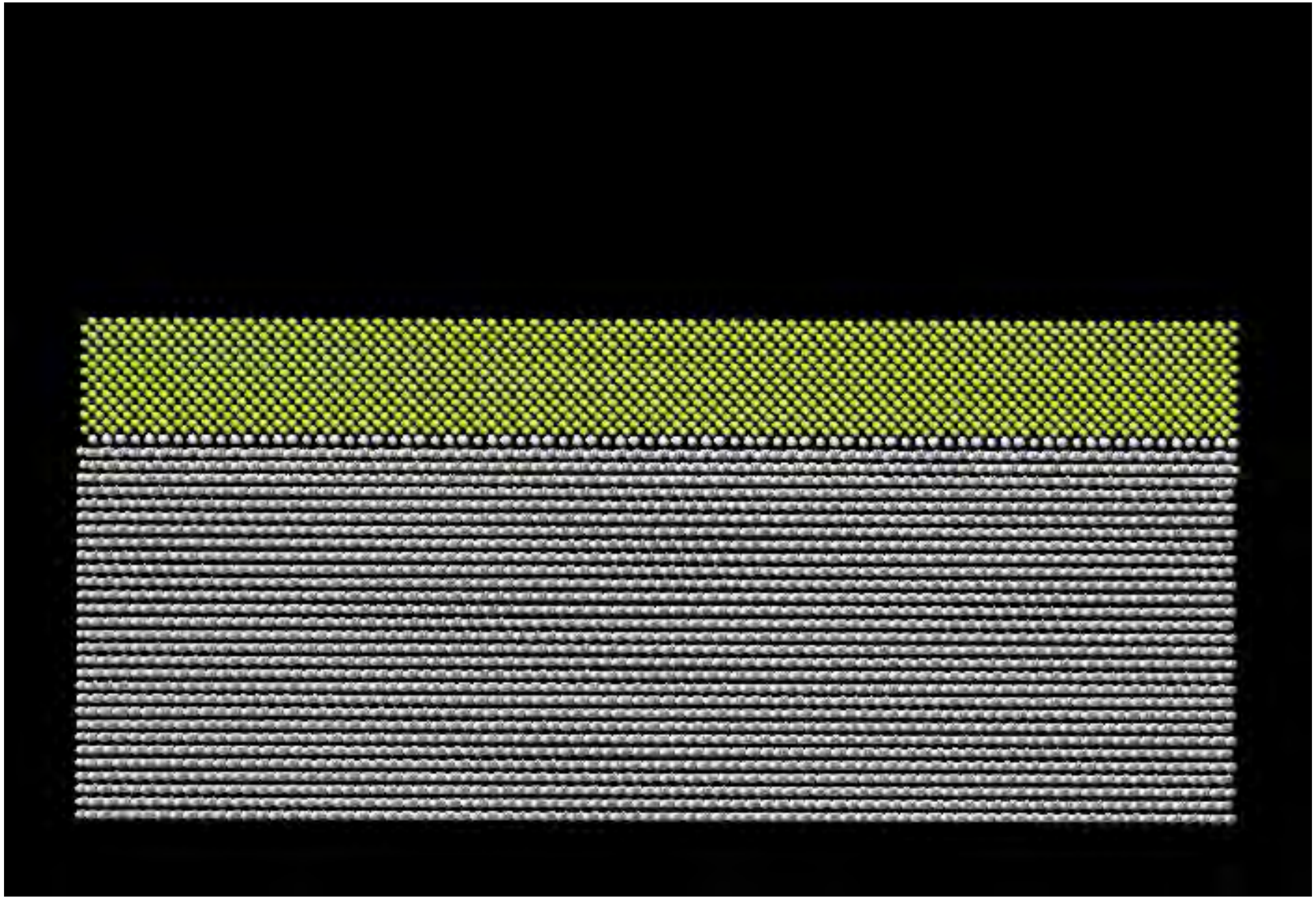
Comparison: Work of separating pure Zr along (0001) planes: **3.2 J/m²**

Through-Coating Defects



- Atomistic simulations show that water molecules preferentially dissociate on the Zr side of an exposed Cr/Zr interface.
- The reactivity of Zr at the Zr/Cr interface is similar to that on pure Zr without a Cr coating.
- **O and H atoms from water dissociation at an exposed Cr/Zr interface will absorb predominantly into bulk Zr (case B) rather than accumulate at the interface (case A).**
- O and H atoms are unlikely to diffuse into bulk Cr.
- Thermodynamics precludes the formation of Cr_2O_3 in the proximity of unoxidized Zr.
- Given a sufficient O source, O-saturated α-Zr at the base of a through-thickness Zr/Cr coating defect will transform into Zr oxides, as it would without a Cr coating. This oxidation process will create a local interface between Zr-oxides and Cr.

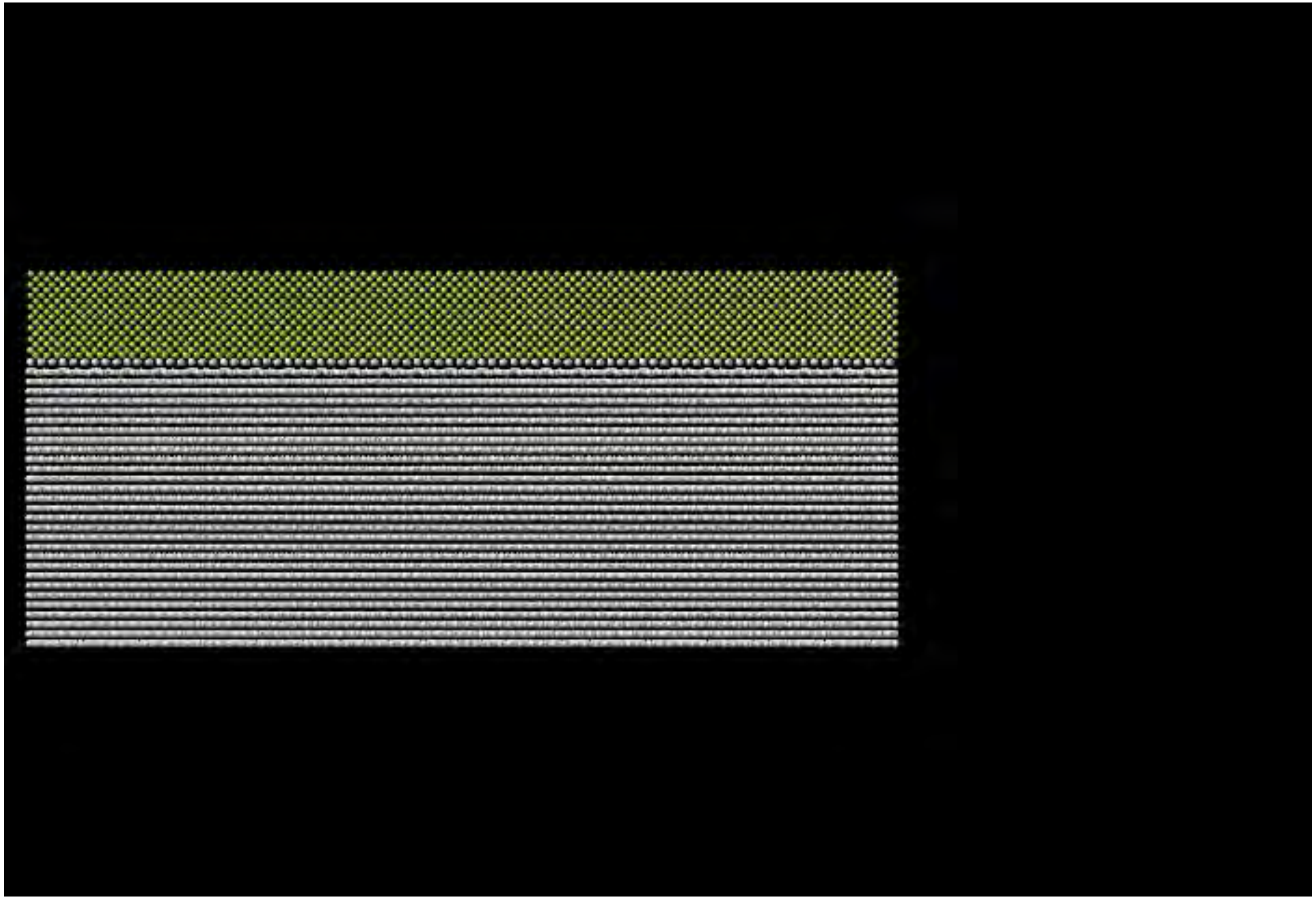
Mechanical Properties of Cr-Coating: Delamination



- Delamination of a Cr coating from a Zr substrate leaves a monolayer of Zr atoms on the Cr coating.

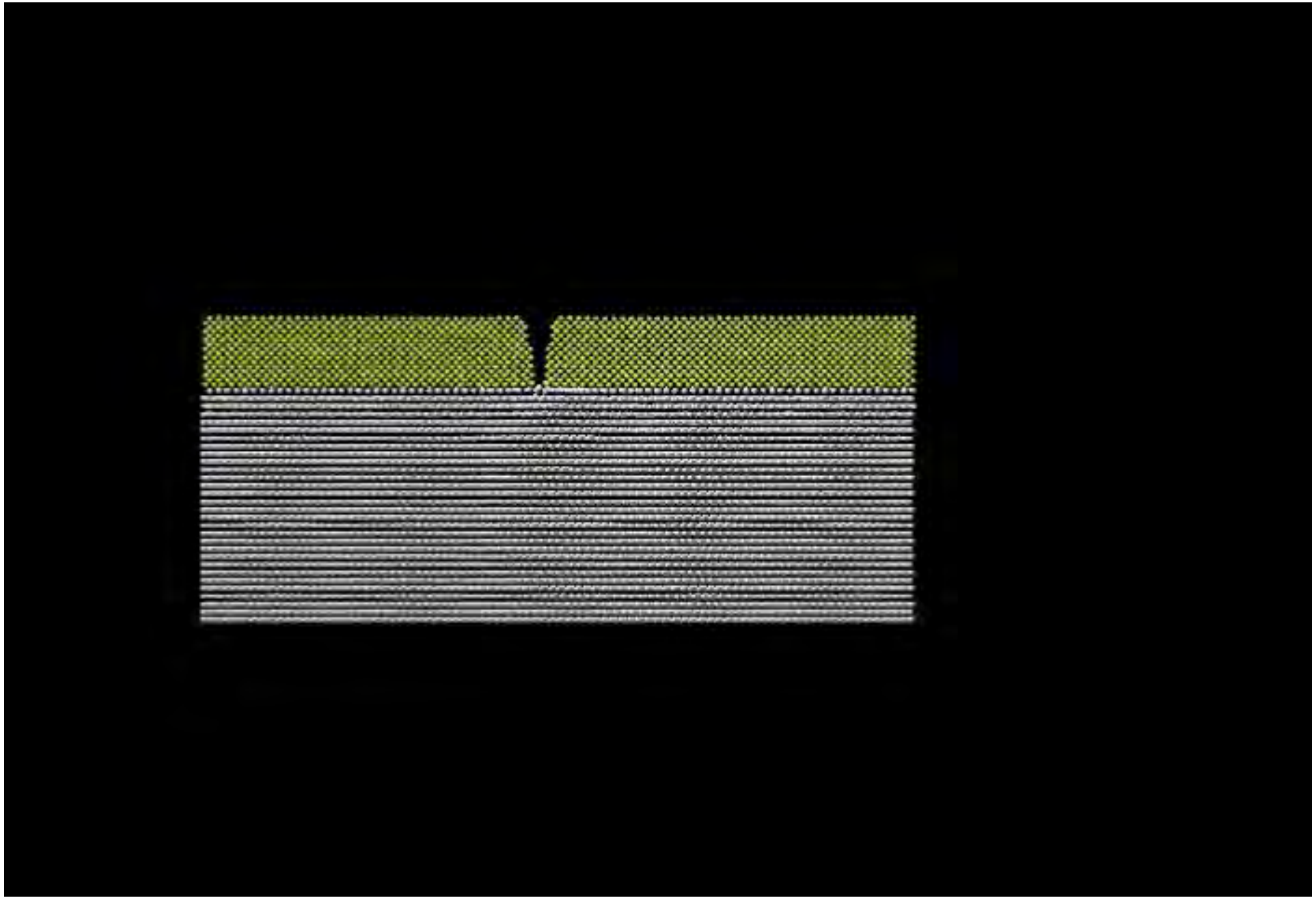
- Molecular dynamics simulation at 600 K using machine-learned interatomic potential trained on ab initio data.

Mechanical Properties of Cr-Coating: Lateral Strain



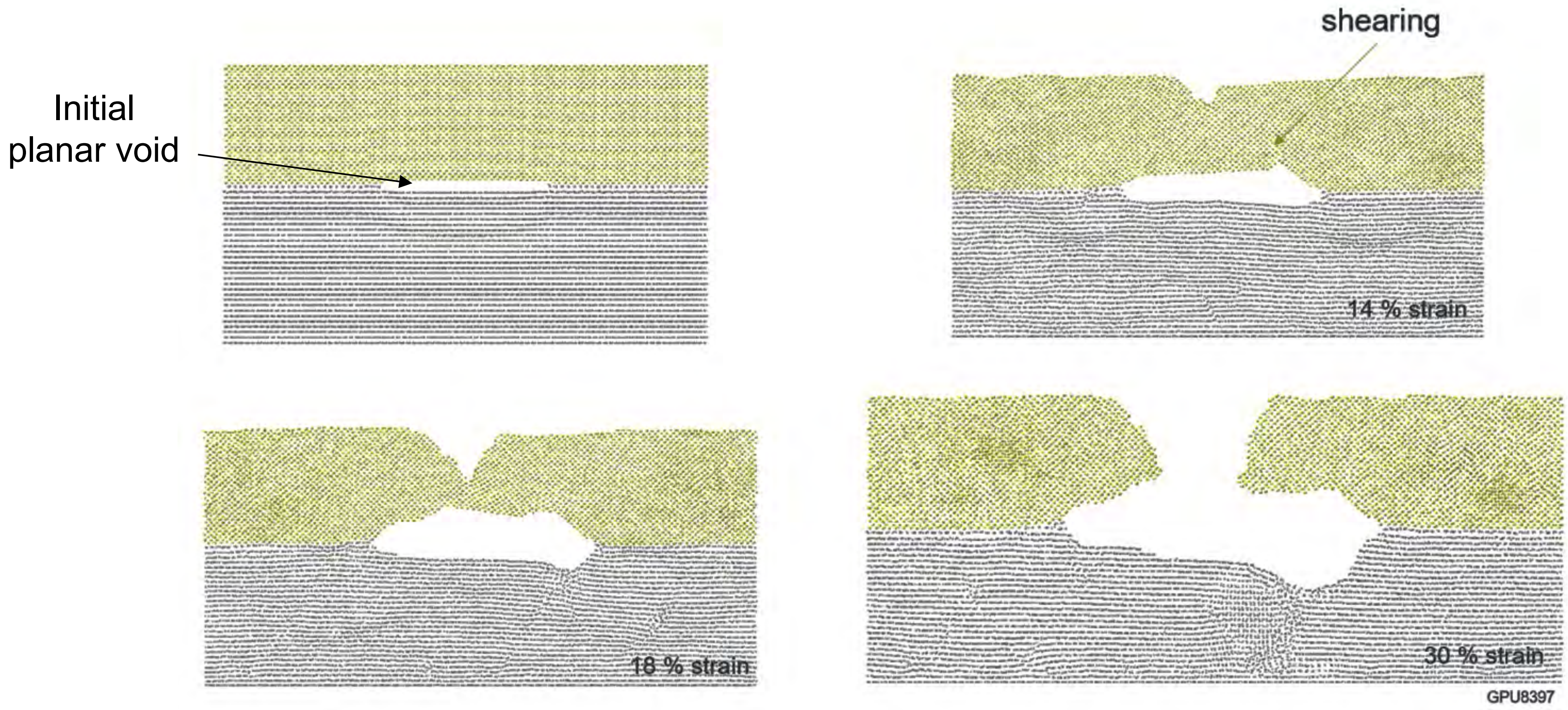
- Shearing inside the Cr coating occur almost simultaneously with shearing inside the Zr substrate.
- The Cr coating remains adherent up to large strains until through-coating gaps expose the underlying Zr.

Mechanical Properties of Cr-Coating with Crack



- Under lateral strain, a pre-existing crack in the Cr coating causes pitting of the exposed Zr substrate.

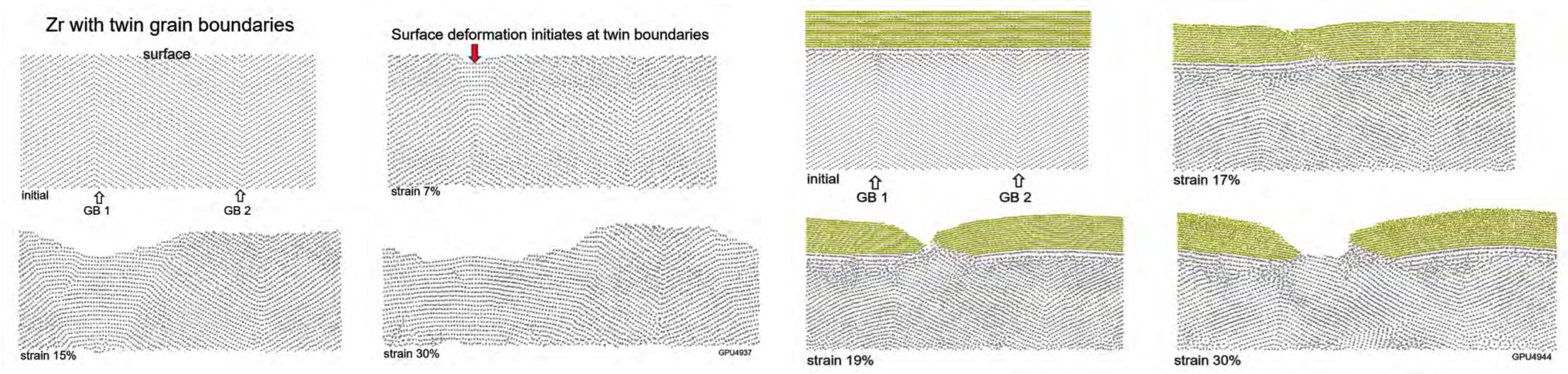
Effect of Voids at Interface



- Due to the initial planar void at the interface, a large area of Zr becomes exposed.
- The initial contact area of the Cr/Zr interface remains intact.

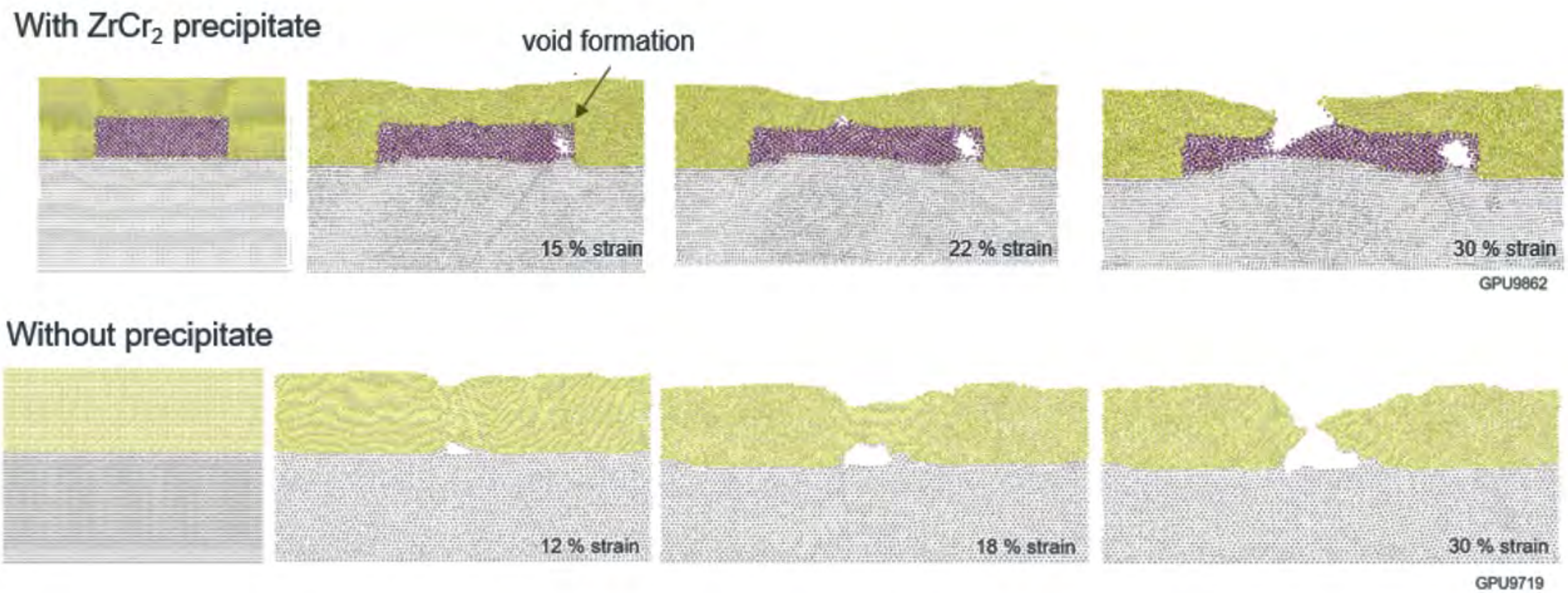
Effect of Tilted Zr Substrate

Suggested by Rob Daum



- A larger strain (17 % vs. 10 %) is required to initiate shearing with the tilted substrate compared to a Zr substrate with the (0001) planes parallel to the interface.
- The Zr substrate is exposed at similar strains for the tilted and parallel case (19 % vs. 18%)

Effect of Intermetallic $ZrCr_2$ Phase at Interface



- At 15% strain, a void forms inside the precipitate, further strain causes the formation of a second void which extends to the surface; finally, shearing inside the intermetallic phase leads to exposure of the Zr substrate.
- The reference simulation without the precipitate shows the formation and growth of a void at the interface which eventually exposes the Zr substrate.

Summary and Engineering Implications – Benefits of Cr (I)

The present atomistic simulations support the viability of Cr coating of Zr cladding in PWR's

- Cr coating reduces the rate of oxidation.
- A Cr_2O_3 scale and metallic Cr coating are barriers for H ingress.
- The adhesion of a Cr coating to Zr surfaces is very strong and resists large strains.
- Corrosion at through-coating defects is similar to that of un-coated Zr.
- Atoms of O and H created in a through-coating defect by reaction with water do not preferentially diffuse along the Cr-Zr interface but diffuse into bulk Zr; there is no significant driving force for oxide or hydride formation at Cr/Zr interfaces.
- The presence of O on the Zr substrate prior to Cr coating would not be expected to have dramatic effects on Cr adhesion.
- Cr coating adhesion to Zircaloy or Zr-Nb alloy substrates is likely to be equally robust.

Summary and Engineering Implications – Benefits of Cr (II)

- The presence of $ZrCr_2$ precipitates at the Cr/Zr interfaces has no major impact on the resilience of the coating.
- Tilting of the basal plane of crystalline Zr substrates by 30° relative to the surface normal increases the strain at which shearing in the Cr coating occurs. Once initiated, exposure of the underlying Zr substrate proceeds faster compared with un-tilted substrate grains. However, the strain to exposure is similar in both cases.
- Under very large tensile in-plane strains, voids form close to the Cr/Zr interface inside the Cr coating. The present simulations show that the effect of these voids remains localized and is unlikely to lead to spallation and decohesion.
- Compressive pre-straining was found to make a Cr coating more resistant to tensile stresses that may develop during service.

Potential Engineering Concerns

- Formation of zirconium hydrides at the Cr/Zr interface weakens its strength.
- Precipitation of $ZrFe_2$ at the Cr/Zr interface may increase the brittleness of the interface.
- Pre-existing voids at the Cr/Zr interface may lead to pitting and thus expose large areas of the Zr substrate.
- Above the eutectic temperature (2430 °F) the Cr coating will be dissolved. This temperature is much lower than the melting temperature of pure Zr (3371 °F).
- The behavior of Cr coating under irradiation is unclear and has not yet been investigated by simulations.

Acknowledgements

This work represents the product of a team effort, and I want to thank the EPRI participants for their support and advice, and all my colleagues at Materials Design, especially

Volker Eyert

Clive Freeman

Clint Geller

Benoit Minisini

Walter Wolf

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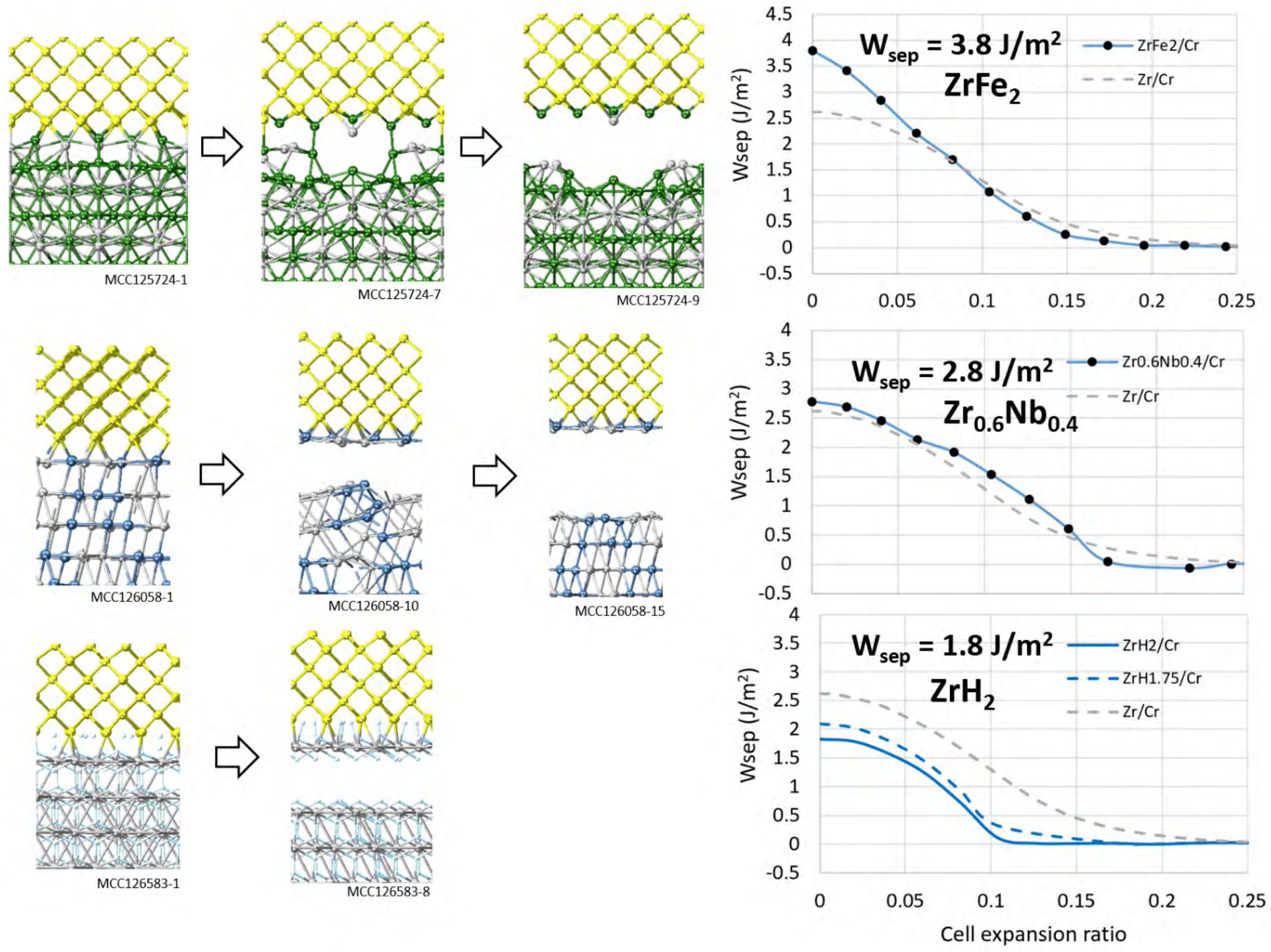
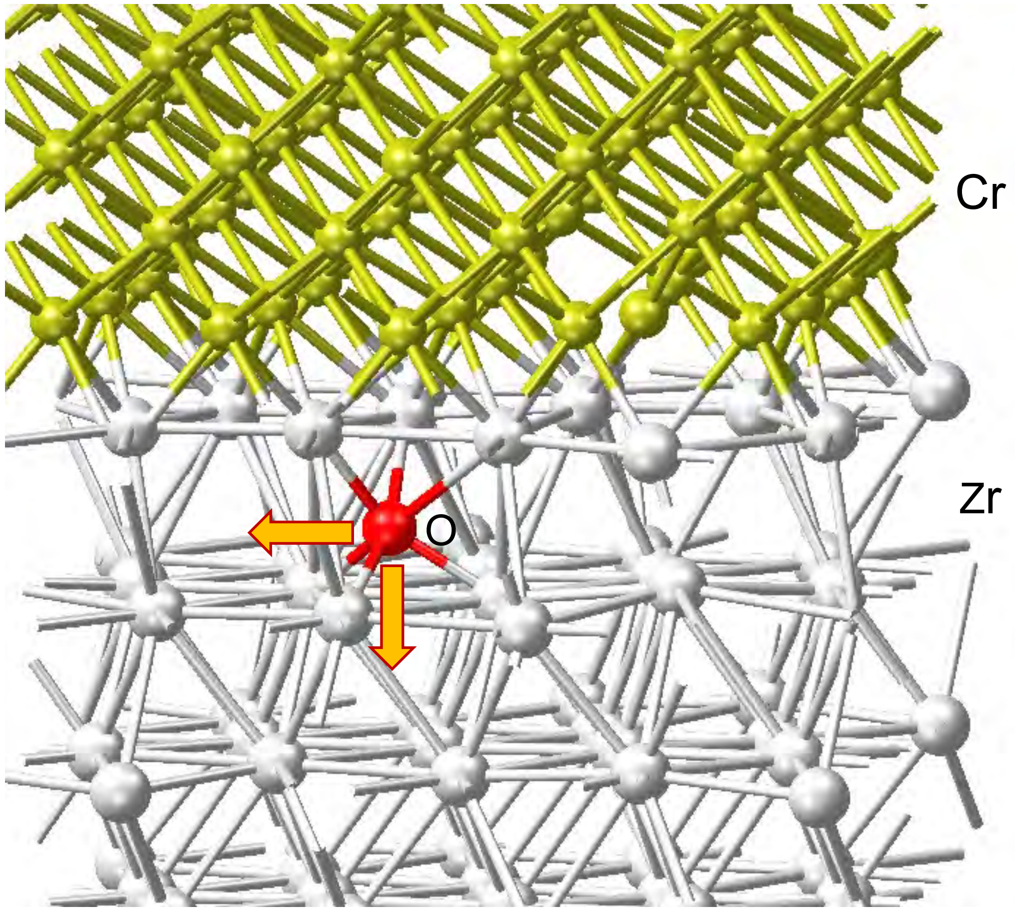
BACKUP SLIDES

Effect of Precipitates on Interface Strength



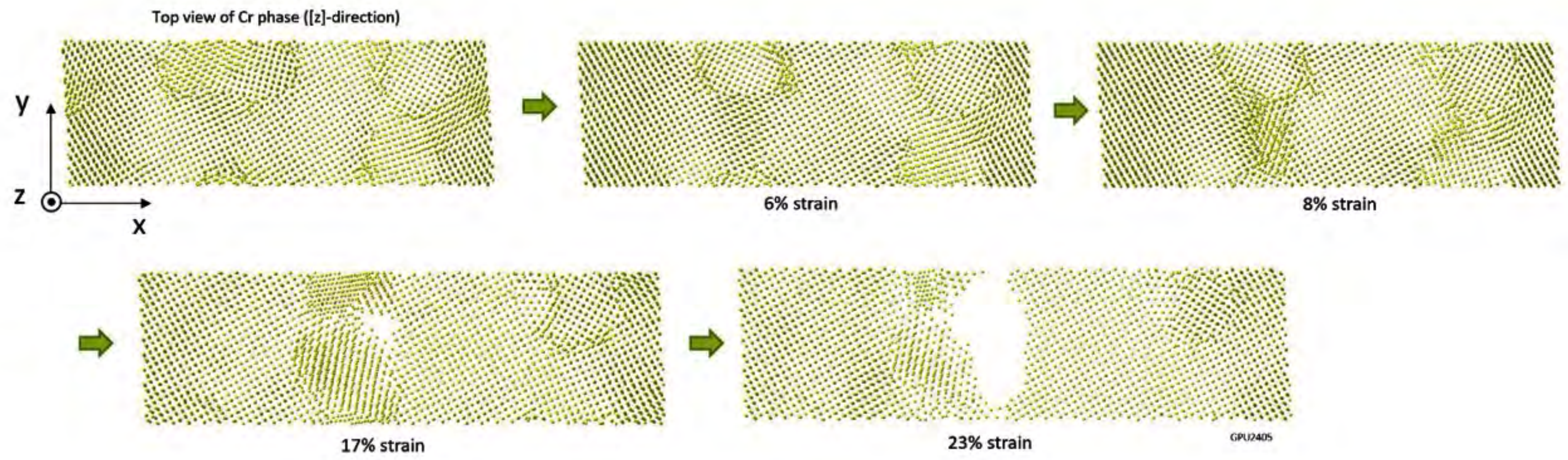
- Precipitation of $ZrFe_2$ strengthens the interface, but potentially introduces brittleness
- β-phase Nb precipitates have little effect
- Zr hydrides could be detrimental

Oxygen at the Cr/Zr Interface



- Oxygen atoms are thermodynamically more stable in bulk Zr than at a Zr/Cr interface.
- The diffusivity of O parallel to a Zr/Cr interface is substantially the same as that of O in bulk Zr.
- Hence, O atoms entering Zr from a through-coating defect primarily will diffuse into the bulk Zr phase, rather than accumulating at the interface.

Effect of Strain on Columnar Cr Grains



- Subject to tensile strain, a model with columnar grains (shown above) expose the underlying Zr at a strain of 17%, which is similar to that of a monocrystalline Cr coating.

Collaborative Research on Advanced Fuel Technologies for LWRs (CRAFT)

Kurshad Muftuoglu
Technical Executive

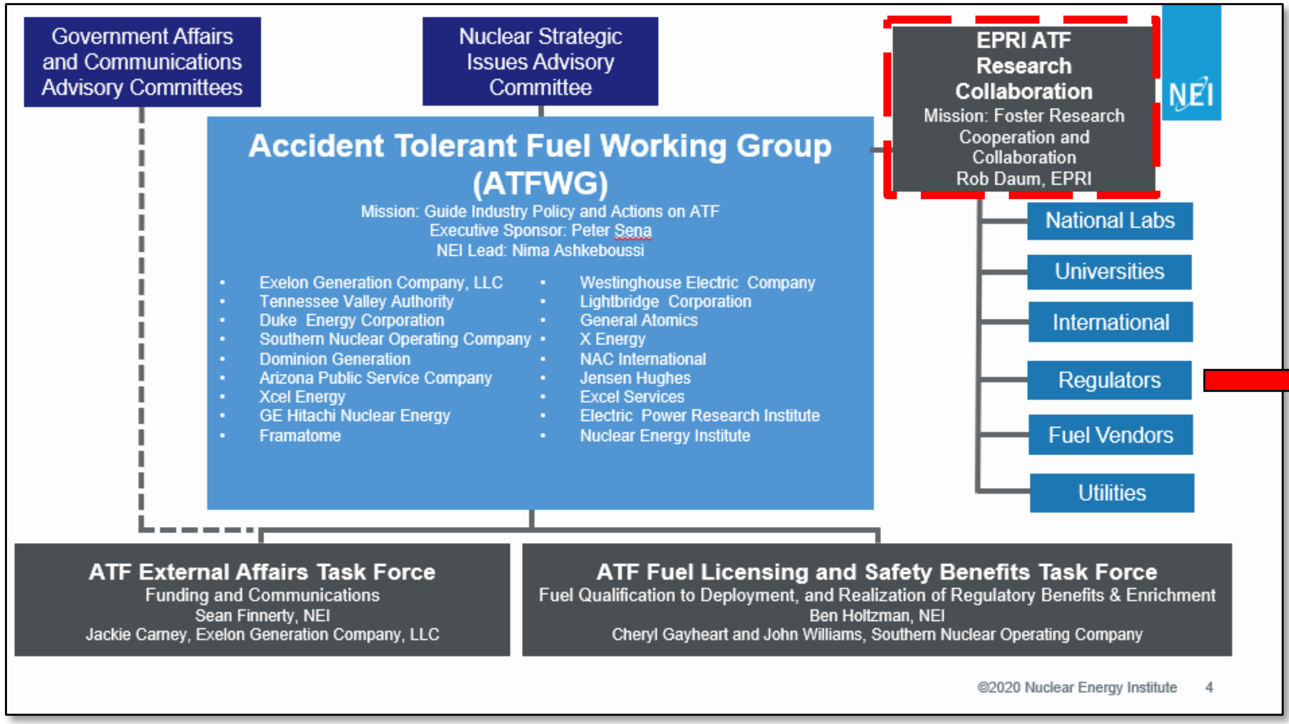
May 18, 2023

ACRS Fuel, Materials, and Structures Subcommittee Meeting

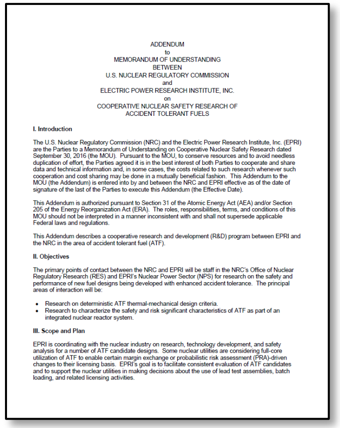
Collaborative Research on Advanced Fuel Technologies for LWRs (CRAFT) Framework

CRAFT Mandate and Purpose

- Foster research cooperation and collaboration
  - Bring technical subject matter experts from all stakeholders together
  - Present deliverables to optimize R&D resources and accelerate timelines toward licensing submittals and regulatory reviews
- Emulate the Extended Storage Collaboration Program (ESCP) on dry storage issues



EPRI-NRC/RES
MOU Addendum on ATF



ADDENDUM
to
MEMORANDUM OF UNDERSTANDING
BETWEEN
U.S. NUCLEAR REGULATORY COMMISSION
and
ELECTRIC POWER RESEARCH INSTITUTE, INC.
on
COOPERATIVE NUCLEAR SAFETY RESEARCH OF
ACCIDENT TOLERANT FUELS

I. Introduction

The U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute, Inc. (EPRI) are the Parties to a Memorandum of Understanding on Cooperative Nuclear Safety Research dated September 30, 2016 (the MOU). Pursuant to the MOU, to conserve resources and to avoid needless duplication of effort, the Parties agreed it is in the best interest of both Parties to cooperate and share data and technical information and, in some cases, the costs related to such research whenever such cooperation and cost sharing may be done in a mutually beneficial fashion. This Addendum to the MOU (the Addendum) is entered into by and between the NRC and EPRI effective as of the date of signature of the last of the Parties to execute this Addendum (the Effective Date).

This Addendum is authorized pursuant to Section 31 of the Atomic Energy Act (AEA) and/or Section 205 of the Energy Reorganization Act (ERA). The roles, responsibilities, terms, and conditions of this MOU should not be interpreted in a manner inconsistent with and shall not supersede applicable Federal laws and regulations.

This Addendum describes a cooperative research and development (R&D) program between EPRI and the NRC in the area of accident tolerant fuel (ATF).

II. Objectives

The primary points of contact between the NRC and EPRI will be staff in the NRC's Office of Nuclear Regulatory Research (RES) and EPRI's Nuclear Power Sector (NPS) for research on the safety and performance of new fuel designs being developed with enhanced accident tolerance. The principal areas of interaction will be:

- Research on deterministic ATF thermal-mechanical design criteria.
- Research to characterize the safety and risk significant characteristics of ATF as part of an integrated nuclear reactor system.

III. Scope and Plan

EPRI is coordinating with the nuclear industry on research, technology development, and safety analysis for a number of ATF candidate designs. Some nuclear utilities are considering full-core utilization of ATF to enable certain margin exchange or probabilistic risk assessment (PRA)-driven changes to their licensing basis. EPRI's goal is to facilitate consistent evaluation of ATF candidates and to support the nuclear utilities in making decisions about the use of lead test assemblies, batch loading, and related licensing activities.

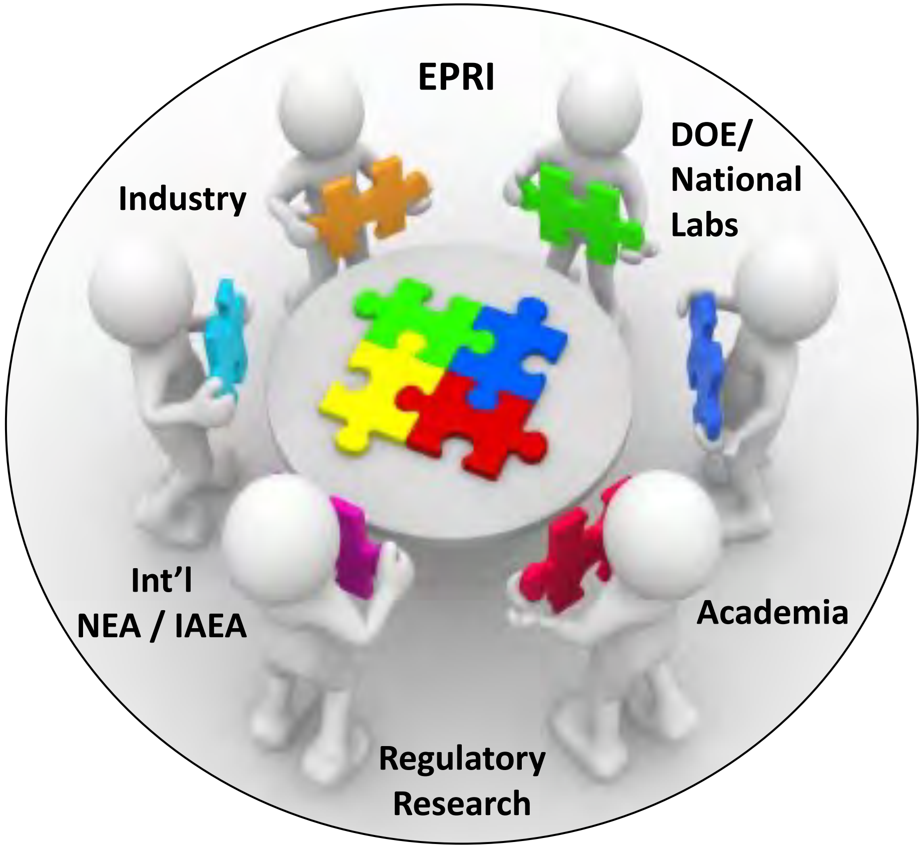
Objectives

1. **Bring together subject matter experts** from U.S. organizations, and when appropriate international organizations.

2. **Identify both short and long-term technical options and recommendations** for supporting the highest priority RD&D needs.

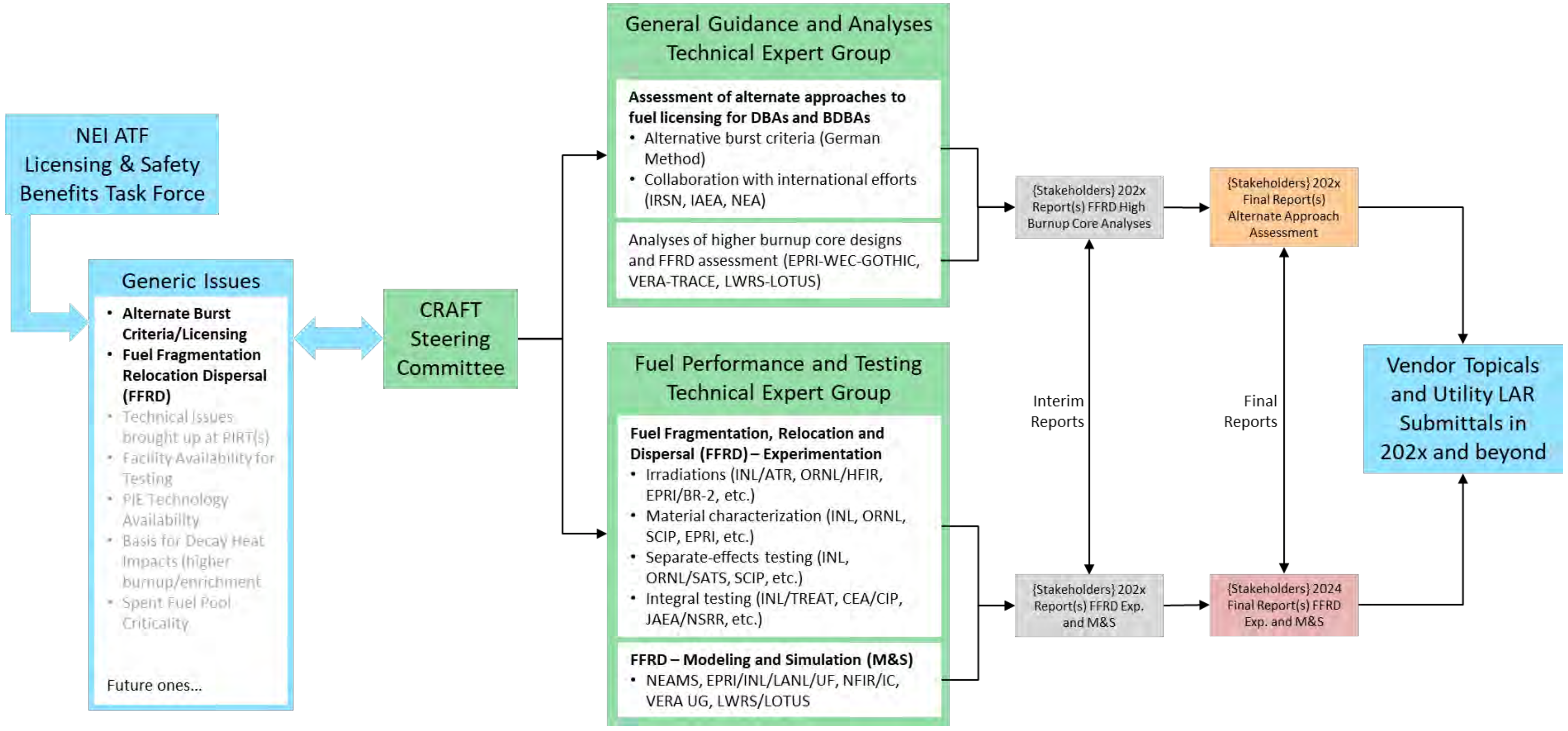
3. **Support gap analyses** and/or **Phenomena Identification and Ranking Table (PIRT)** processes.

4. **Compile, analyze and synthesize generic RD&D results to form technical bases** in targeted deliverables.



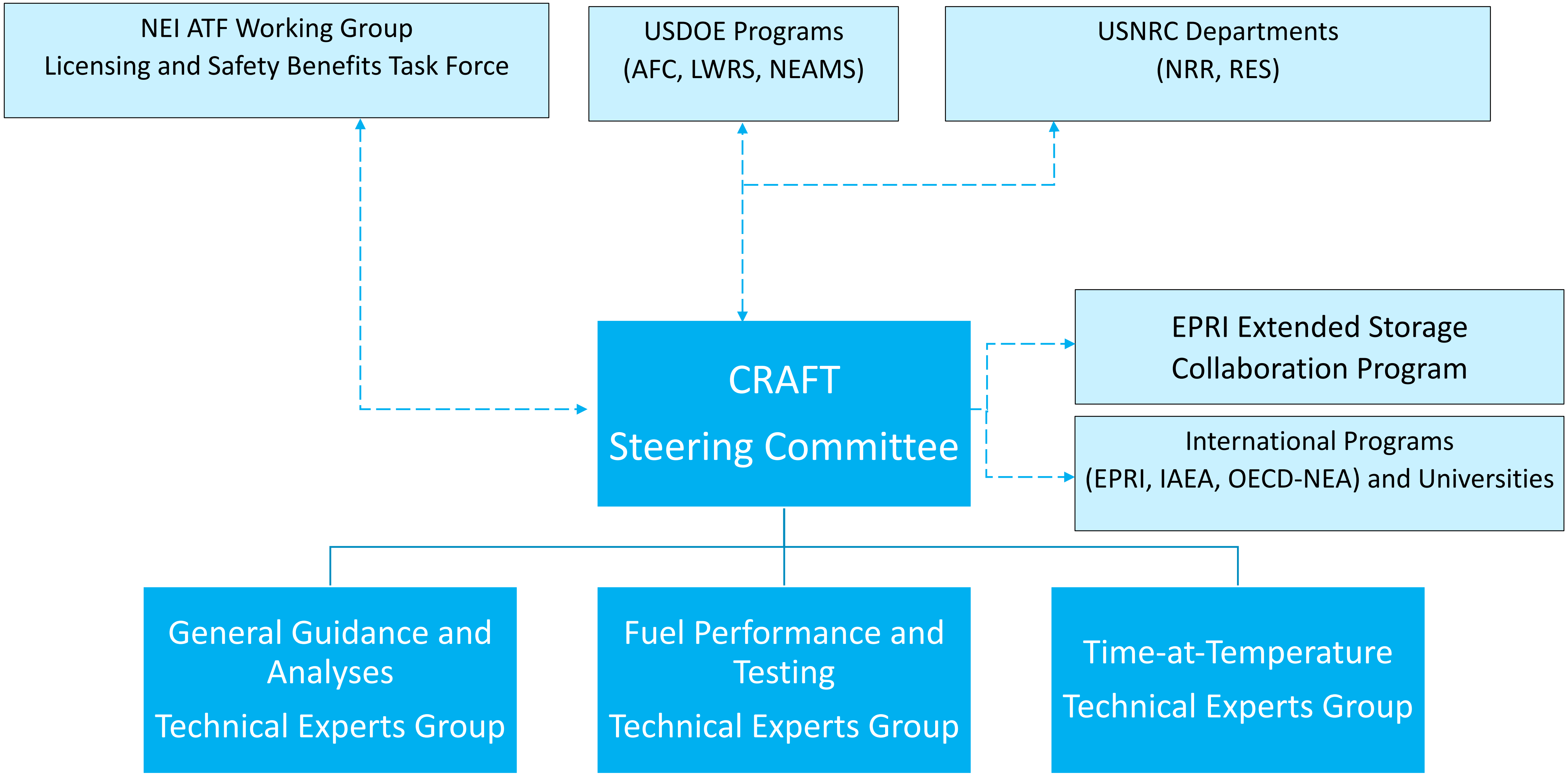
CRAFT Technical Focus

- Initial focus of CRAFT to inform technical bases toward various licensing approaches and implications of **Fuel Fragmentation, Relocation and Dispersal (FFRD)** for higher burnup operations (~75 GWd/MTU)
- Stakeholders on CRAFT Steering Committee to discuss other issues that would benefit from the CRAFT framework



Deliverables to Inform Industry, DOE, NRC and the Global Nuclear Community

CRAFT Structure and Key Stakeholder Interfaces



CRAFT ITM – Research Evaluation Activities

- **Fuel Fragmentation**
  - Higher burnup PIE
  - Advanced fuel characterization and tests
  - In- and out-pile testing
  - Transient Fission Gas Release testing
  - Modeling / Simulation
  - Quantification of fuel susceptible to fragmentation

- **Fuel Relocation**
  - Clad balloon propensity, size, and dynamics
  - Effect of rod internal pressure and clad creep and associated thermal ramp conditions
  - No rupture and rupture cases
  - Quantification of fuel susceptible to relocation
  - Acceptability and applicability of relocated fuel (core, ATF, non-ATF)

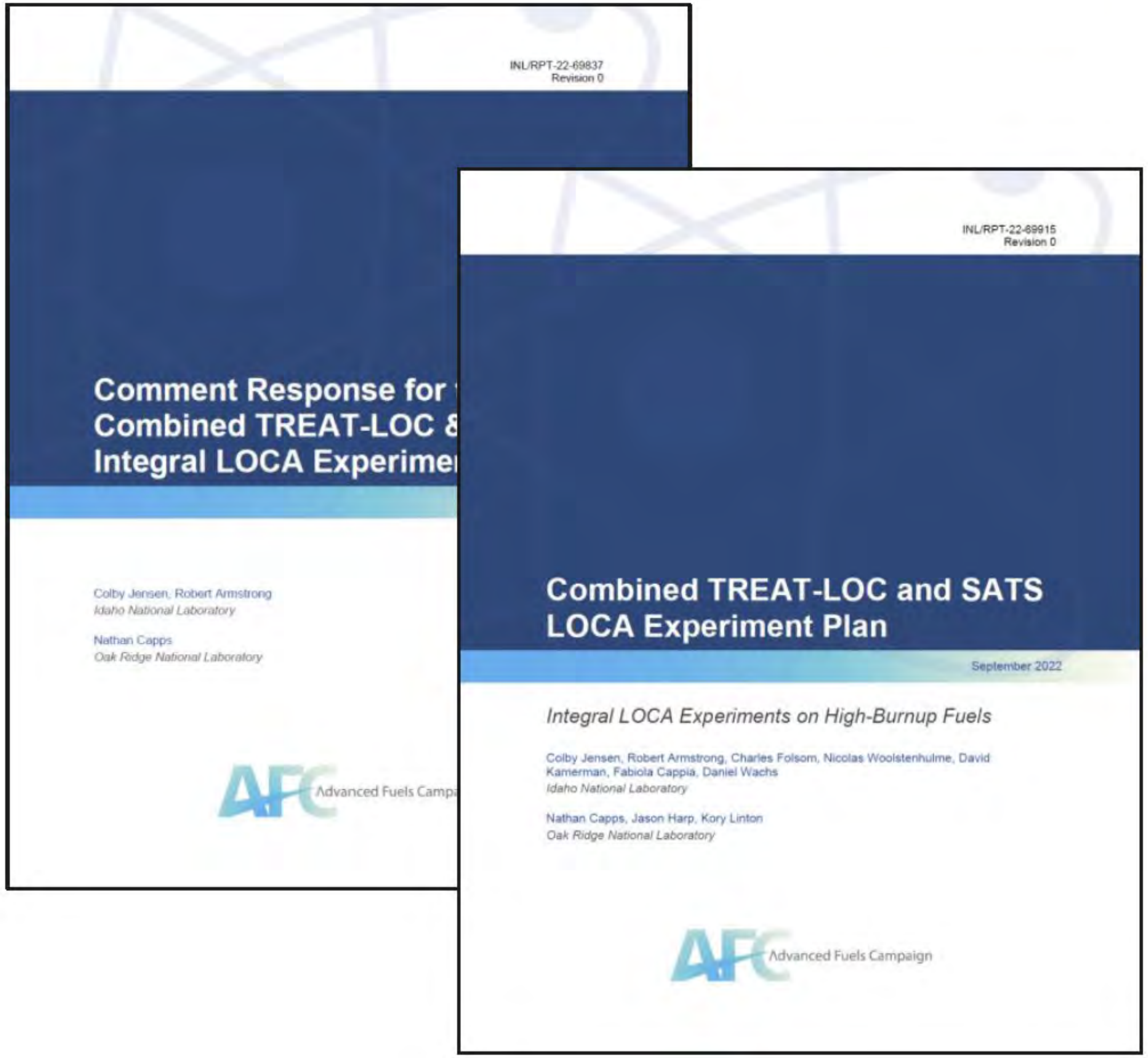
- **Fuel Dispersal**
  - Experimental methodologies for quantifying fragment dispersal
  - Quantification of fuel susceptible to dispersal
  - Acceptability and applicability of dispersed fuel (core, ATF, non-ATF)
  - Tracking of dispersed fuel
  - Consequence analyses of dispersed fuel

Fuel Performance and Testing TEG ⟷ General Guidance and Analyses TEG

Recent Developments

Supported the development of DOE-AFC LOCA Testing Plan

- Combined integral and semi-integral LOCA test plan developed to address cross-cutting stakeholder needs and it leverages the best PIE capabilities in the country.
- Primary emphasis on experimental evaluation of identified R&D gaps in FFRD



Recent Developments (cont.)

Technical Expert Panel Assessment of Existing Fuel Fragmentation, Relocation and Dispersal Data, EPRI 3002025542.

- CRAFT is performing an official review of the published White Paper. Now, comments are being addressed in a revision.

Time at Temperature Criteria

- Steering Committee agreed to form a Technical Expert Group to prepare a material testing plan.

New TEG for Time-at-Temperature

Time at Temperature Criteria

– Time at temperature material testing and fuel performance will be a CRAFT focus area for 2023.
– T/H aspects of TaT, including modeling and testing, are handled by respective fuel suppliers.
– Beneficial for potential power uprate projects as well as fuel cycle economy.
– A new TEG is formed to develop the material testing plan including identification of testing facility, defining the test protocol, and selection of materials.

Summary

- CRAFT provides a forum for various stakeholders
- Focuses on issues that are relevant to
  - deployment of advanced LWR fuel technologies
  - improvements in plant safety, economic, and operational flexibility including for example power uprates and extended cycles
- It is aligned with industry needs and continues to provide valuable contributions

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EPRI Used Fuel and High-Level Waste Program Overview

Bob Hall
May 18, 2023

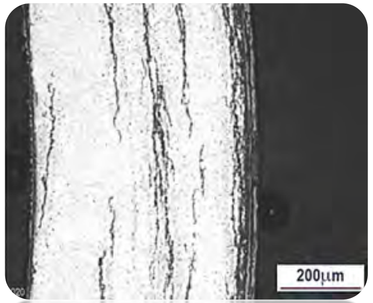
ACRS Meeting

5/10/23 Rev. 0

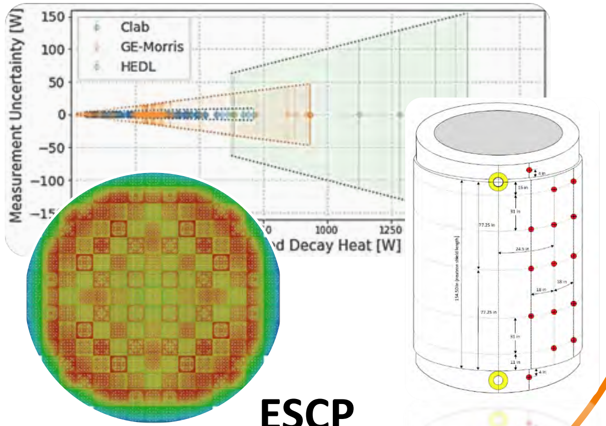
UF&HLW Research Focus Areas

Aging Management of Dry Storage Systems

Cladding Performance

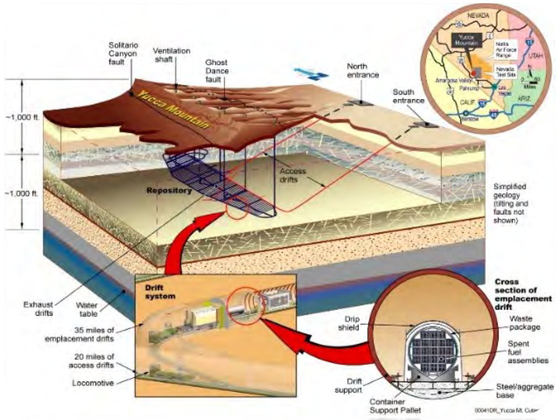


Cross-cutting

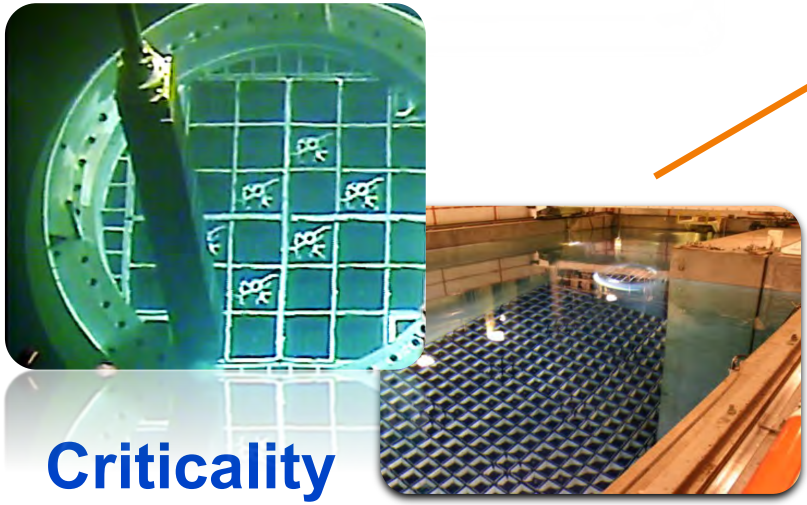


ESCP

Disposal



Criticality



UF&HLW 2023 Research Focus Area Projects

Aging Management of Dry Fuel Storage Components

- Dose Consequences / Internal Particle Deposition
- Dry Storage System Mitigation & Repair
- Canister NDE Demonstration and Support
- Dry Transfer System Options
- Bolted Cask Seal Leak Indication Response

Used Fuel Cladding Performance During Storage & Transportation

- High Burnup Demonstration
- Alternate Fuel Performance Metric PIRT
- HBU International Cladding Collaborations (NFIR, SCIP, IAEA SFERA)
- Fuel Cladding Analysis

Used Fuel Criticality Control During Storage & Transportation

- ATF/HBU/HE SFP Criticality*
- i-LAMP: Global SFP NAM Monitoring
- Neutron Absorber Materials / NAUG
- Metamic Performance Evaluation
- SFP NAM In-situ Measurement Tool

Cross-Cutting Research

- Extended Storage Collaboration Program*
- Decay Heat Measurements and Validation*
- DSS Dose Modeling
- Canister Sensors
- DOE Canister Testing
- UNFSTANDARDS Enhancements

*Covered in another presentation today

i-LAMP

i-LAMP: Industrywide Global Learning Aging Management Program

Global program – Initial focus is on BORAL®

- NAM specifications (type, vintage)
- NAM history (installation and manufacturing years)
- SFP water chemistry history
- NAM performance (coupon monitoring)

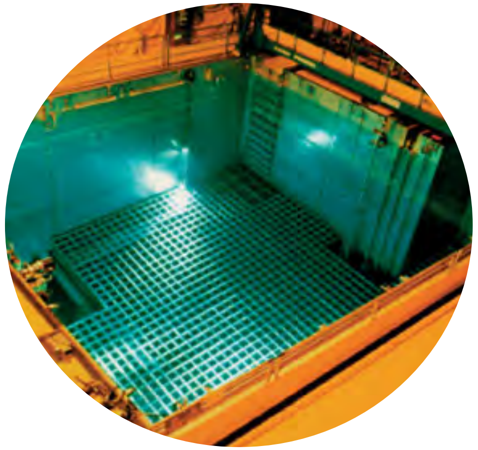
Sibling Pool Process – If No Coupons

- Identify sibling(s)
- Commitment to i-LAMP for AMP
- Periodic data updates ("learning")
- Periodic sibling performance update

SFP With Coupons

Siblings

Similar NAM Specifications
Similar Water Chemistry
Similar NAM Vintage



SFP Without Coupons

EPRI report, 3002018497, that summarizes i-LAMP is published and i-LAMP is currently under NRC review as part of NEI 16-03 Revision 1.

PIRT Activities

Phenomena Identification and Ranking Table (PIRT) Activities

Fuel

- Published in EPRI report, 3002018439, in 2020
- Led to the Gross Rupture PIRT,
  - New definition of GR that is more actionable
  - Published in EPRI report 3002020929
- Alternate Fuel metric PIRT is being finalized
  - Report will be published in August 2023
- Next steps, for regulatory review/implementation, are being discussed

Thermal Modeling

- Published in EPRI report, 3002018441, in 2020
- Need for evaluation of
  - Code-to-code variations
  - User-to-user variations
- Led to the international thermal benchmark project

Decay Heat

- Published in EPRI report, 3002018440, in 2020
- Identified gaps
  - Lack of measurement data for high burnup and short cooling times
- Recommended publication of “unpublished” Clab decay heat measurements
  - Due to high quality of measurements
- Led to SKB-EPRI joint project

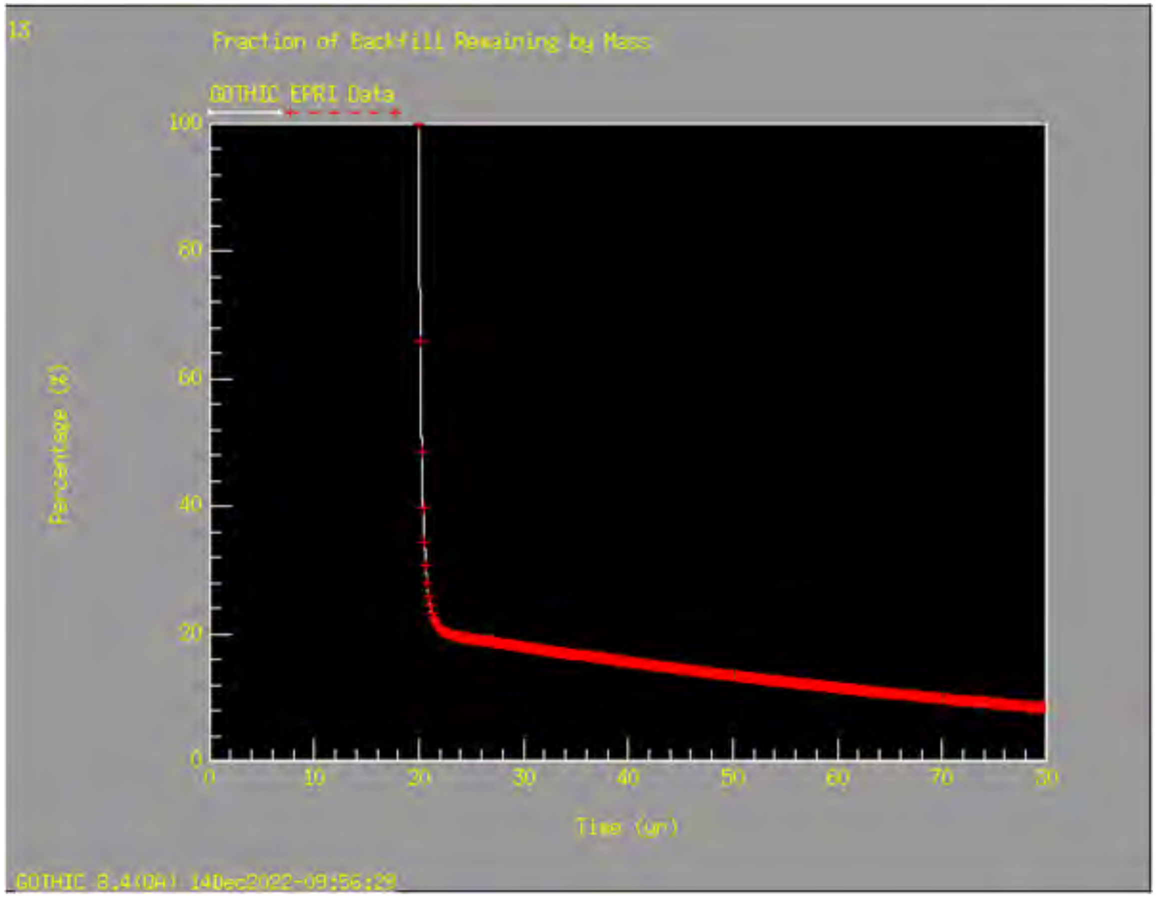
Experts from many organizations (DOE Labs, NRC, vendors, utilities) participated in PIRTs

Reports are publicly available from epri.com

Used Fuel Canister Consequence Modeling

GOTHIC Benchmarking Against Canister CISCC Canister-to-Environment Flow Rate

- Initial investigation of GOTHIC capability
  - Comparison to EPRI 3002015062 results.
- Modeled characteristics
  - Canister internal free volume, backfill gas, crack characteristics (size, roughness, tortuosity, etc.)
  - 19.5 Kw initial decay heat, gas temperature a function of decay heat (t), external temperature
- Not modeled
  - Internal geometry details
  - Particulates
  - Detailed heat transfer, convective flow
- Results match closely with EPRI 3002015062.
- Technical report, *GOTHIC 8.4 Benchmarking Against Canister CISCC Flow Rate*, published April 2023



EPRI

Wireless Internal Canister Sensors Update

Cross-Cutting: Canister Internal Sensors

Goals:

- No wires, internal power, or penetrations
- Measure temperature and pressure
- Direct confirmation of thermal margin
- Direct confirmation of canister pressure

4Q 2021 — Stakeholder Meeting

Project goals and requirements [3002021192]

1Q 2022 — Component Lifetime

Testing at high temperature and radiation dose [3002023140]

2Q 2022 — System Design

Operating principles, testing needs, gaps [3002022694]

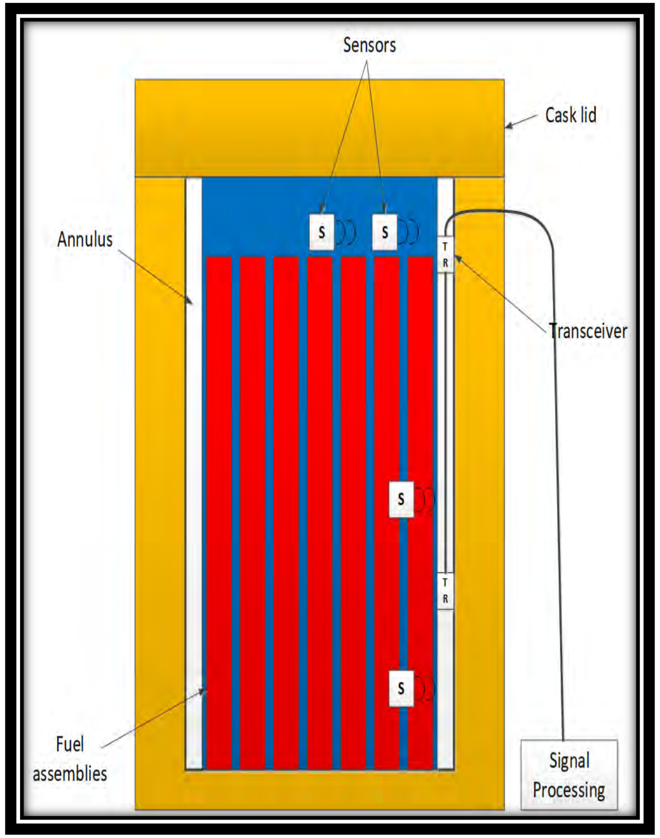
2H 2022 — Prototype Testing

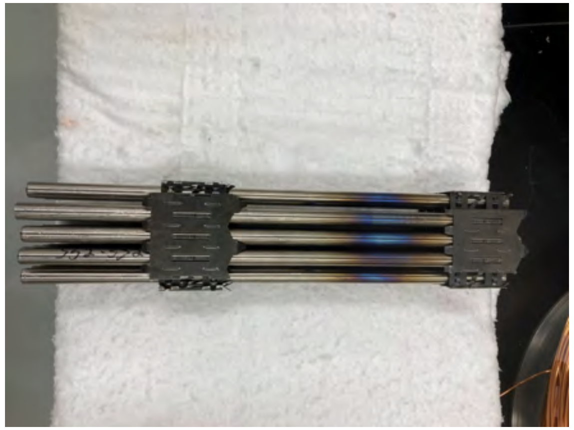
System testing and iterative design improvement

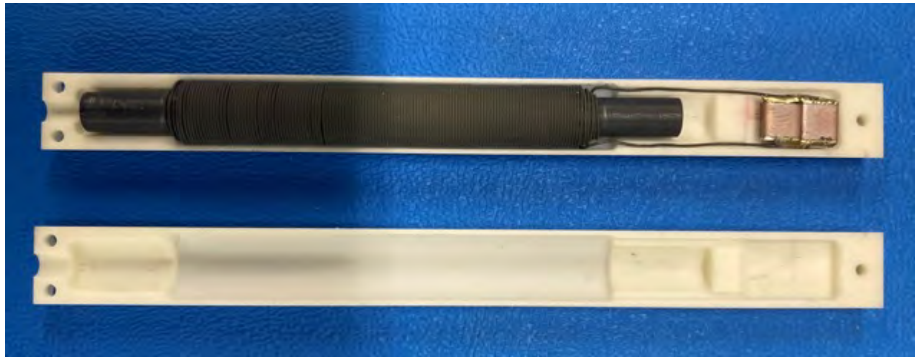
2023 — Demonstration

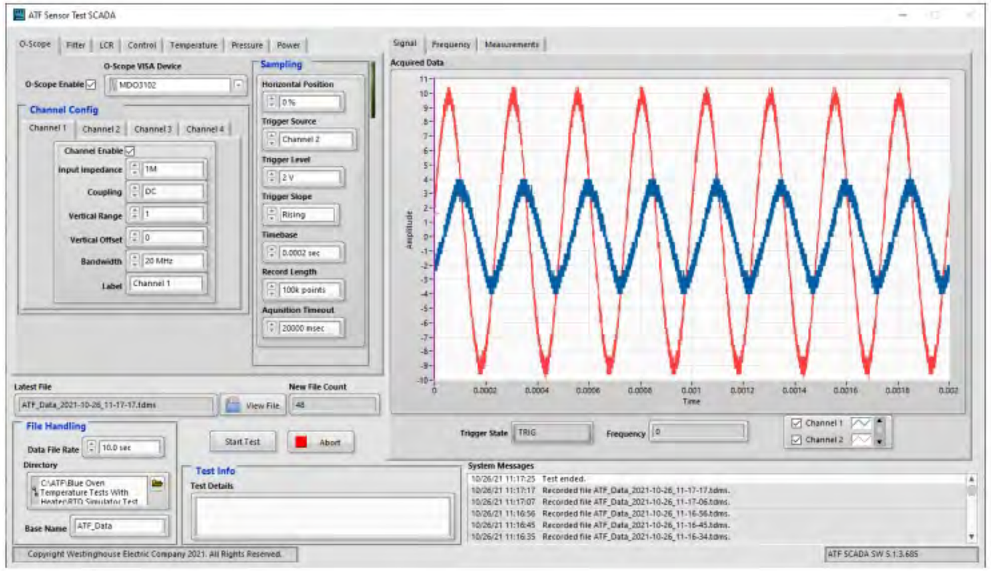
Lab-scale and full-scale performance demonstration

Cross-Cutting: Canister Internal Sensors









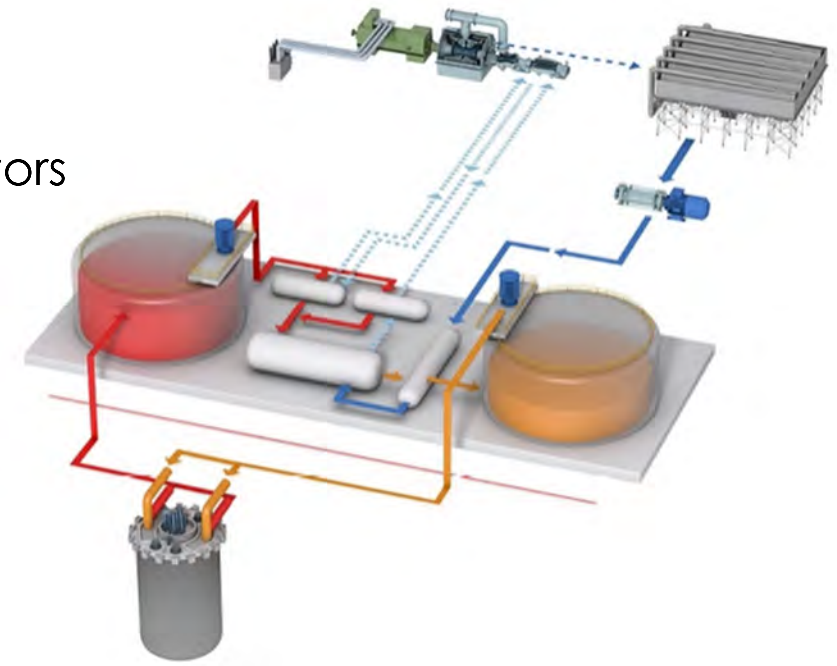
Pictures Courtesy of Westinghouse Electric Company LLC

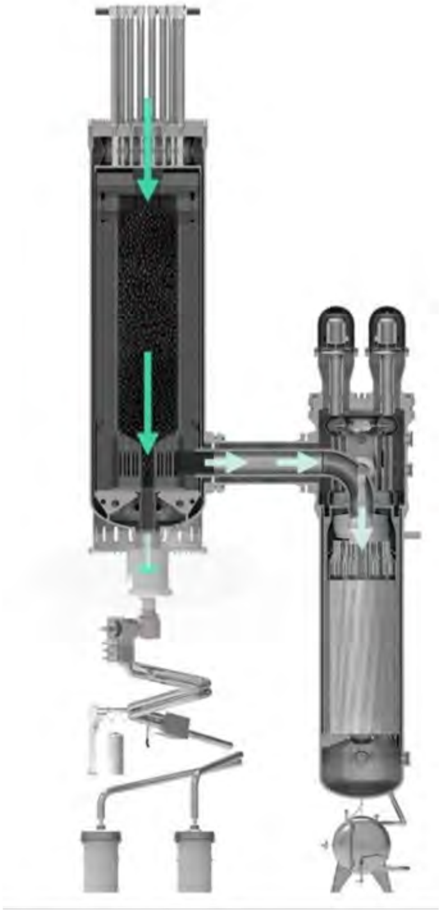
Advanced Reactors and Used Fuel Recycling

Used Fuel Reprocessing in an Advanced Reactor Era

Concise summary and options

- Fuel recycling history
- Technology readiness
- Cost estimates
- Resource requirements
- Recycle and waste products
- Integration with advanced reactors



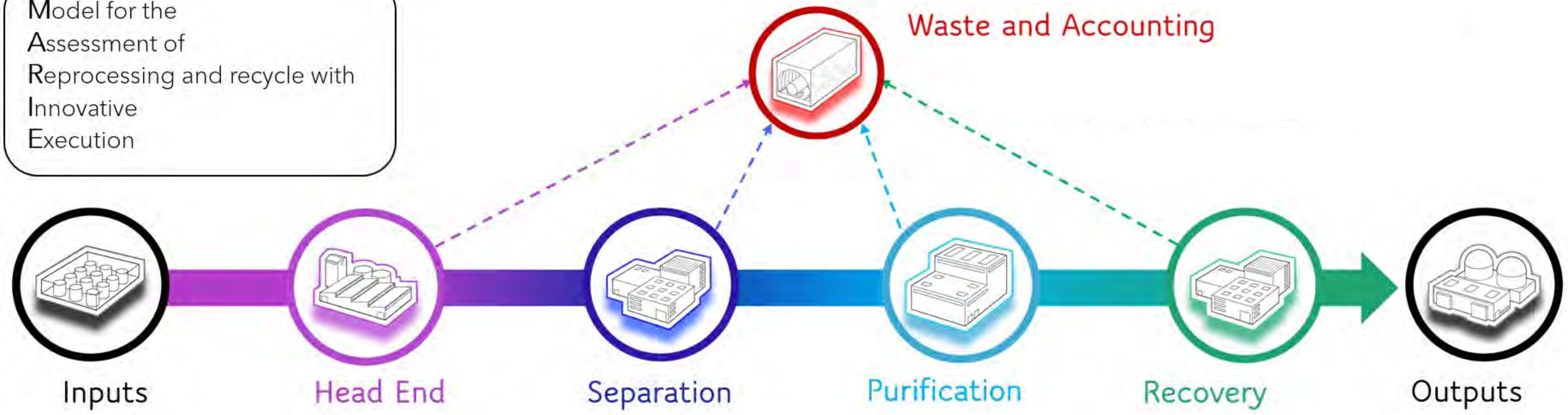


Are there reactor/recycle combinations that make sense?

Converting UNF Radioisotopes into Energy (ARPA-E CURIE)

Project MARIE, Selected by ARPA-E

Model for the
Assessment of
Reprocessing and recycle with
Innovative
Execution



Developing	Flowsheet Optimization Tool Preconceptual Reprocessing Facility Design Technology Commercialization Plan	

Partners	Southern Company Deep Isolation Oak Ridge National Lab

Maturing historic and evolving technologies for economic recycling

Questions?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2022-2023 UFHLW Papers and Publications

1. [IHLRWM2022] Hatice Akkurt, "*i-LAMP: Industrywide Learning Aging Management Program for Global Monitoring of Spent Fuel Pools*," published in Proceedings of International High Level Radioactive Waste Management (IHLRWM 2022) Conference, November 2022. [Presented by Akkurt at IHLRWM 2022 conference].

2. [IHLRWM2022] Hatice Akkurt and **Robert Hall**, "*Recent Advancements in SFP Criticality for Existing LWR Fuels and Roadmap for Advanced LWR Fuels*," published in Proceedings of International High Level Radioactive Waste Management (IHLRWM 2022) Conference, November 2022. [Presented by Akkurt at IHLRWM 2022 conference].

3. [IHLRWM2022] Henrik Liljenfeldt, **Hatice Akkurt**, **Robert Hall**, Fredrik Johansson, Jesper Kierkegaard, "*Decay Heat Measurements and Analysis for BWR Fuel with Shorter Cooling Time*, " published in Proceedings of International High Level Radioactive Waste Management (IHLRWM 2022) Conference, November 2022. [Presented by Akkurt at IHLRWM 2022 conference].

4. [IHLRWM2022] Amanda Jenks and **Hatice Akkurt**, "*Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Storage and Transportation Applications*," published in Proceedings of International High Level Radioactive Waste Management (IHLRWM 2022) Conference, November 2022. [Presented by Akkurt at IHLRWM 2022 conference].

5. [IHLRWM2022] R. Ferrer, J. Hykes, **H. Akkurt**, **R. Hall**, "*Extension of EPRI Benchmarks to Advanced LWR Fuels for SFP Criticality Depletion Uncertainty*," published in Proceedings of International High level Radioactive Waste Management Conference, November 2022. [Presented by Ferrer-Studsvik at IHLRWM 2022 conference].

2022-2023 UFHLW Papers and Publications

6. [Global2022] Fredrik Johansson, Jesper Kierkegaard, Henrik Liljenfeldt, **Robert Hall**, **Hatice Akkurt**, "*Extending the Validation Range for Decay Heat Measurements*," published in Proceedings of Global 2022 conference, Reims, France, July 6-8, 2022. [Presented by Fredrik Johansson (SKB) at Global conference].

7. [Nuclear Technology Journal] Hatice Akkurt, "*Evaluation of Boral Panels from an Operating Spent Fuel Pool*," under review - Nuclear Technology journal.

8. [PATRAM2023] Hatice Akkurt, Fredrik Johansson, Henrik Liljenfeldt, Amela Mehic, Jesper Kierkegaard, **Robert Hall**, "*Decay Heat Evaluation for Extended Range Using Clab Measurements*," abstract submitted to PATRAM 2023 conference, June 2023, France.

9. [PATRAM2023] Keith Waldrop, "*Transport License Approach to Maintain Thermocouples in High Burnup Research Project Cask*," abstract submitted to PATRAM 2023 conference, June 2023, France.

10. [M&C 2023] Rodolfo Ferrer, Joshua Hykes, **Hatice Akkurt**, **Robert Hall**, "*Extension of Reactivity Decrement Uncertainty to Advanced LWR Fuels via Stochastic Sampling and Sensitivity-Based Verification*," extended summary submitted to the M&C 2023 - The International Conference on Mathematics and Computational Methods, August 2023, Ontario, Canada.

2022-2023 UFHLW Papers and Publications

11. [EPRI Journal] "*A Collective Approach to Safe Used Nuclear Fuel Storage,*" EPRI Journal, March 2022. [H. Akkurt and B. Hall contributors - article highlights EPRI Extended Storage Collaboration Program (ESCP)]

12. [OECD/NEA] Hatice Akkurt, "*Technical Challenges, Solutions and Opportunities for Collaboration for Managing Extended Storage for LWR*," invited speaker at NEA's 55th Plenary Meeting of the Radioactive Waste Management Conference (RWMC), March 2022 - Virtual presentation.

13. [OECD/NEA] Hatice Akkurt, "*EPRI-SKB Collaboration on Decay Heat Measurements and Validation*," invited speaker at OECD/NEA's Working Group on Decay Heat (WG12), June 22, 2022 - Virtual presentation.

14. [ESCP] H. Akkurt and **B. Hall**, "*Extending Validation Range for Decay Heat and Re-assessment of Uncertainties in Measurements*," EPRI ESCP Winter 2022 meeting, November 2022, Charlotte, NC.

15. [ESCP] K. Waldrop, "*Alternate Fuel Performance Metrics Phenomena Identification and Ranking Tables (PIRT)*," EPRI ESCP Winter 2022 meeting, November 2022, Charlotte, NC.

16. [ESCP] Shannon Chu, "*EPRI Mitigation & Repair Activities*," EPRI ESCP Winter 2022 meeting, November 2022, Charlotte, NC.

SFP Criticality for LEU+: Depletion Uncertainty and Criticality Code Validation

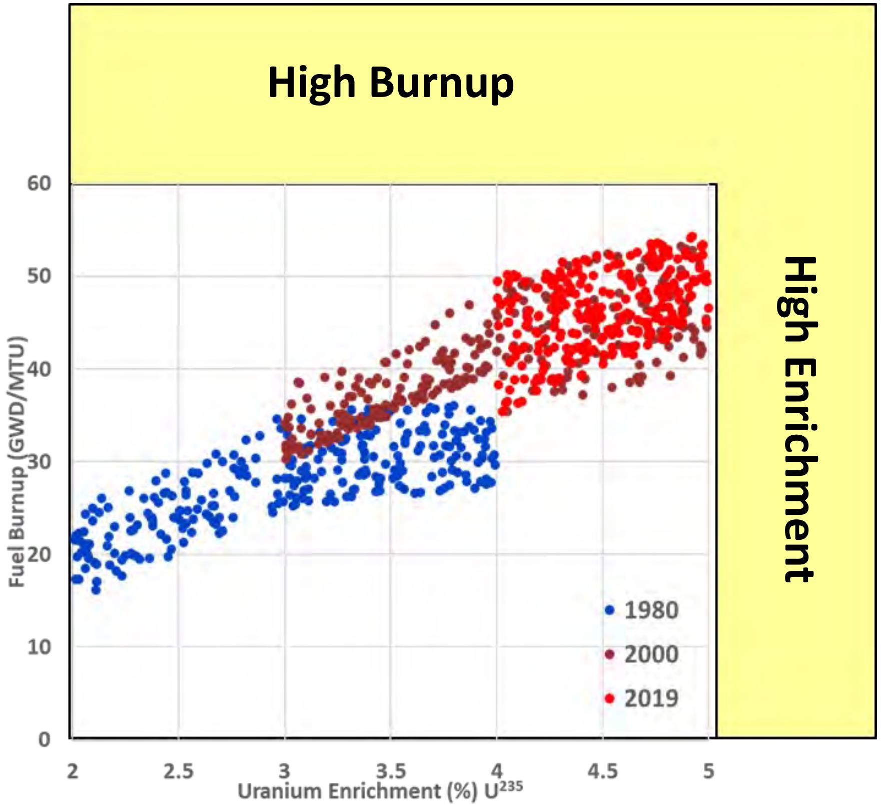
Hatice Akkurt and Bob Hall
Used Fuel and High-Level Waste Program

May 18, 2023

ACRS Meeting

SFP Criticality for Advanced Fuels (LEU+)

- **EPRI formed SFP Criticality for Advanced Fuels Working Group**
  - Composed of representatives from
    - Utilities (Exelon, Southern, Entergy, Duke, Dominion, TVA)
    - Vendors (Westinghouse, GE, Framatome, Studsvik, and Holtec)
    - NEI
  - Conducted working groups meetings
    - **Identified gaps, issues that needs to be addressed**
    - Categorized by leading organization (EPRI, utility, vendor)
  - **Generic issues** (to be led by EPRI)
    1. **Criticality code validation**
    2. **Depletion (burnup credit) uncertainty and bias**
  - Vendor and utility specific issues/gaps identified



LEU+ requires technical bases for SFP, New Fuel Vault criticality safety analyses.

Current guidance (RG 1.240) extends to 5% enrichment and 60 GWd/MTU burnup.

EPRI

Depletion Uncertainty and Bias

Background: Spent Fuel Pool (SFP) Criticality and Depletion Uncertainty and Bias*

- No critical experiments using spent fuel
- Critical experiments are very expensive
- Using fresh fuel assumption for spent fuel causes loss of SFP storage space
- How to account for uncertainty and bias for spent fuel?

1998 Kopp Memo:

"In the absence of any other determination of the depletion uncertainty, an uncertainty equal to 5 percent of the reactivity decrement to the burnup of interest is an acceptable assumption."

1998-2009

Easy to use, implement, justify; subsequently, used by many utilities

NRC: What is the technical justification or where is the documentation for 5% decrement?

Burnup Credit Approaches

ORNL: Chemical Assay Based Approach**

NUREG/CRs

7108: Validating isotopics for BC

7109: Validating isotopics for k_{eff}

EPRI: Depletion Benchmarks Using Flux Maps

EPRI reports

1022909: Benchmarks for Depletion

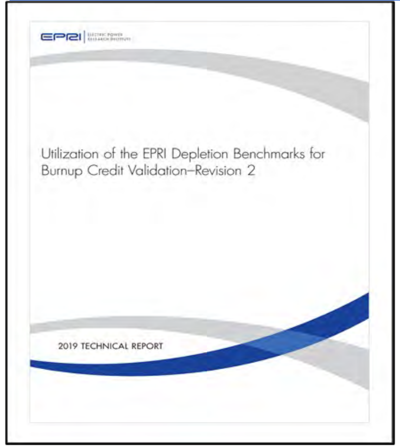
1025203: Utilization of EPRI Benchmarks

*Funded by NRC

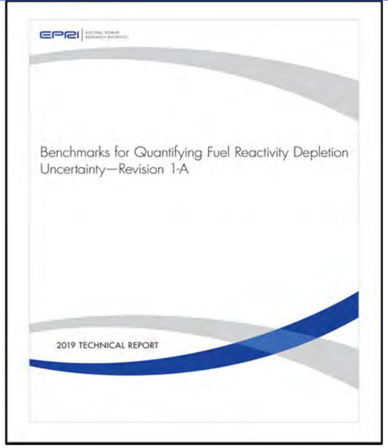
*Slide from ACRS meeting on RG 1.240 - March 3, 2021

Background: EPRI Benchmarks*

Received final SER on July 26, 2019



3002016888, *Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation - Revision 2*, published **August 29, 2019**

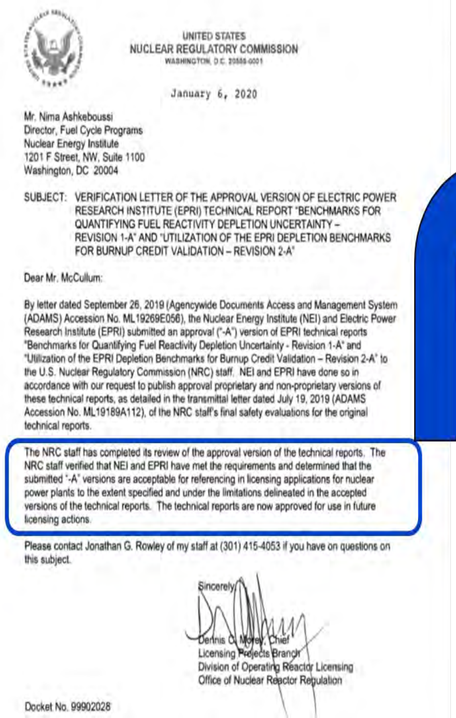


3002016035, *Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty-Revision 1-A*, published **September 18, 2019**

3002017254, *Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation - Revision 2-A*, published **September 18, 2019**

Burnup (GWd/MTU)	EPRI Uncertainty (%)	Additional NRC Bias (%)
10	3.05	0.0
20	2.66	0.0
30	2.33	0.0
40	2.12	0.15
50	1.95	0.35
60	1.81	0.54

EPRI benchmarks showed that Kopp memo (5%) is conservative and provided technical justification for additional margins



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 6, 2020

Mr. Nima Ashkeboussi
Director, Fuel Cycle Programs
Nuclear Energy Institute
1201 F Street, NW, Suite 1100
Washington, DC 20004

SUBJECT: VERIFICATION LETTER OF THE APPROVAL VERSION OF ELECTRIC POWER RESEARCH INSTITUTE (EPRI) TECHNICAL REPORT "BENCHMARKS FOR QUANTIFYING FUEL REACTIVITY DEPLETION UNCERTAINTY – REVISION 1-A" AND "UTILIZATION OF THE EPRI DEPLETION BENCHMARKS FOR BURNUP CREDIT VALIDATION – REVISION 2-A"

Dear Mr. McCullum:

By letter dated September 26, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19269E056), the Nuclear Energy Institute (NEI) and Electric Power Research Institute (EPRI) submitted an approval ("-A") version of EPRI technical reports "Benchmarks for Quantifying Fuel Reactivity Depletion Uncertainty - Revision 1-A" and "Utilization of the EPRI Depletion Benchmarks for Burnup Credit Validation – Revision 2-A" to the U.S. Nuclear Regulatory Commission (NRC) staff. NEI and EPRI have done so in accordance with our request to publish approval proprietary and non-proprietary versions of these technical reports, as detailed in the transmittal letter dated July 19, 2019 (ADAMS Accession No. ML19189A112), of the NRC staff's final safety evaluations for the original technical reports.

The NRC staff has completed its review of the approval version of the technical reports. The NRC staff verified that NEI and EPRI have met the requirements and determined that the submitted "-A" versions are acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the accepted versions of the technical reports. The technical reports are now approved for use in future licensing actions.

Please contact Jonathan G. Rowley of my staff at (301) 415-4053 if you have on questions on this subject.

Sincerely,

Dennis C. Morey, Chief
Licensing Projects Branch
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 99902028

The NRC staff has completed its review of the approval version of the technical reports. The NRC staff verified that NEI and EPRI have met the requirements and determined that the submitted "-A" versions are acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the accepted versions of the technical reports. ***The technical reports are now approved for use in future licensing actions.***

Received final approval letter on January 6, 2020

*Slide from ACRS meeting on RG 1.240 - March 3, 2021

Burnup Credit Uncertainty and Bias for LEU+ SFP Criticality

1st Question: Is the Kopp memo depletion uncertainty (5%) sufficient for ATF/HE/HBU?

2nd Question: Can regulator-approved EPRI benchmarks be extended for ATF/HE/HBU?

Burnup (GWd/ MTU)	Uncertainty (%)	Bias (%)
10	3.05	0.0
20	2.66	0.0
30	2.33	0.0
40	2.12	0.15
50	1.95	0.35
60	1.81	0.54

Multiple independent analysis approaches

1
- CASMO5 Lattice Physics Analysis
- Similar reactivity contributions, similar uncertainty

2
- Sensitivity/Uncertainty (S/U) Similarity Methods (c_k)
- SCALE Sampler confirmation of S/U total XS uncertainty

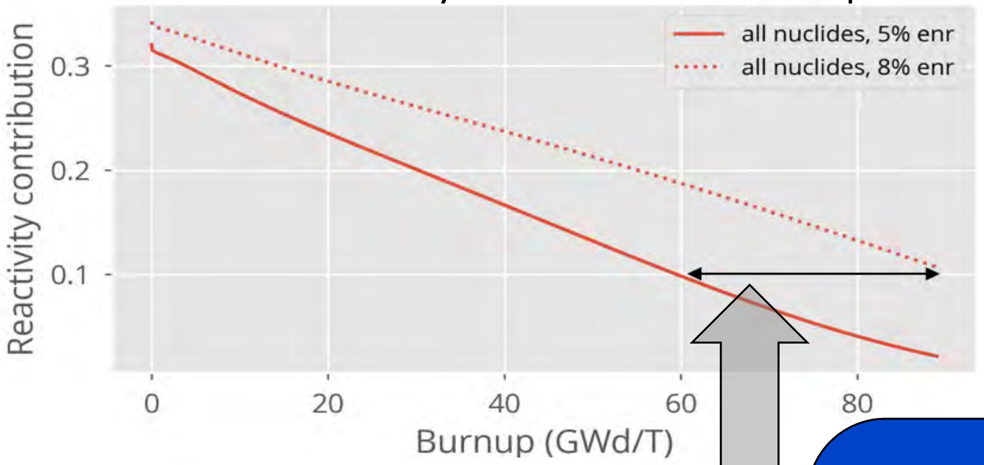
3
- Extension of EPRI Depletion Benchmarks
- Stochastic sampling bias and uncertainty estimates

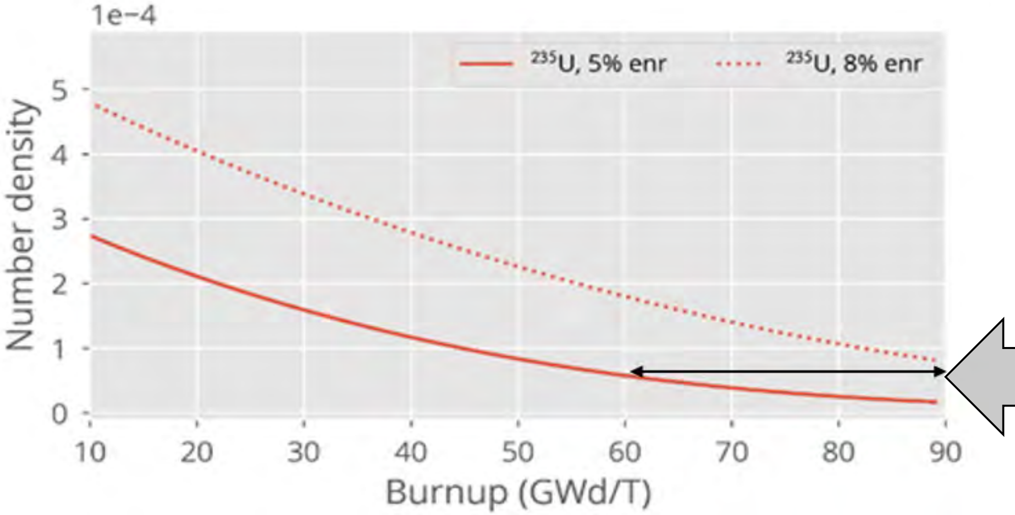
If analysis supports, simple use of 5% uncertainty without additional experimental data, no added conservatism

Comparison of Actinide Production and Depletion

Nuclide	Number Density (8% vs 5%)	Reactivity (8% vs 5%)
U-235	**Higher**	**Higher**
U-238	Lower	Lower
Pu-238	Similar	Similar
Pu-239	**Higher**	Lower
Pu-240	Lower	Lower
Pu-241	**Lower/Higher**	Lower
Am-241	**Lower/Higher**	Lower/Similar
Am-242	Small	Small
Am-242m	Small	Small
Am-243	Lower	Lower
Cm-242	Lower/Similar	Small
Cm-244	Lower	Lower

Actinide reactivity as a function of burnup





U-235 number density as a function of burnup

5%, 60 GWd/T
and
8%, 90 GWd/T
are very similar

Of the actinides, only U-235 is more important for LEU+, accuracy of U-235 is well known

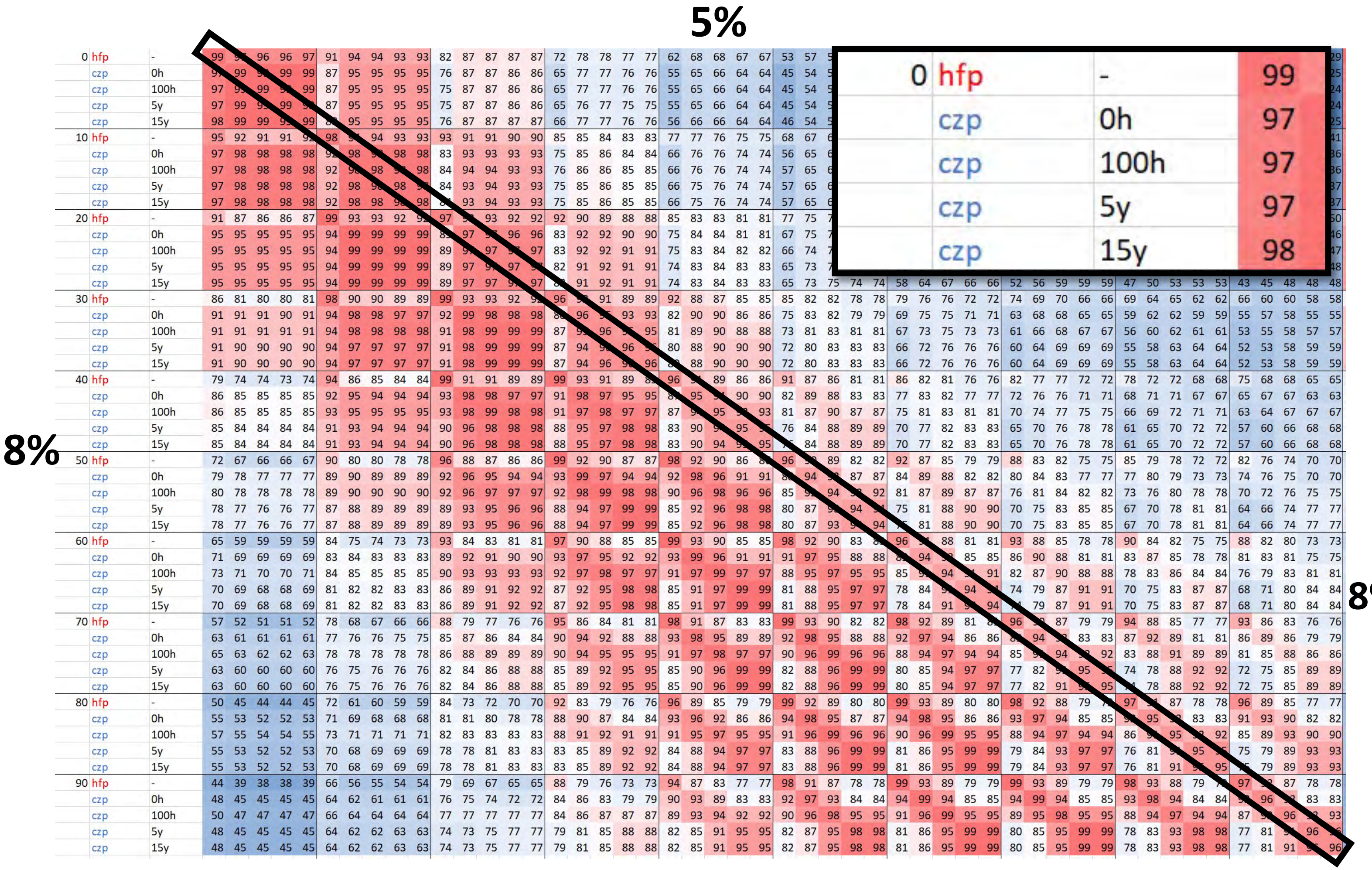
SCALE/Tsunami Sensitivity/Uncertainty

- **Similarity coefficient c_k**
  - $c_k = 1$, same neutronic sensitivity and uncertainty
  - $c_k = 0$, not neutronically similar
  - Typical values range from 0.8 to 0.9 for "similar"

NUREG-2216: In recent years, some analytical tools have been developed that ***may be useful for identifying applicable benchmark experiments*** and evaluating the quality of the experiments. These tools include ***SCALE's TSUNAMI tools, which use sensitivity and uncertainty techniques to provide a quantitative measure of the overall similarity*** of an experiment to the analyzed package…

ISG-10 Rev. 1: The NRC staff currently considers ***a correlation coefficient of ck ≥ 0.95 to be indicative of a very high degree of similarity***. This is based on the staff's experience comparing the results from TSUNAMI to those from a more traditional screening criterion approach. Conversely, ***a correlation coefficient less than 0.90 should not be used as a demonstration of a high degree of benchmark similarity***. ***Because of limited use of the code to date, these observations should be considered tentative and thus the reviewer should not use TSUNAMI as a "black box," or base conclusions of adequacy solely on its use.*** However, it may be used to test a licensee's statement that there is a high degree of similarity between experiments and applications.

Similarity Analysis Using Tsunami



Similarity coefficient c_k

- $c_k = 1$, same neutronic sensitivity and uncertainty
- $c_k = 0$, not neutronically similar

CZP with 100 hours cooling

5% (columns) / 8% (rows)

Bu	0	0.5	10	20	30	40	50	60	70	80	90
0	0.99	0.99	0.95	0.87	0.77	0.66	0.54	0.45	0.36	0.3	0.25
0.5	0.99	0.99	0.95	0.87	0.78	0.67	0.56	0.46	0.37	0.31	0.26
10	0.98	0.98	0.98	0.94	0.86	0.76	0.66	0.57	0.49	0.42	0.37
20	0.95	0.96	0.99	0.97	0.92	0.84	0.75	0.67	0.59	0.53	0.48
30	0.91	0.92	0.98	0.99	0.96	0.9	0.83	0.75	0.68	0.62	0.58
40	0.85	0.86	0.95	0.99	0.98	0.95	0.9	0.83	0.77	0.72	0.67
50	0.78	0.79	0.9	0.97	0.99	0.98	0.94	0.89	0.84	0.8	0.76
60	0.71	0.72	0.85	0.93	0.98	0.99	0.97	0.94	0.9	0.86	0.83
70	0.62	0.64	0.78	0.89	0.95	0.98	0.99	0.97	0.94	0.91	0.88
80	0.54	0.56	0.71	0.83	0.92	0.97	0.99	0.99	0.97	0.95	0.93
90	0.47	0.48	0.64	0.77	0.87	0.94	0.98	0.99	0.98	0.97	0.96

5% 60 GWd and 8% 90Gwd ➔ c_k= 0.99 (highly similar)
Tsunami and physics results are in agreement – no increase in uncertainty for LEU+

EPRI

EPRI Depletion Benchmark Extension

- **Bias and Uncertainty from previous work, approved by the regulator, based on measured flux map data**

Burnup (GWd/MTU)	10	20	30	40	50	60
Bias (% of depletion reactivity)	0.30	0.34	0.36	0.38	0.40	0.41
Uncertainty (% of depletion reactivity)	3.05	2.66	2.33	2.12	1.95	1.81

- **Bias and Uncertainty from current work, using stochastic sampling, as confirmatory analysis for 5% enrichment**

Burnup (GWd/MTU)	10	20	30	40	50	60
Bias (% of depletion reactivity)	0.55	0.58	0.58	0.54	0.50	0.45
Uncertainty (% of depletion reactivity)	3.2	2.6	2.5	2.4	2.2	2.0

Good agreement between bias and uncertainty derived from stochastic sampling approach and measured data – stochastic sampling method confirmed to be conservative

EPRI Depletion Benchmark Extension

- Bias and Uncertainty from current work for 5% enrichment – Based on stochastic sampling

Burnup (GWd/MTU)	10	20	30	40	50	60
Bias (% of depletion reactivity)	0.55	0.58	0.58	0.54	0.50	0.45
Uncertainty (% of depletion reactivity)	3.2	2.6	2.5	2.4	2.2	2.0

- Bias and Uncertainty from current work for 8% enrichment

Burnup (GWd/MTU)	10	20	30	40	50	60	70	80	90
Bias (% of depletion reactivity)	0.64	0.62	0.62	0.62	0.61	0.59	0.57	0.56	0.54
Uncertainty (% of depletion reactivity)	3.7	3.0	2.8	2.7	2.6	2.6	2.5	2.4	2.3

Depletion uncertainty is smaller than 5% of reactivity decrement (Kopp) for higher enrichment and burnup

Burnup Credit Uncertainty and Bias for LEU+ SFP Criticality

1st Question: Is the Kopp memo depletion uncertainty (5%) sufficient? -- **YES**

2nd Question: Can regulator-approved EPRI benchmarks be extended for LEU+? -- **YES**

Multiple independent analysis approaches

1
- CASMO5 Lattice Physics Analysis
- Similar reactivity contributions, similar uncertainty

2
- Sensitivity/Uncertainty (S/U) Similarity Methods (c_k)
- SCALE Sampler confirmation of S/U total XS uncertainty

3
- Extension of EPRI Depletion Benchmarks
- Stochastic sampling bias and uncertainty estimates

EPRI report that describes the approaches and results for depletion uncertainty will be published in late 2023. Report will be publicly available.

Multiple independent approaches support the conclusions

LEU+ SFP Criticality Code Validation Preliminary Results

EPRI

LEU+ Criticality Code Validation

Research Questions:

- **Are there sufficient critical benchmark experiments for the 5-8% enrichment range?**
- **Are they needed?**

Analysis Approach:

Survey available benchmark experiments using NUREG/CR-6698 guidance

Perform fresh fuel similarity analysis using TSUNAMI

- Use of TSUNAMI similarity/uncertainty (S/U) tools is an NRC recognized approach (NUREG-2216, Draft FCSS-ISG-10 Rev. 1)
- SFP and New Fuel Vault (NFV) Models
- Multiple PWR and BWR assembly types, rack types, neutron absorber loading, soluble boron, storage configurations, ATF features
- Compare 5 wt% storage to 6%, 7% and 8% storage
- If c_k>0.9, neutronically similar, validation for 5 wt% is adequate

Survey of Benchmark Experiments

Critical Benchmark Selection Methods: NUREG/CR-6698

Examples of Table 2.3 Parameters

Characteristic	Explanation / Guidance / This Analysis
Fuel fissionable material	Same fissionable material as the application (^{235}U)
Fuel isotopic Composition	Values close to range of application (Enrichment within ~2% of SFP cases)
Fuel physical form	UO2, pellets/pins
Moderator in the fuel	Water in the fuel lattice
Moderator form	Liquid water
Moderator density	Liquid water
Moderator ratio to fissile material	Within 20% of application range H/U similar to fuel assembly (pin pitch, pin OD, etc.)
Geometry	As close as possible to actual case, not as important as materials (Fuel pins in water, etc.)
Neutron energy	Similar energy range LCT – LEU-COMPOUND-THERMAL, or EALF (eV)

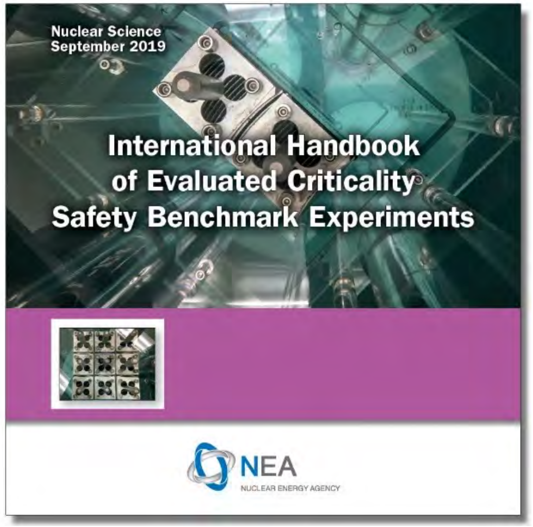
- **Traditional method widely used by industry**
- **Choose experiments based on characteristics**
- **Table 2.3**
  - **Parameters**
  - **Parameter ranges**
  - **Area of applicability**

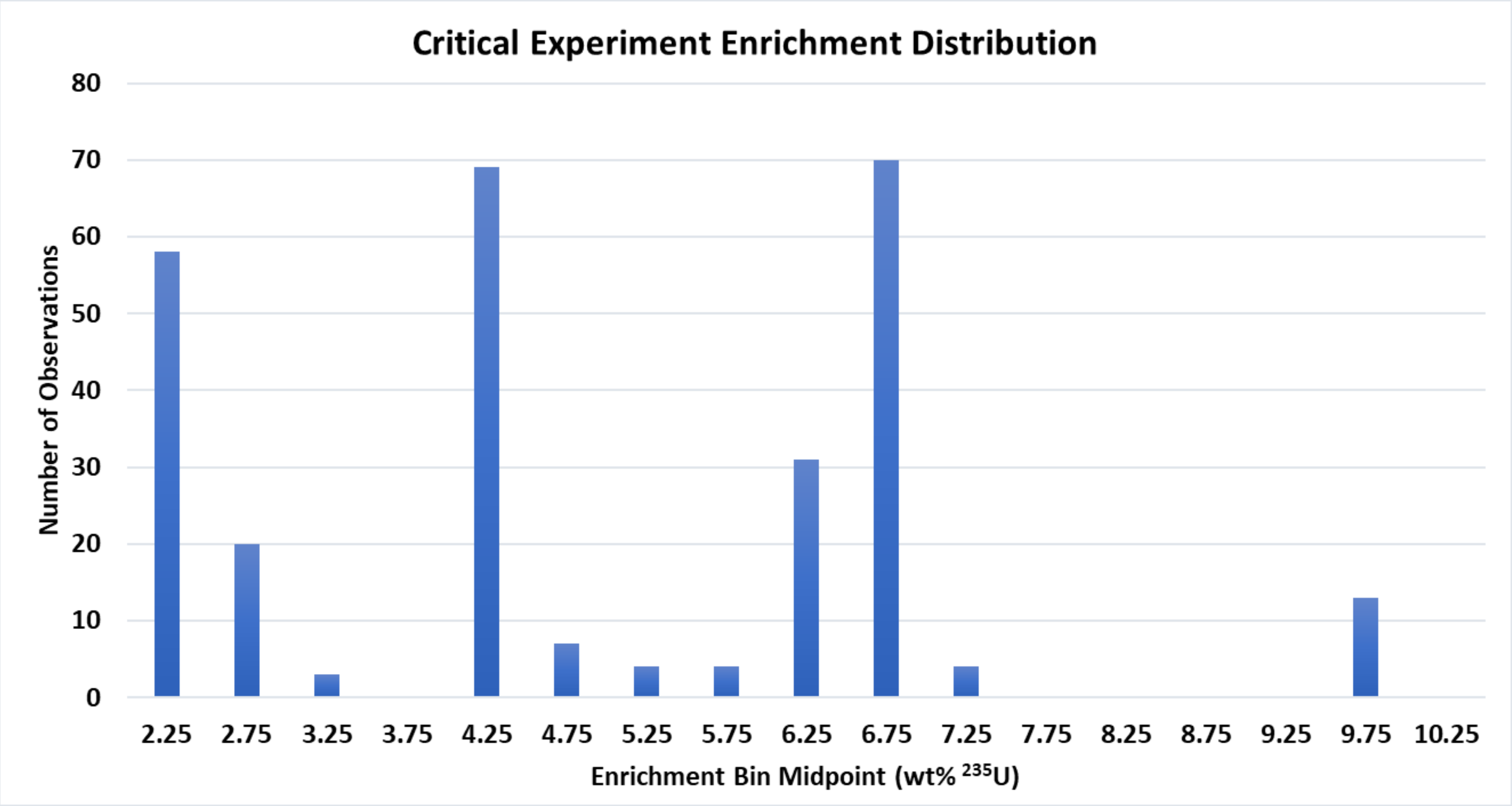
"These values [Table 2.3] *are derived by a number of experienced criticality safety specialists and are necessarily conservative in order for a consensus to be obtained".*

Experiments chosen based on materials/geometry/energy spectrum similarity

Analysis Approach 2: Critical Experiments Selected

- Chosen using NUREG/CR-6698 method
- Most from ICSBEP Handbook
- 280 benchmark experiments
  - 2-10 wt% ^{235}U
  - Fuel pins in water
  - 0.05 – 2.1 eV neutron energy





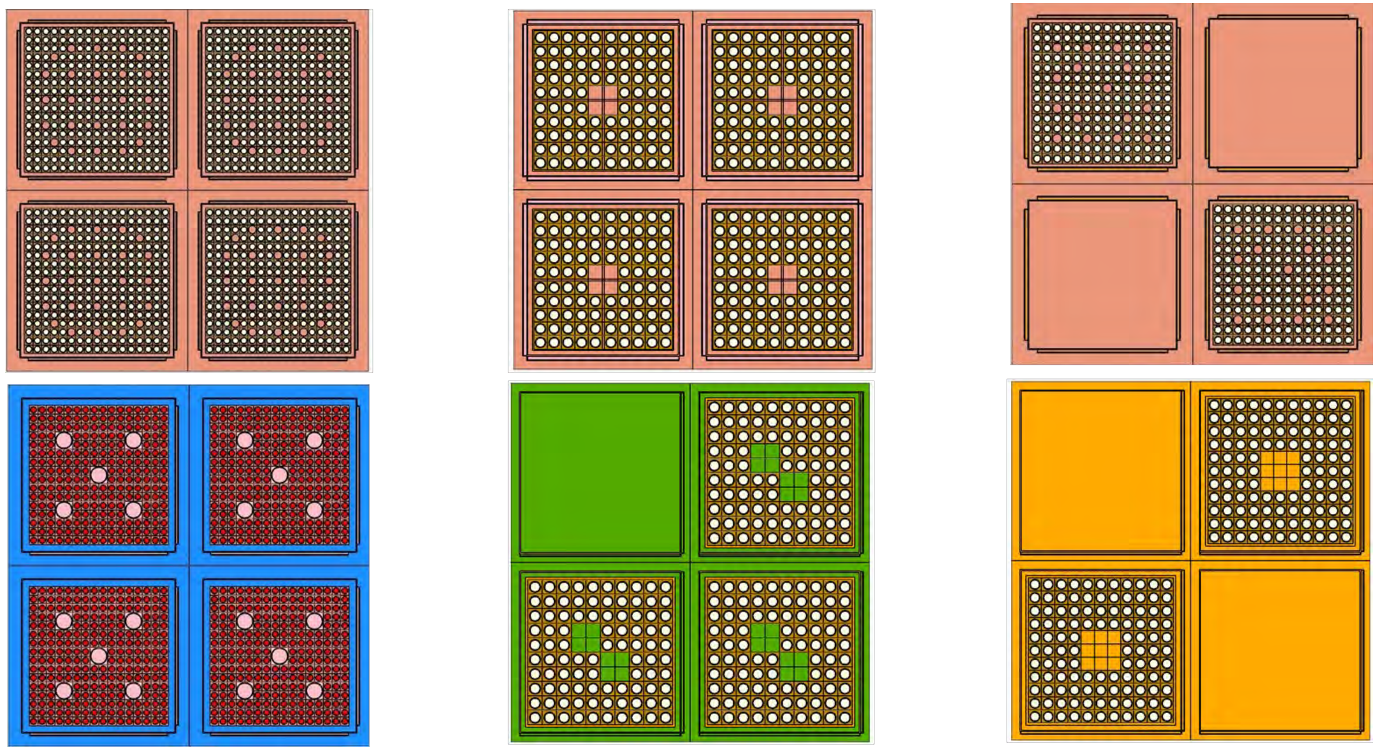
Significant number of experiments in the 5-8% range

Enrichment Similarity in the SFP

Criticality Code Validation: Spent Fuel Pool Application Range

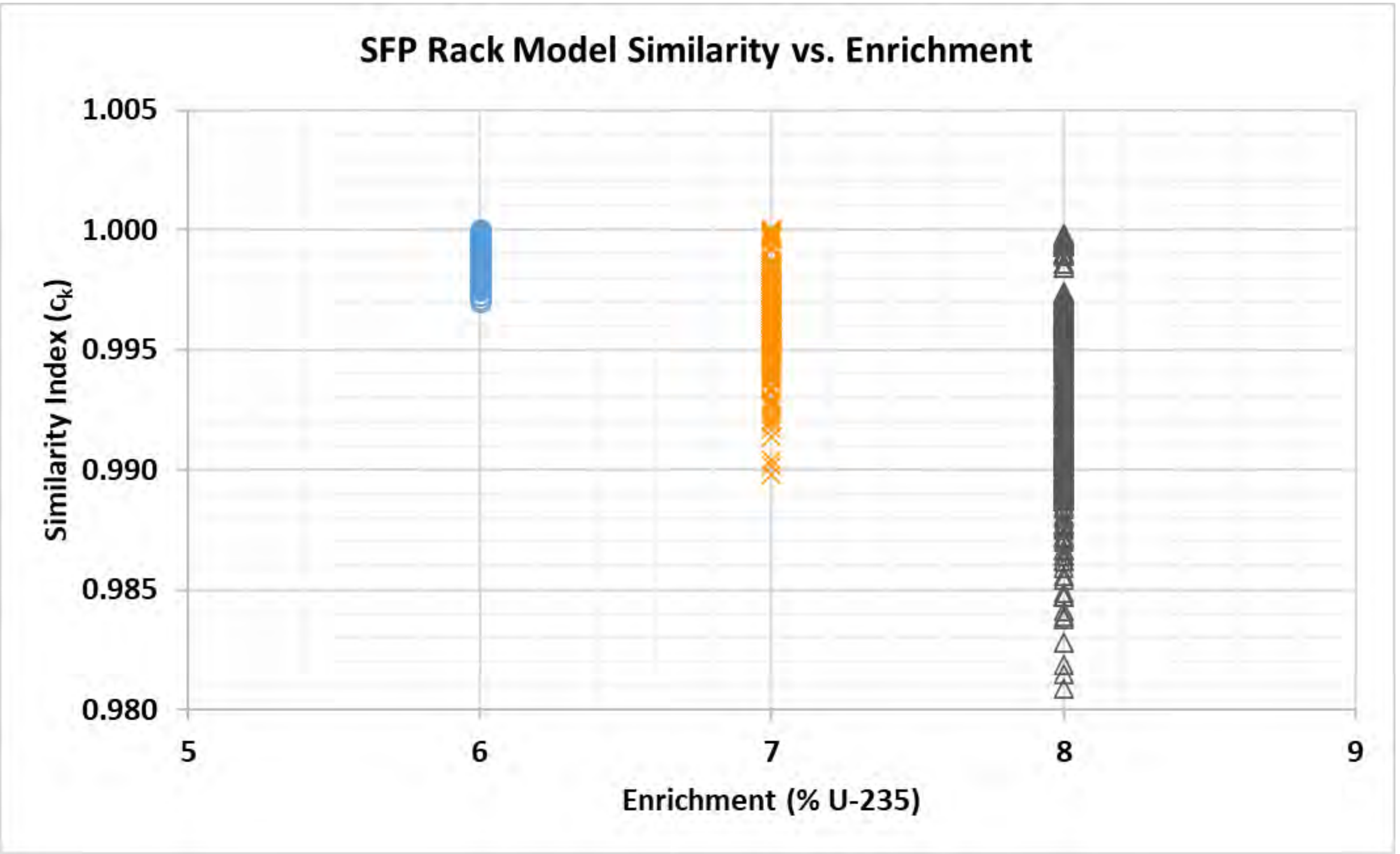
Large Analysis for SFP

- PWR and BWR
  - Multiple designs
  - Poisoned and un-poisoned
  - Multiple enrichments
  - ATF Clad Coatings
- Region 1 and Region 2 Racks
  - Low, and high poison loading
  - 0, 400, 2500 ppm boron for PWR
- Multiple Loading Configurations
  - 1-out-of-4
  - 2-out-of-4
  - 3-out-of-4
  - 4-out-of-4
- Calculate Similarity Index (c_k)
  - "Apples and apples" comparison varying only enrichment



Flux-trap and non-flux-trap SFP rack designs, 1-out-of-4 to 4-out-of-4 configurations

Analysis Approach 1 Results: Increased Enrichment



c_k vs Enrichment for PWR and BWR Scenarios (Nominal Models at 5 wt. % U-235)

- SCALE Tsunami Similarity/Uncertainty
- Compare same rack model with different fuel enrichment
- Very high similarity indicates acceptable 5% enrichment validation is also acceptable for < 8% enrichment

Very weak enrichment effect, c_k > 0.95 indicates very high neutronic similarity

Analysis Summary and Conclusions

- Significant number of benchmark experiments in the 5-8% enrichment range
- Enrichment is a very weak variable for new fuel in the SFP
  - Same result across numerous fuel types, rack SFP types, configurations

Publications

1. H. Akkurt and R. Hall, "*Recent Advancements in SFP Criticality for Existing LWR Fuels and Roadmap for Advanced LWR Fuels*," Proceedings of International High level Radioactive Waste management Conference, November 2022.

2. R. Ferrer, J. Hykes, H. Akkurt, R. Hall, "*Extension of EPRI Benchmarks to Advanced LWR Fuels for SFP Criticality Depletion Uncertainty*," Proceedings of International High level Radioactive Waste Management Conference, November 2022.

3. R. Ferrer, J. Hykes, H. Akkurt, R. Hall, "*Extension of Reactivity Decrement Uncertainty to Advanced LWR Fuels via Stochastic Sampling and Sensitivity-Based Verification*," accepted for M&C 2023 conference, August 2023.

Questions/Comments?


Together...Shaping the Future of Energy®

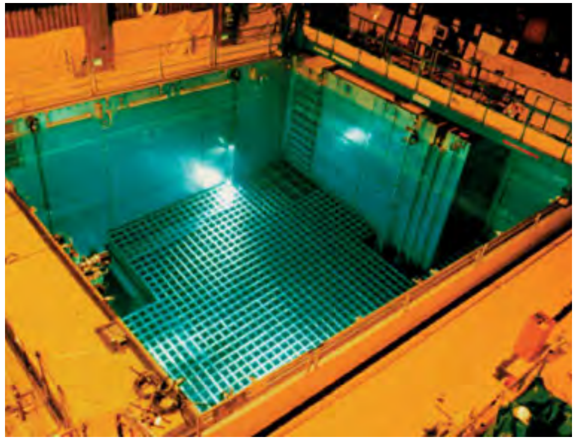
Decay Heat: EPRI-SKB Collaboration for Extending Validation Range

Hatice Akkurt
Used Fuel High Level Waste Management Program

ACRS Meeting

May 18, 2023

Decay Heat is an Important Parameter That Impacts



Spent Fuel Pool (SFP)

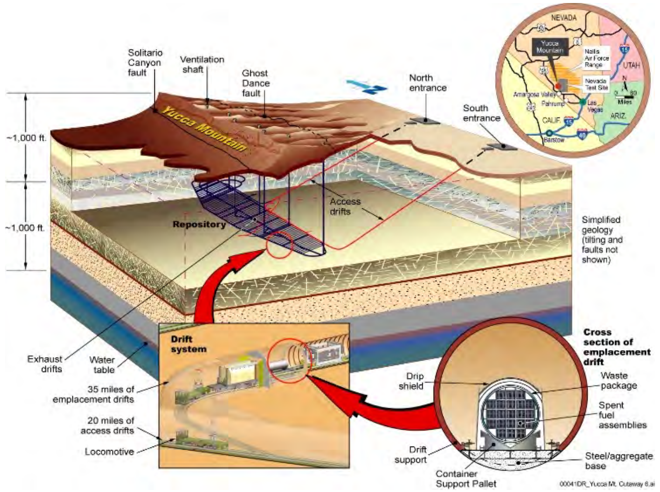
- SFP Heat Management
- Available storage capacity (due to limits for fuel off-loading time)

Dry Storage & Centralized Storage

- Loading (Thermal limits – decay heat value and profile)
- Fuel/Cladding Integrity
- Canister Integrity
- Fuel/Clad and Canister integrity impacts are in opposite directions

Transportation

- Transportation limits



Disposal

- Heat load management
- Dictates number of casks/canisters that can be stored in repository

Reasonably accurate estimation of decay heat is important for the entire back-end

Background: Decay Heat PIRT Report – Identified Technical Gaps

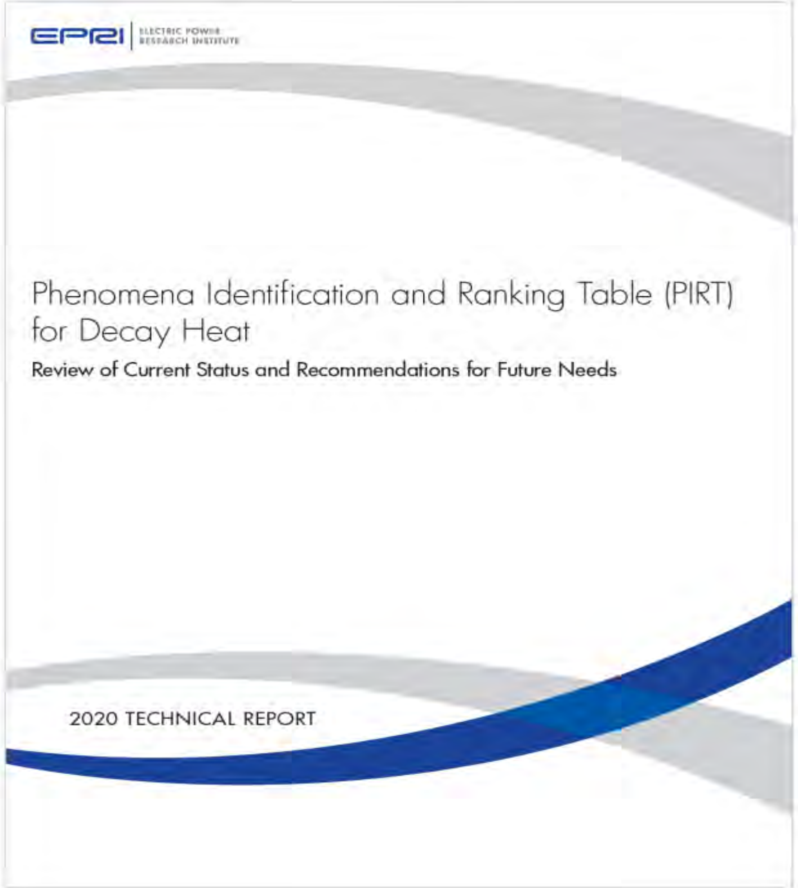
<u>Recommendation:</u> Publication of "unpublished" Clab decay heat measurements (see next slide)

<u>Gaps:</u> Need to expand parameter space for validation

1. Decay heat measurements for
   - Shorter cooling time (1-2 years)
   - High burnup (above 51 GWd/MTU)

2. For **advanced fuels**, decay heat measurements for increased enrichment (above 4.0%) and increased burnup are needed

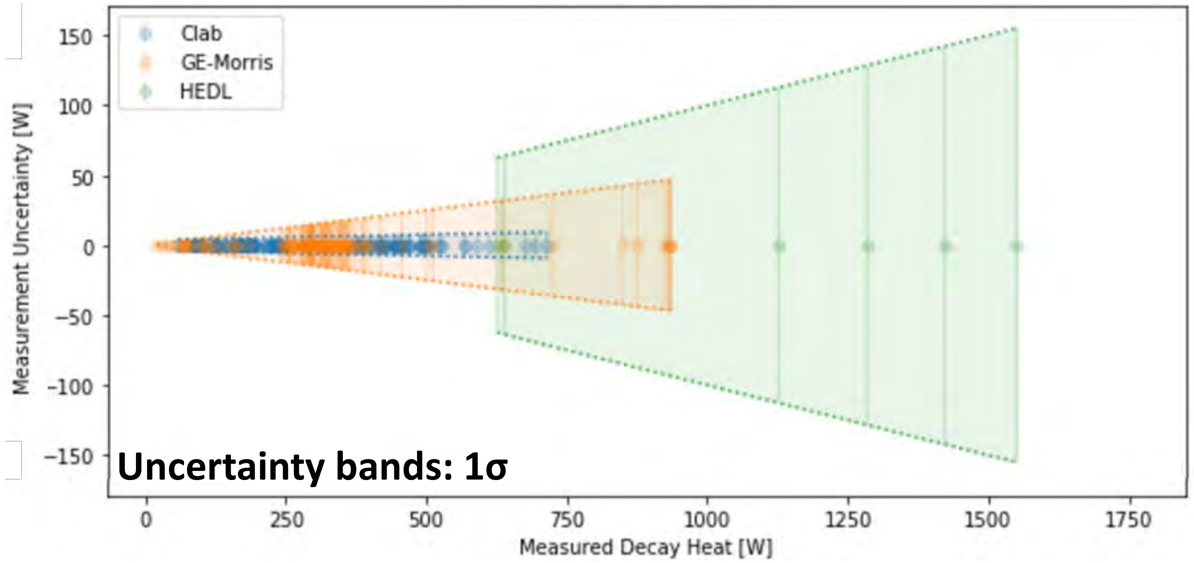
Parameter	Published Data
Decay Heat (W)	25–1550
Cooling Time (years)	2.5–27
Burnup (GWd/MTU)	up to 51
Enrichment (wt%)	1.1–4.0

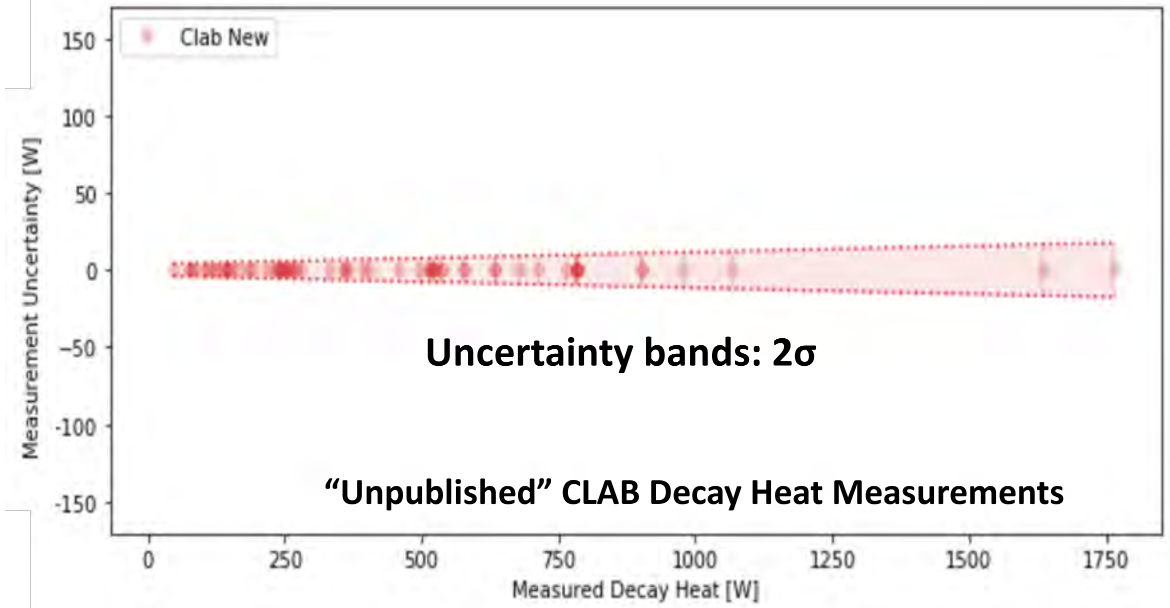


EPRI report **3002018440**, published July 30, 2020 – Publicly available

EPRI

Decay Heat (DH) Measurements – Published and Unpublished





- **HEDL:** Large measurements uncertainty; no other measurements for high DH range ➔ can't be taken out of validation set yet
- **GE-Morris:** Measurement quality issues at higher DH; no other measurements ➔ can't be taken out of validation set yet
- **CLAB:** Low measurement uncertainty; focus on low DH and only facility that continues measurements

- Over 60 new DH measurements that are not published
- Only two measurement points for high decay heat values ➔ Can it be increased?
- High quality data ➔ better validation set ➔ decrease DH uncertainty and increase margins for global industry

Recommendation: Publication of unpublished CLAB measurements and performing new measurements to close the gaps ➔ EPRI initiated a collaborative project with SKB

EPRI-SKB Collaboration Agreement and Ongoing Collaborative Efforts

EPRI-SKB collaboration agreement signed in December 2020. Collaboratively working on the following tasks:

1. **Publication of unpublished CLAB decay heat measurements**
   - EPRI received > 150 unpublished decay heat measurements
     - Since agreement signed, SKB performed additional measurements using existing calorimeter for
       - High burnup, shorter cooling time, and GE14 fuel
   - Report will be a publicly available EPRI report
2. **Validation of decay heat measurements**
   - Using SNF and ORIGEN codes
   - Validation report will be a publicly available EPRI report
3. **Performing new decay heat measurements to close remaining technical gaps**
   - Building a new calorimeter to enable decay heat measurements for shorter cooling times
   - Performing new and repeat decay heat measurements

Parameter	Unpublished Measurements from Clab
Decay Heat (W)	up to 1725
Cooling Time (years)	1.5–35
Burnup (GWd/MTU)	up to 55
Enrichment (wt%)	up to 4.1

Parameter	Capacity of Current Clab System	Remaining Gaps
Decay Heat (W)	up to 2000	**2000–4500**
Cooling Time (years)	1–43	**1–1.5**
Burnup (GWd/MTU)	up to 60	**Above 55**
Enrichment (wt%)	up to 4.5	**Above 4.5**

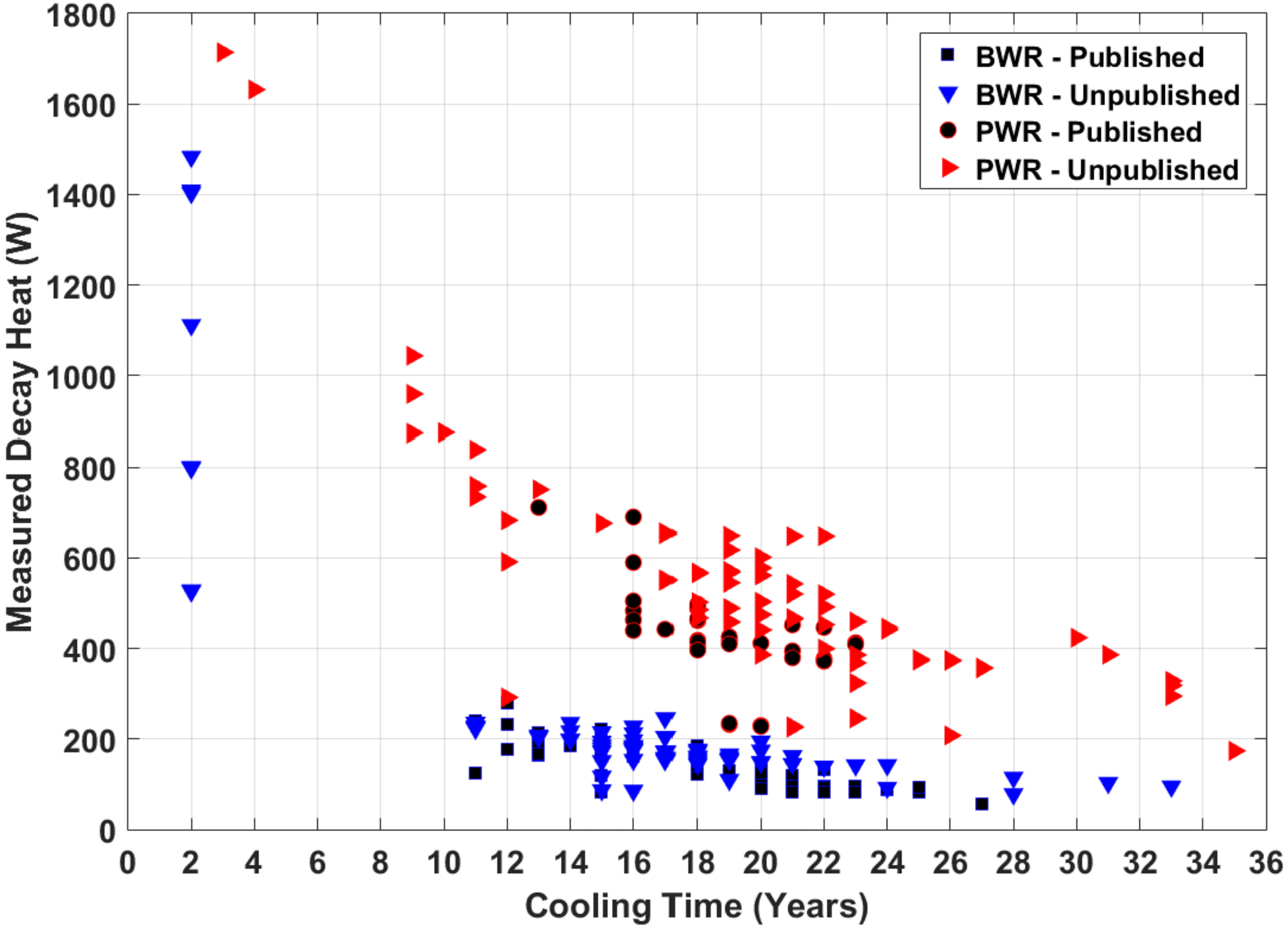
What is Clab?

Intermediate wet storage for the whole Swedish nuclear program. In total 13 NPPs, (1 PHWR, 3 PWR, 9 BWR)

Today 33000 fuel assemblies with enrichments between 0,7-4,6 % U-235 and BU between 0-61.2 MWd/kgU. Cooling times between1-40 years.

Unique possibilities to do decay heat measurements on a variety of fuel types and fuel with different characteristics.

Extending Validation Range for Decay Heat

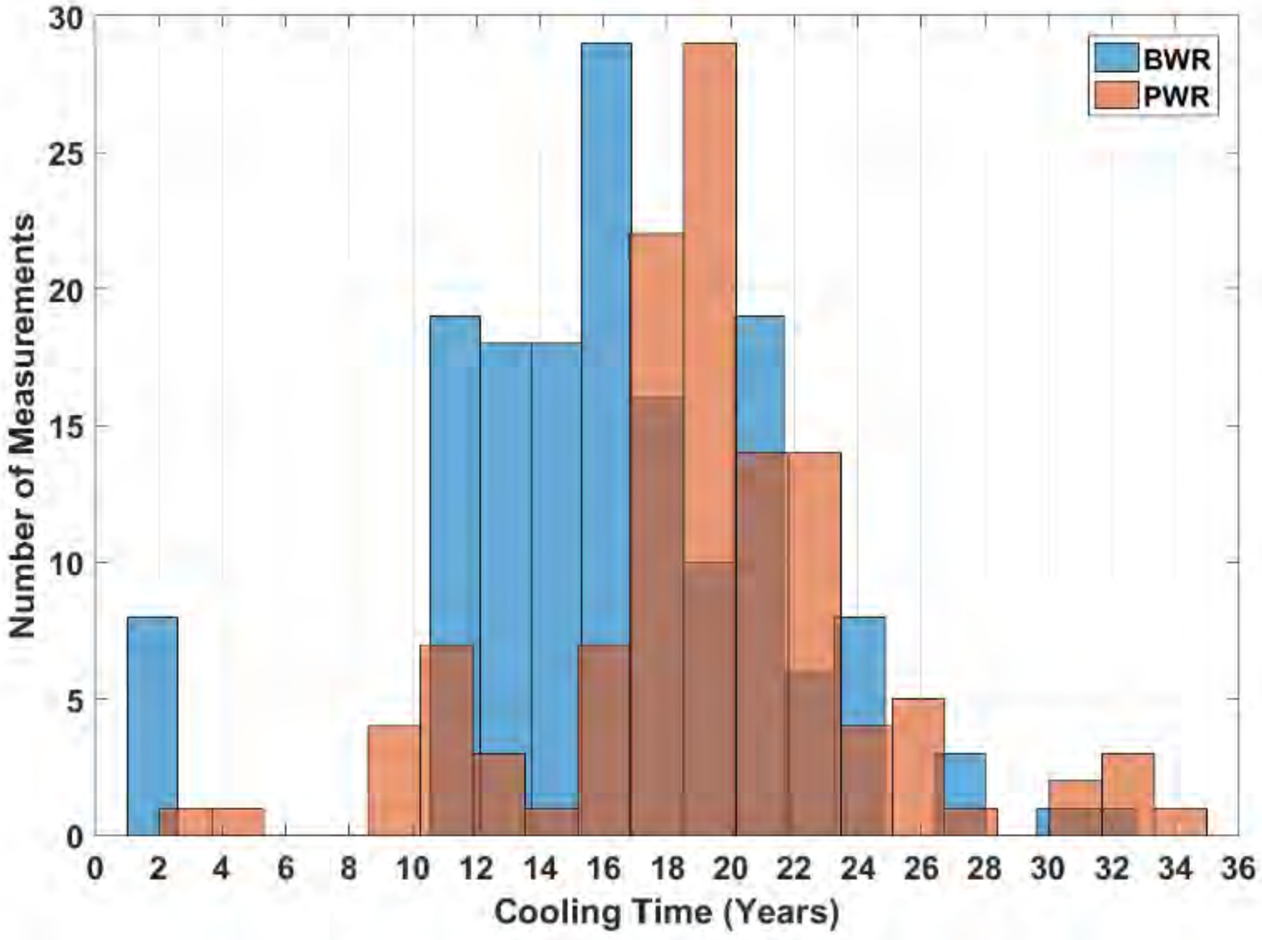


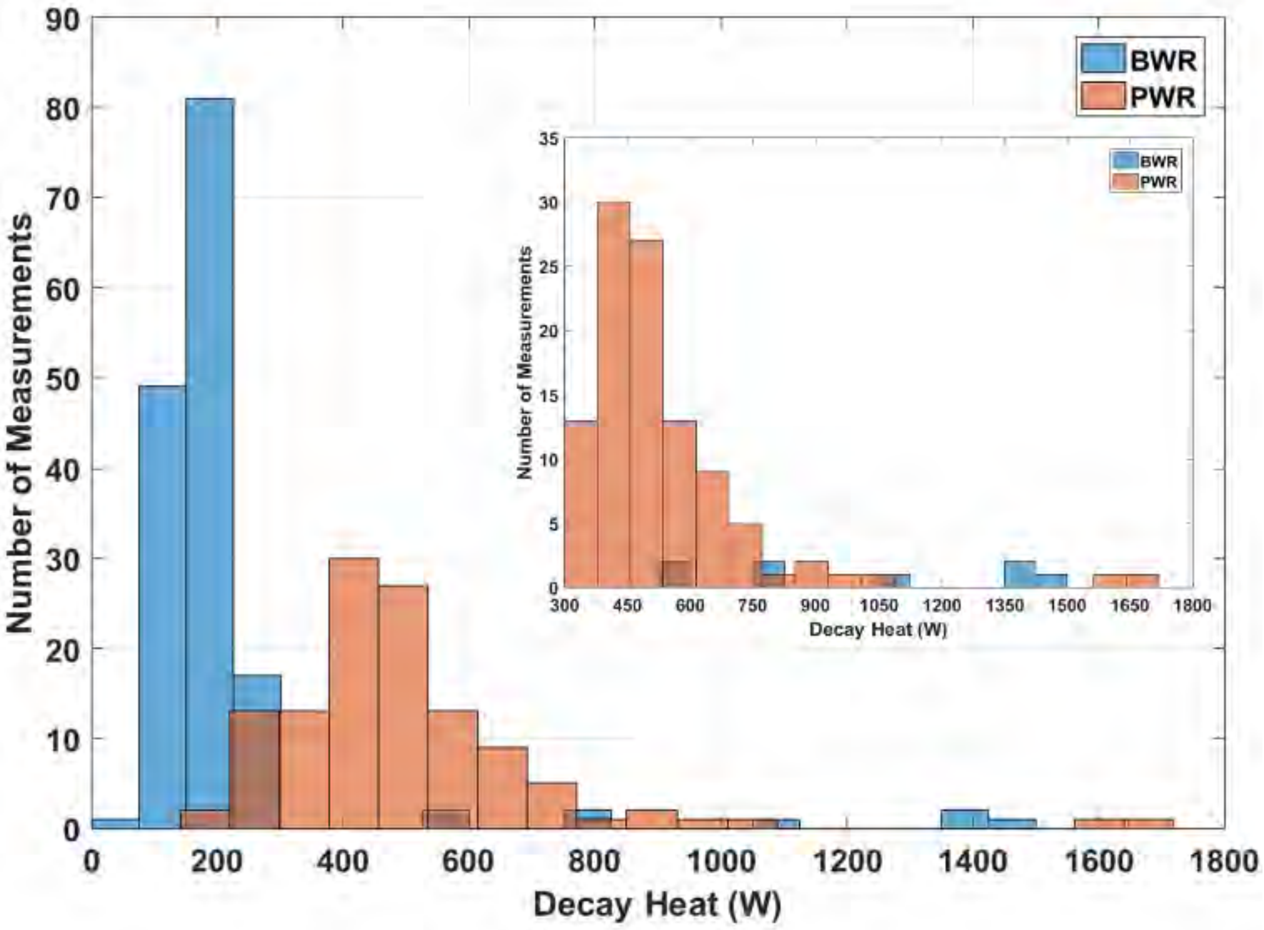
	SKB report*	Unpublished
Measurement interval	2003-2004	**2005-2021**
Number of Measurements	109	**166**
Enrichment range (%)	2.1-3.4	2.1-4.1
Burnup range (GWd/MTU)	15-51	20-55
Cooling time range (Years)	11-27	**1.5-35**
Decay heat range (W)	55-710	**70-1725**

*Published in SKB Report R-05-62 in 2006

Unpublished measurements significantly extends decay heat validation ranges for cooling time and decay heat

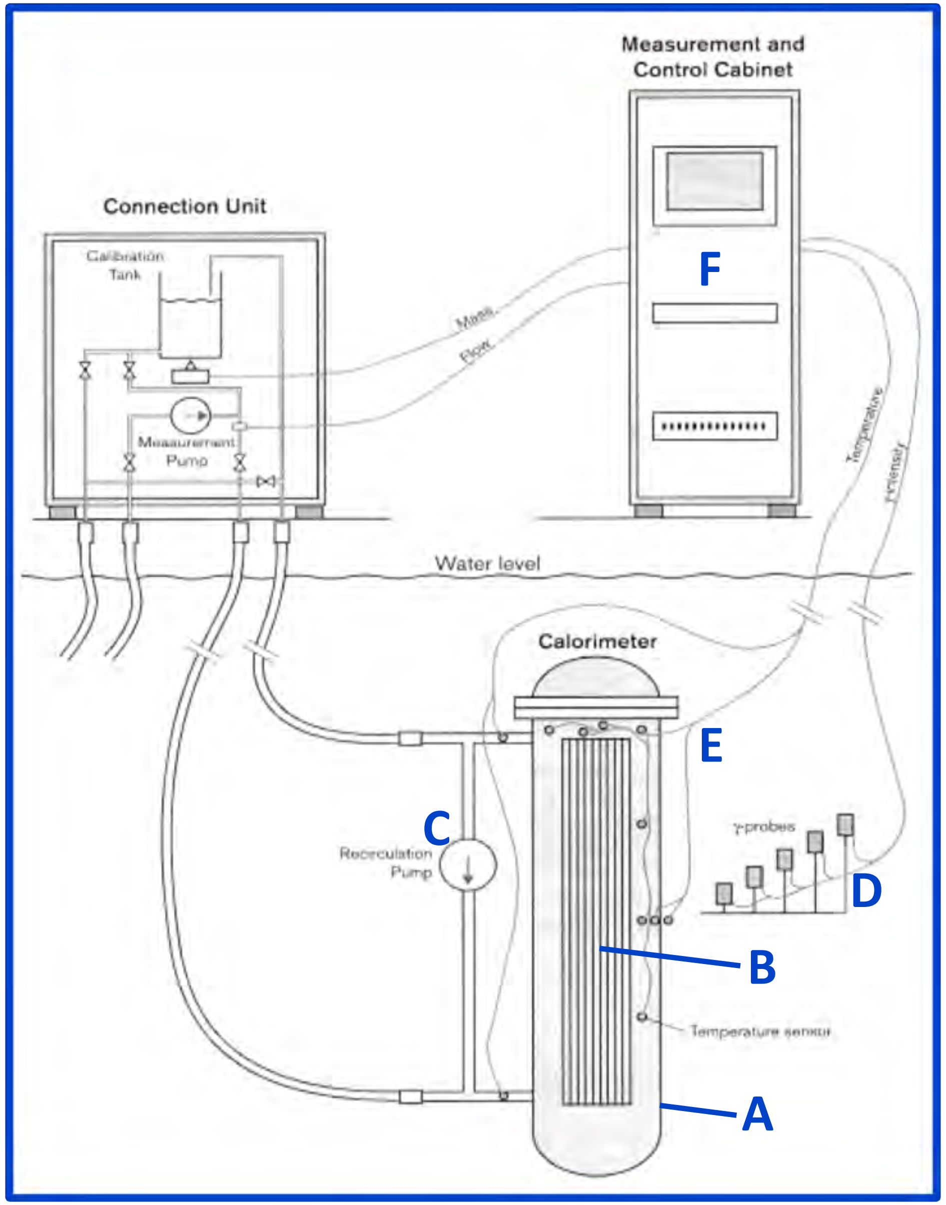
Distribution of Clab Decay Heat Measurements





To date, measurement focus for lower decay heat and longer cooling times

2002 Calorimeter Schematic



Key Calorimeter Components

A. Insulated Container

B. Heater Assembly or Fuel Assembly

C. Circulation Pump

D. Gamma Detectors

E. PT100 Resistance Temperature Detectors

F. Data Acquisition and Recording

G. Heater Power Cable (Not Shown)

H. Pump Power Cable (Not Shown)

Sources of Uncertainty - Components

Calibration (Heater) Measurement

Component	Uncertainty
Calorimeter	Heat loss to pool
	RTD accuracy
Pump	Heat added to calorimeter
Data analysis	T vs. time curve fit slope uncertainty
Heater	Power variation
	Power measurement accuracy
	Power cable losses

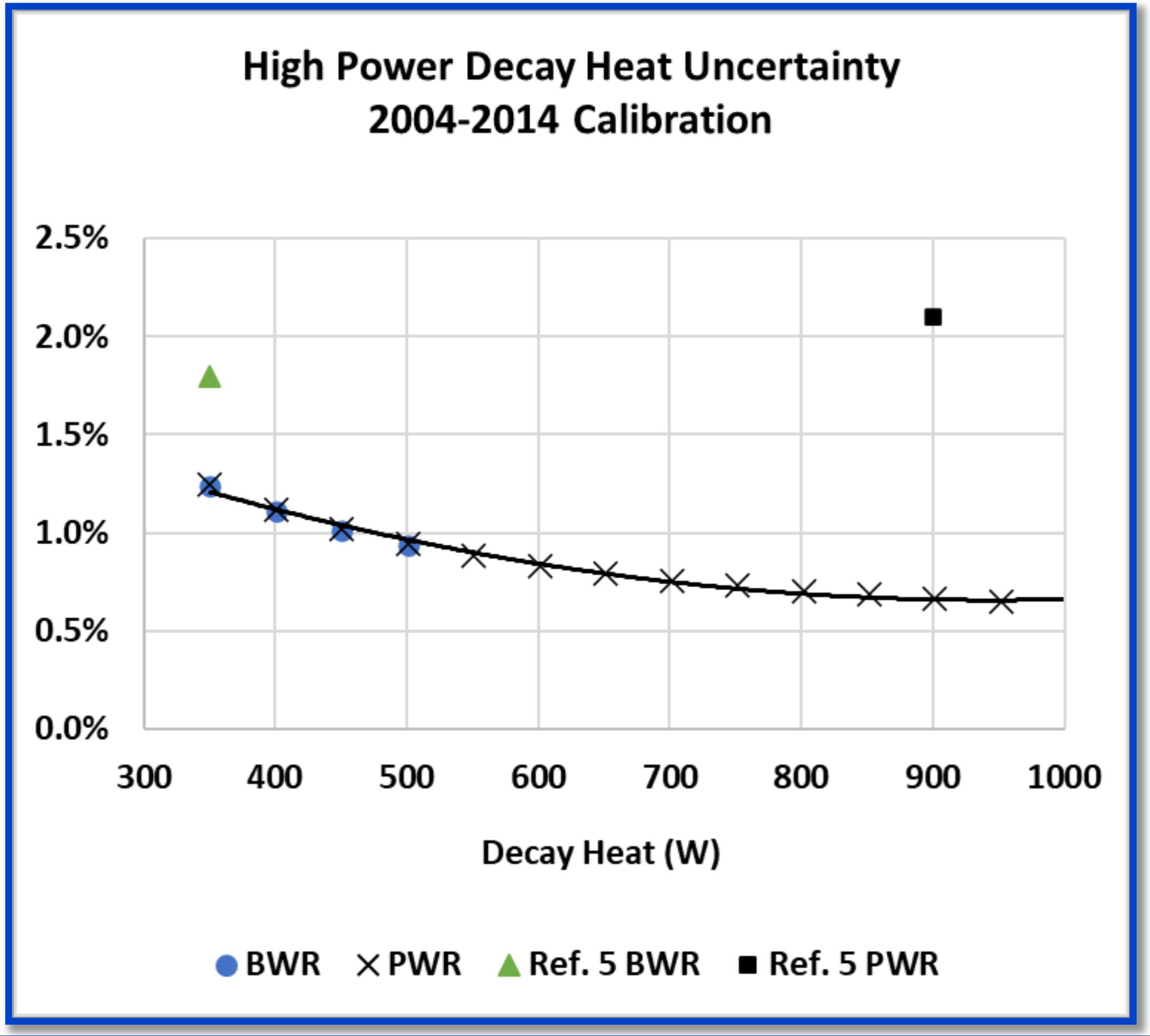
Fuel Assembly (FA) Measurement

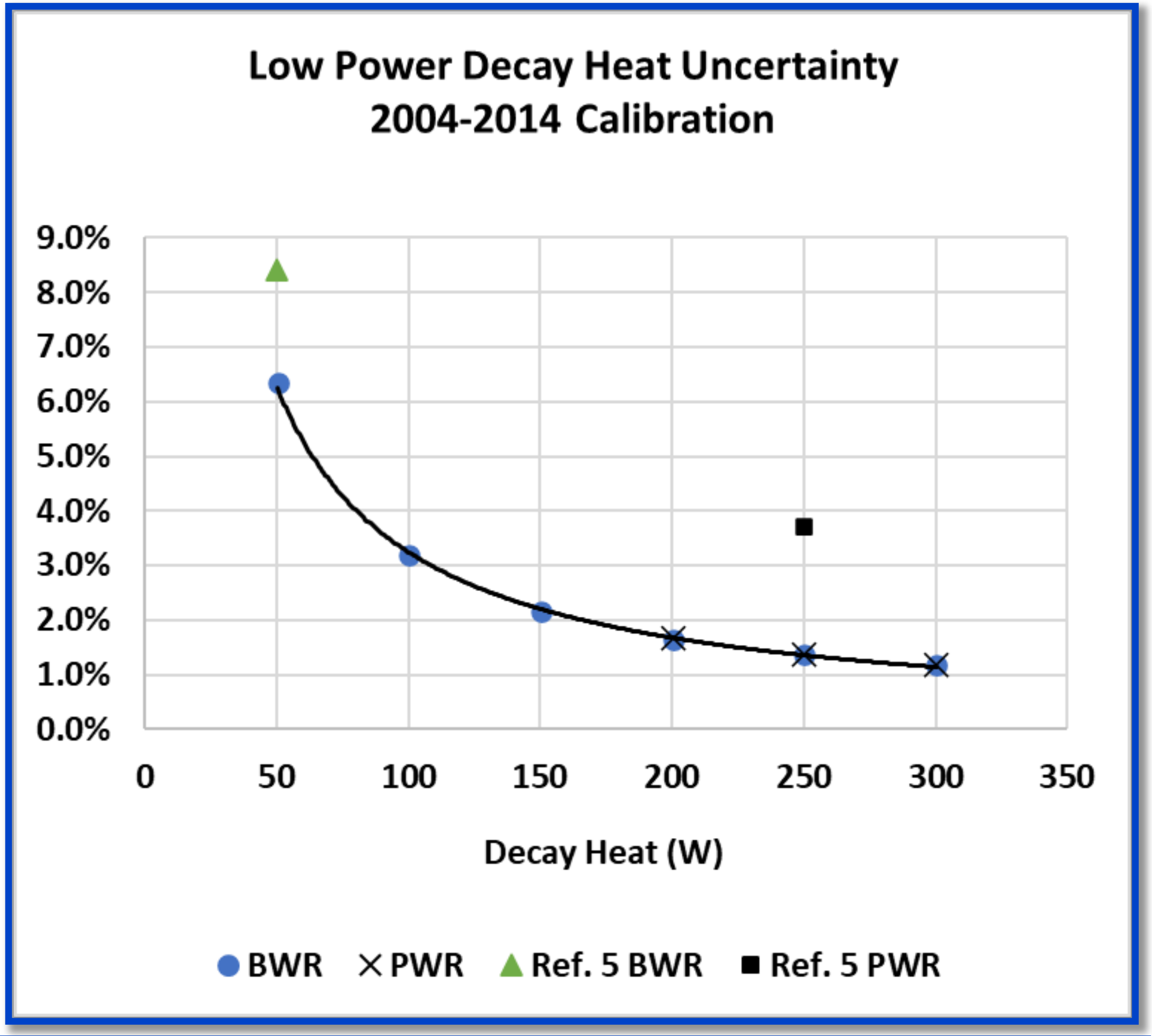
Component	Uncertainty
Calorimeter	Heat loss to pool
	RTD accuracy
Pump	Heat added to calorimeter
Data analysis	T vs. time curve fit slope uncertainty
Corrections	Heat capacity difference vs. heater + water
	Gamma energy loss

Calibration Curve Prediction Interval

FA Measurement Uncertainty: Some Common Components and Some Unique Components

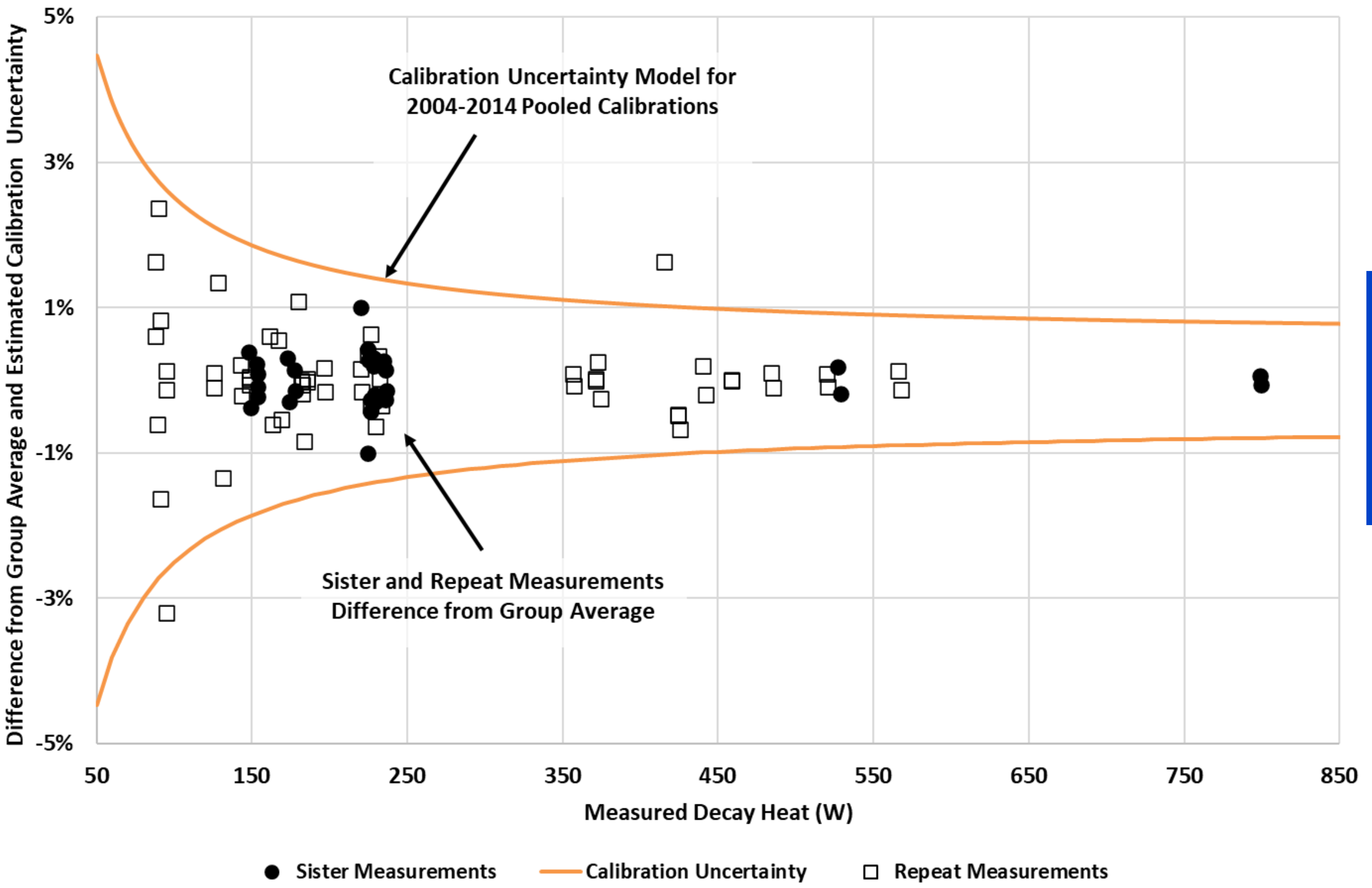
Clab Decay Heat Measurement Uncertainty Assessment





Uncertainty estimate is lower than 2006 report (SKB R-05-62) estimate

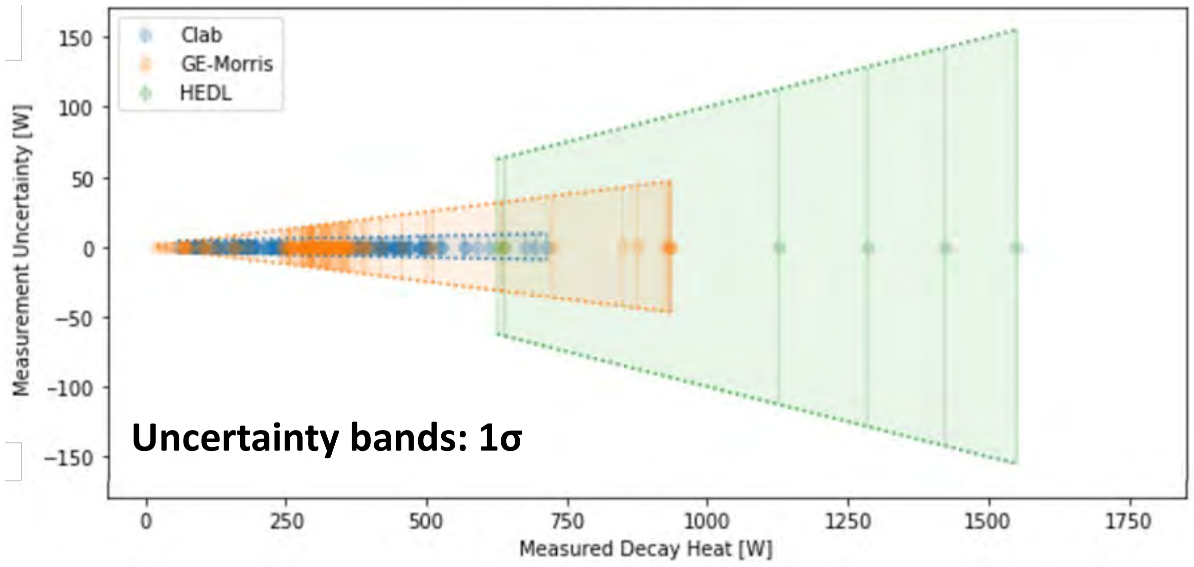
Assessment of Uncertainty Evaluation with Repeat and Sister Assembly Measurements

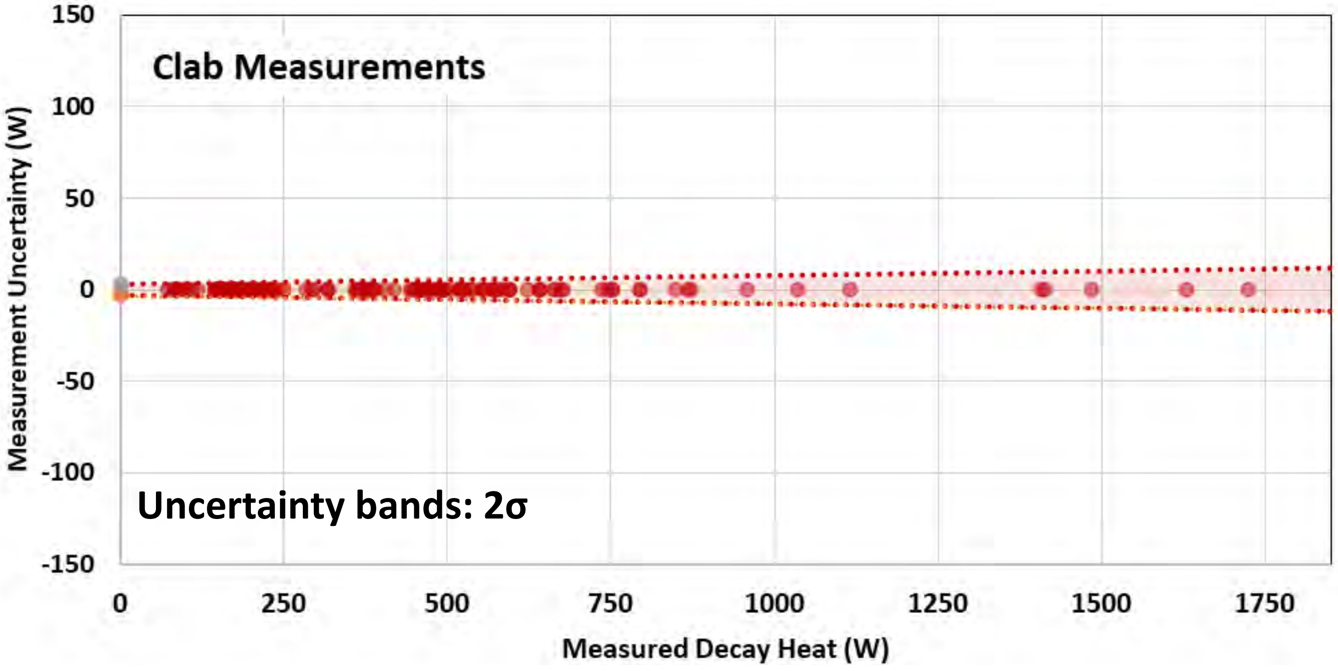


- 60 repeat measurements
- 28 sister assembly measurements

Only 2 (out of 88) points outside the uncertainty band – Very good agreement

Recommendations for Decay Heat Validation Set





- By including all Clab measurements, decay heat validation range is extended significantly.
- Therefore, after the publication of these measurements, recommending:
  - Removing HEDL measurements from validation set (large uncertainties and only few points)
  - If desired, selected GE-Morris measurements can be used but recommend use of inverse uncertainty weighting

Extending decay heat validation range with low uncertainty measurements will benefit LEU+ fuel

Summary and Future Work

1. **Decay heat PIRT identified a number of recommendations and data gaps**
    - Recommendation: Publication of "unpublished" Clab measurements
    - Gaps: Extending decay heat validation range for higher burnup, shorter cooling time
2. **EPRI and SKB signed collaborative work in December 2020**
    - Analyzing all measurements, including measurement uncertainty
    - Measurement report expected to be published, as publicly available EPRI report in late 2023
    - Also working on validation report (using Origen and SNF codes), which will be a published in a publicly available EPRI report
3. **Performing new decay heat measurements to close remaining technical gaps**
    - Upgrading the calorimeter to enable decay heat measurements for shorter cooling times and higher burnup and filling the gaps for low-high decay heats
    - Measurement campaign will start in 2024 with repeat measurements
    - EPRI report for new measurements and validation results are expected to be published in late 2024

Publications

1. H. Akkurt, R. Hall, F. Johansson, A. Mehic, J. Kierkegaard, H. Liljenfeldt, "*Decay Heat Evaluation for Extended Validation Range Using Clab Measurements*," to appear in Proceedings of PATRAM 2023, June 23.
2. H. Liljenfeldt, H. Akkurt, R. Hall, F. Johansson, J. Kierkegaard, "*Decay Heat Measurements and Analysis for BWR Fuel with Shorter Cooling Time*," Proceedings of International High Level Radioactive Waste Management Conference (IHLRWM 2022), November 2022.
3. F. Johansson, J. Kierkegaard, H. Liljenfeldt, R. Hall, H. Akkurt, "*Extending the Validation Range for Decay Heat Measurements*," Proceedings of Global 2022 conference.
4. H. Akkurt, H. Liljenfeldt, G. Ilas, S. Baker, K. Banerjee, J. Scaglione, "*Parameter Identification and Ranking Table (PIRT) for Decay Heat*", Proceedings of Top Fuel 2021, October 2021.
5. *Phenomena Identification and Ranking Table (PIRT) for Decay Heat: Review of Current Status and Recommendations for Future Needs*. EPRI, Palo Alto, CA: 2020. 3002018440.

Questions?


Together...Shaping the Future of Energy™

Scoping Analysis for Decay Heat and Radiation Dose for ATF/LEU+/HBU

Preliminary Scoping Results

Bob Hall
EPRI Used Fuel and High-Level Waste Program
May 18, 2023

ACRS Meeting

www.epri.com

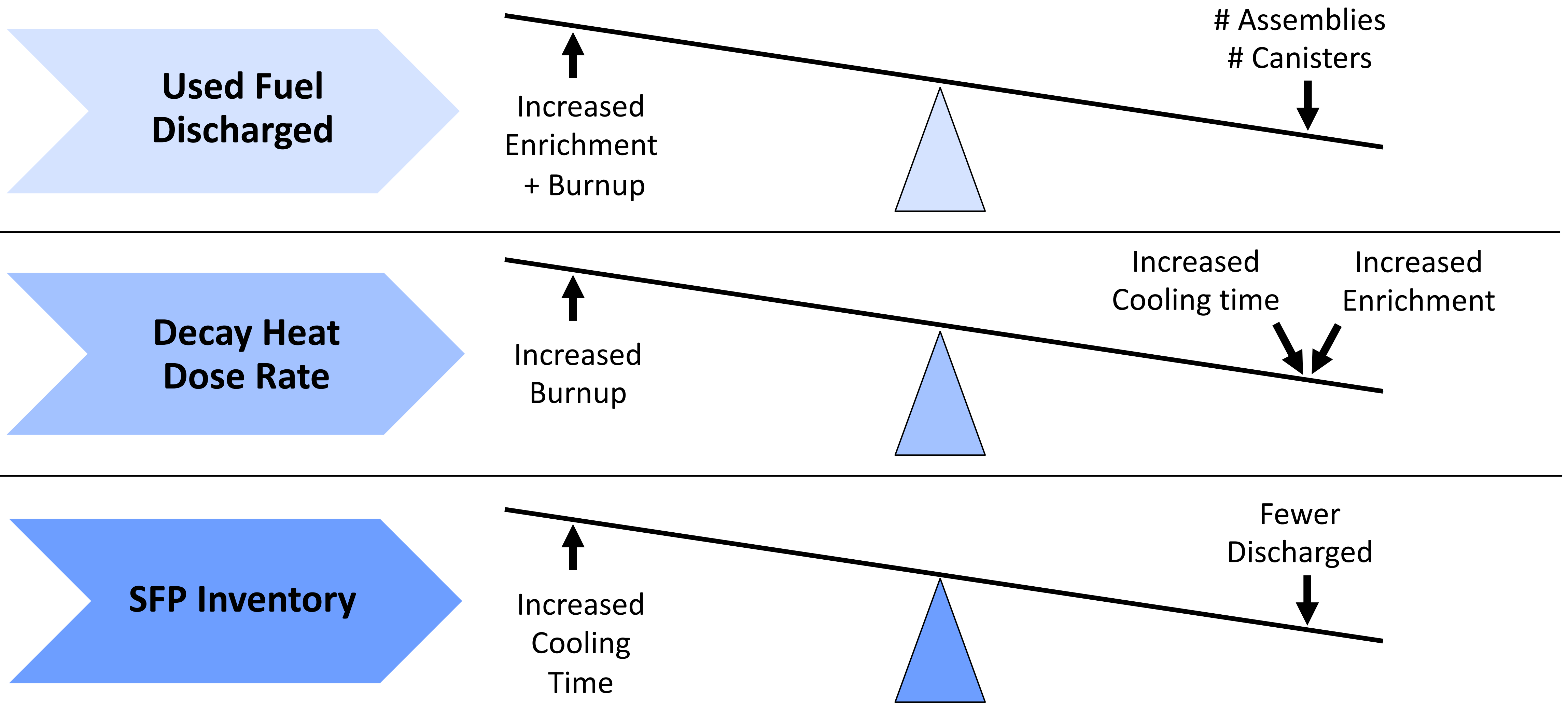
5/7/23 Rev. 0

Scoping Calculation for HBU – Purpose and Goals

- **Better understand effect of higher burnup (HBU) fuel on used fuel storage**
  - Decay heat (DH) and dose rate impacts of higher discharge burnup (DBU)
  - SFP inventory and heat load, transfer to ISFSI and dry storage dose rates
- **Estimate PWR equilibrium impacts of shift to increased burnup and enrichment**
  - 18-month cycle base case, 62 GWd/MTU peak rod burnup limit
  - 18-month cycle HBU case, 75 GWd/MTU peak rod burnup limit
  - 24-month cycle HBU case, 75 GWd/MTU peak rod burnup limit
- **Estimate batch average or sub-batch average enrichments and discharge burnups**
- **Identify key variables**
- **Estimate SFP inventory and decay heat trends, ISFSI dose rate trend**
- **High-level scoping, simplifying assumptions, not plant or dry storage system specific**

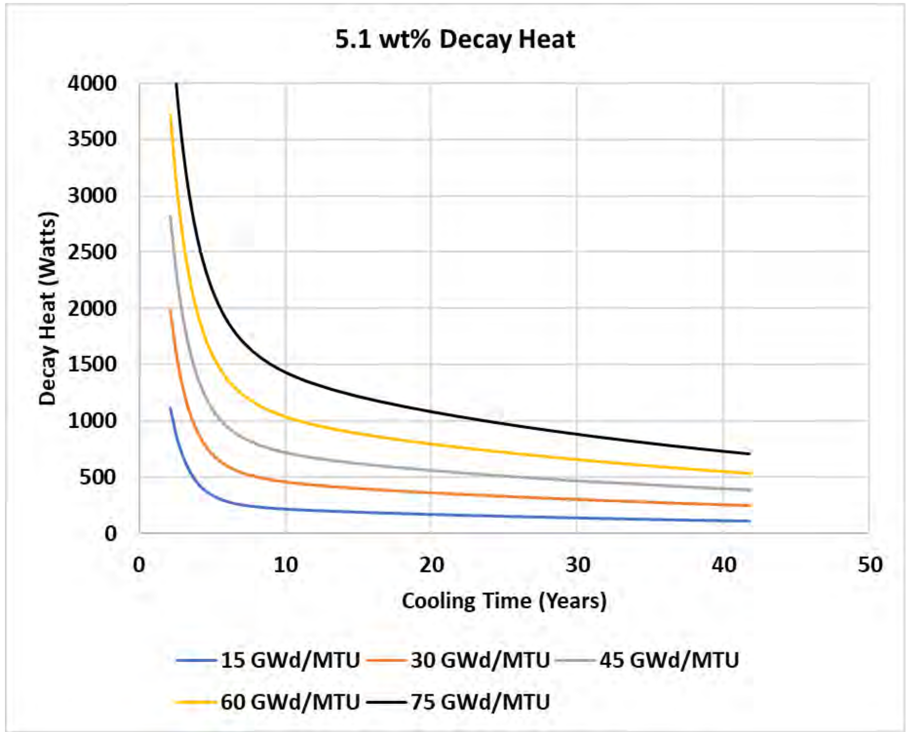
****Results are under review/preliminary, white paper publication in June****

LEU+/HBU: Decay Heat, SFP Inventory, and Dry Storage Dose Rates

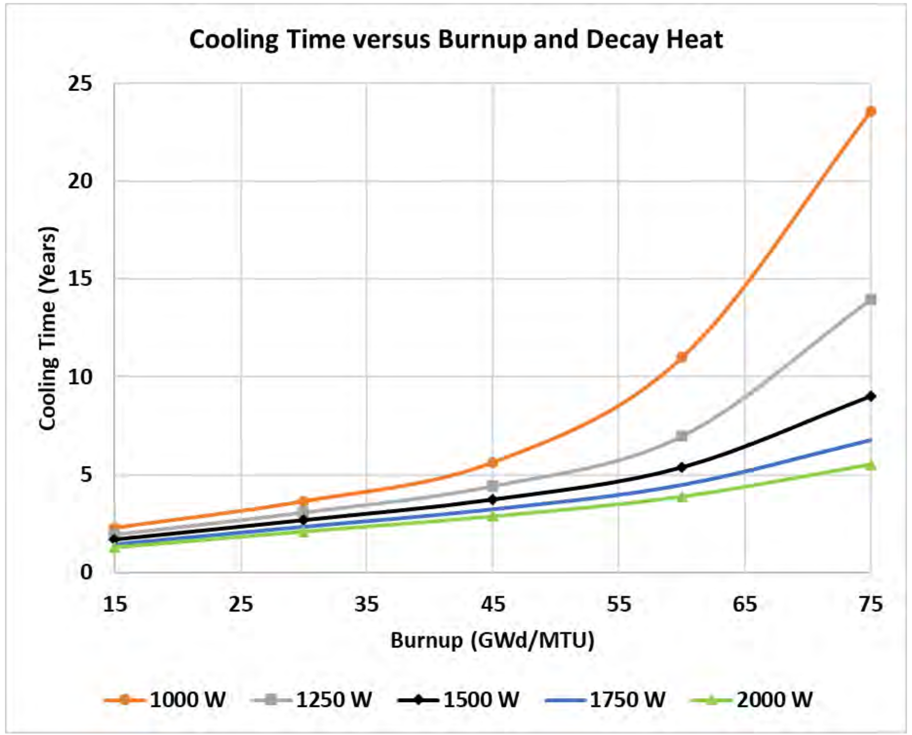


Multiple opposing and offsetting effects, want to know net impacts.

Decay Heat, Burnup, and Cooling Time



- 17x17 PWR
- Constant core-average power depletion
- Increased burnup increases DH
- Increased enrichment reduces DH
- **Cooling time is a strong function of DH limit at high burnup**



Use of realistic burnup and enrichment combinations is important

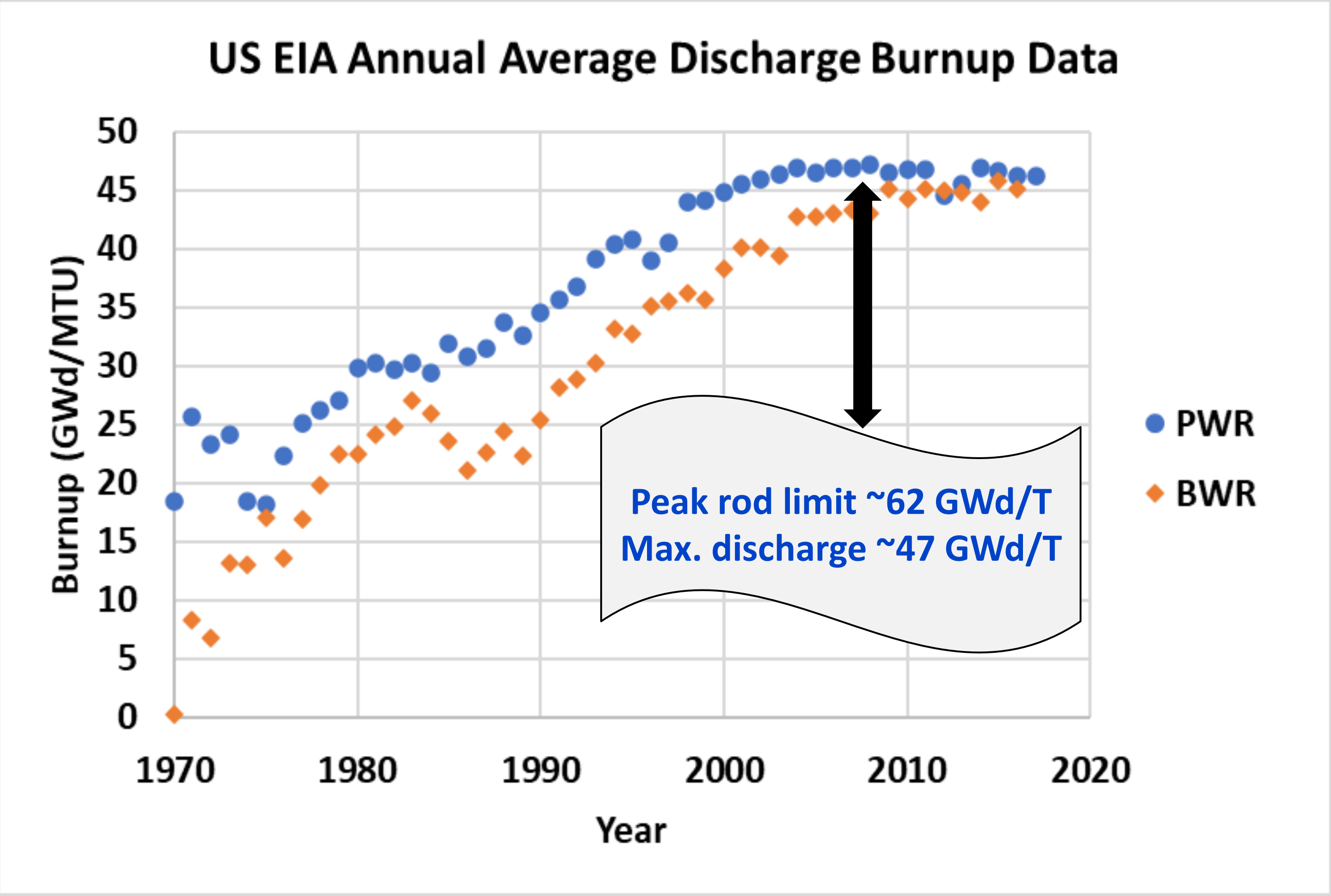
Evaluation Needs Realistic Burnups and Enrichments

Limit
- Peak rod burnup

Design
- Peak rod > max. assembly
- Max. assembly > max. sub-batch
- Max. sub-batch > batch average
- Design code uncertainty > 0
- Operational uncertainty > 0

Average
- Batch Average Burnup*

* **Maximum practical batch average burnup ~80% ±5% of peak rod limit**
(Table 8, GC-859 data, ORNL/TM-2021/1961)



https://www.eia.gov/nuclear/spent_fuel/ussnftab3.php

Need to use realistic burnup and enrichment combinations for 62 and 75 GWd/T limits

Realistic PWR Batch Average Enrichments and Burnups

Current PWR cycles

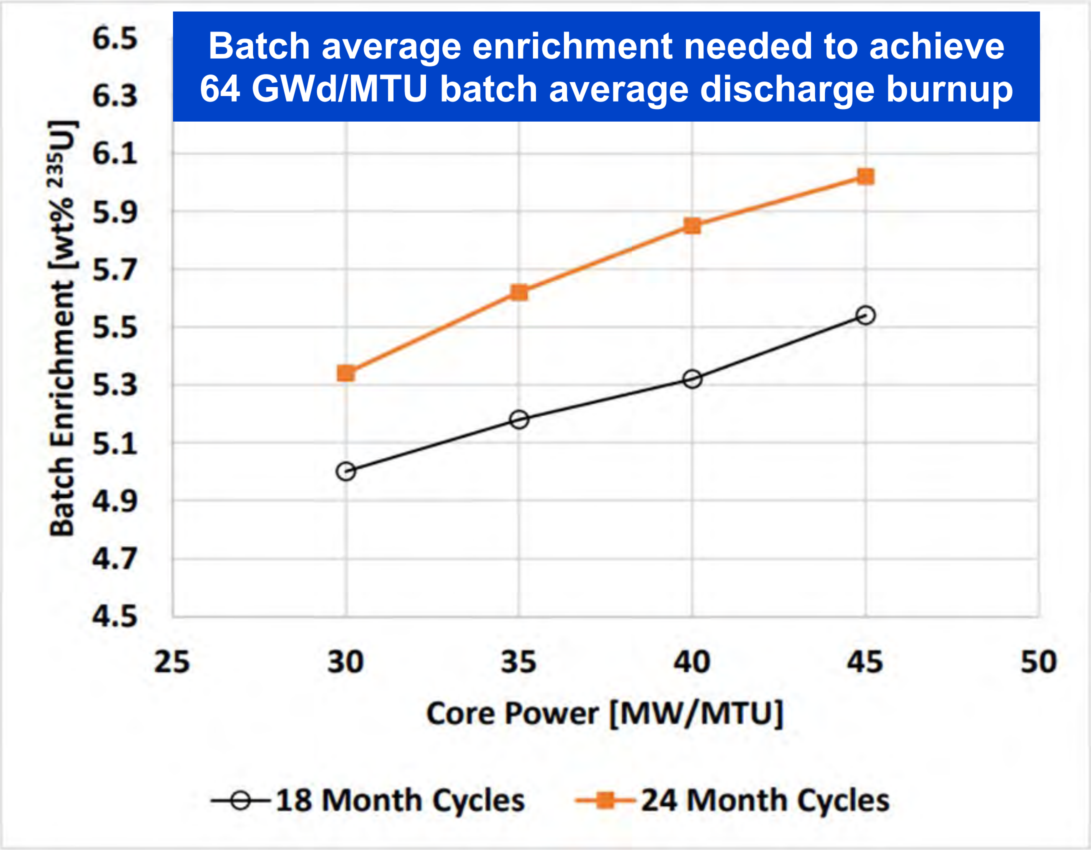
Batch average discharge burnup (DBU), using GC-859 data

- 44 GWd/MTU @ 4.0 wt%
- ~11 GWd/MTU/wt% ^{235}U initial enrichment
- Varies with cycle length, specific core power
- 0.1 to 0.5 wt% for split batch

Scoping assessment PWR cycles

Use ORNL/TM-2021/1961 tool for enrichment, batch size, DBU

- Cycle estimator tool Benchmarked to GC-859, 6 PWRs, 26 cycles, and 2 24-month cycle studies
- Batch average enrichment within 0.2 wt% ^{235}U
- **Estimate 18- and 24-month cycle enrichments and DBU**
  - **51 GWd/MTU batch DBU for 62 GWd/MTU max. pin**
  - **62 GWd/MTU batch DBU for 75 GWd/MTU max. pin**
  - **Final partial batch DBU limit 5% higher**



Courtesy of Oak Ridge National Laboratory, U.S. Dept. of Energy

Data from ORNL/TM-2021/1961

Analysis Cycles

- Core size 157 assemblies
  - 17x17 PWR, 39 MW/MTU, 0.47 MTU/assembly
  - 98% load factor
  - Reload batches are average and equilibrium
    - Some values not realistic for a single reload (e.g., 2 assembly sub-batch)

Case	Cycle Length	Batch Enrichment	Sub-Batch 1 Assemblies (2 cycles)	Sub-Batch 1 DBU	Sub-Batch 2 Assemblies (3 cycles)	Sub-Batch 2 DBU	Average DBU
1 (Base)	18 months (20 day outage)	4.4 wt%	41	45.3 GWd/MTU	25	53.3 GWd/MTU	48.3 GWd/MTU
2 (HBU 18)	18 months (20 day outage)	5.1 wt%	2	47.9 GWd/MTU	51	60.7 GWd/MTU	60.2 GWd/MTU
3 (HBU 24)	24 months (23 day outage)	5.4 wt%	65	56.8 GWd/MTU	9	64.8 GWd/MTU	57.8 GWd/MTU

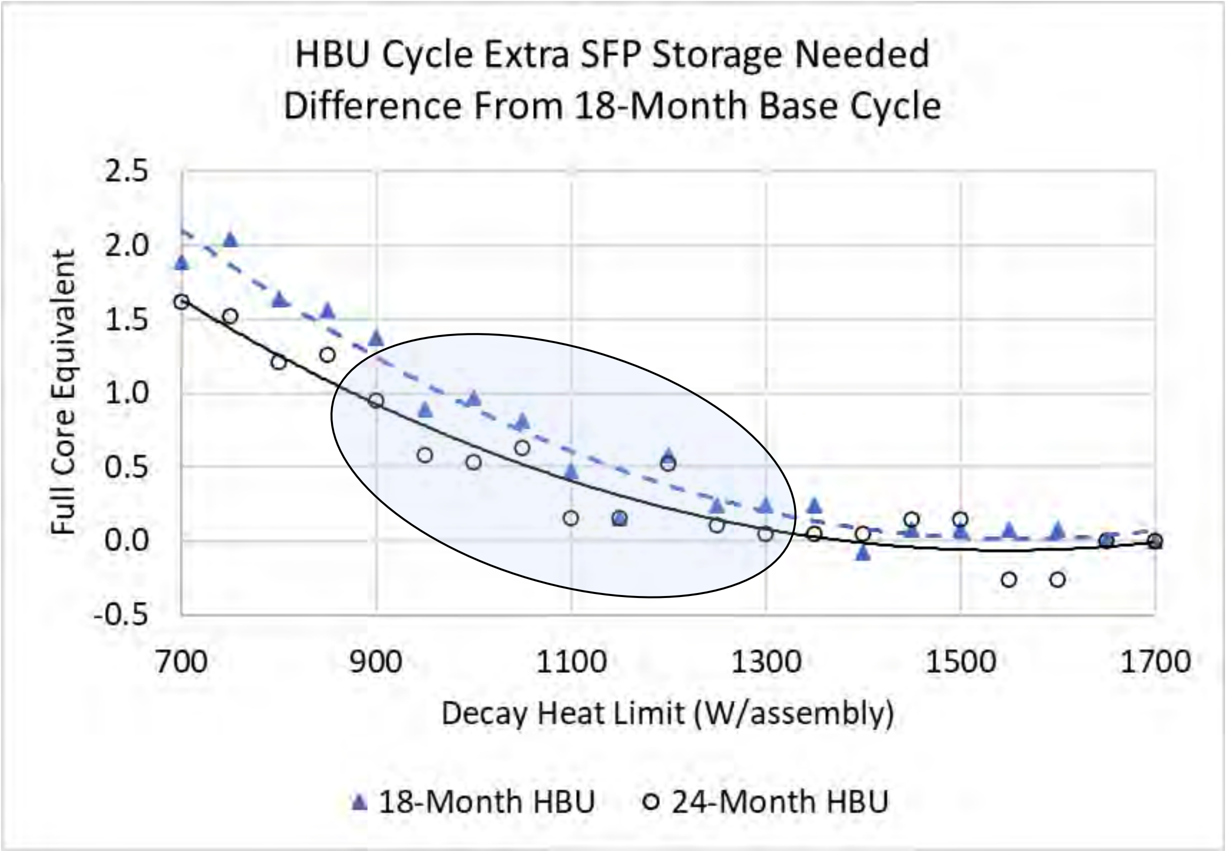
Canister Decay Heat Limits

Simplifying assumptions

- Arbitrary canister size for mock loadings
  - Capacity assumed equal to batch size
  - Attempt to load longest decay time sub-batch first by zone
  - Partial canister load allowed
  - Test case with fixed canister size produced similar results
- Representative of a range of current 37 assembly canister types
  - 42.5 to 50 KW total DH
- 5% conservatism applied to DH limits for simulation
- Calculate difference in un-loadable SFP population and peak SFP decay heat change

Canister 1 (uniform)	**Zone 1**	**Zone 2**	**Zone 3**
Decay heat limit (W)	1149	N/A	N/A
Zone fraction	1.0	N/A	N/A
Canister 2 (zoned)	**Zone 1**	**Zone 2**	**Zone 3**
Decay heat limit (W)	1000	2000	1313
Zone fraction	0.35	0.22	0.43
Canister 3 (zoned)	**Zone 1**	**Zone 2**	**Zone 3**
Decay heat limit (W)	874	1700	890
Zone fraction	0.24	0.32	0.43

Simple HBU / SFP Inventory Illustration



- Equilibrium SFP inventory
- Single dry storage DH limit
- Smaller HBU batch size reduces SFP inventory
- Increased burnup increases cooling time and SFP inventory
- **Net effect is higher SFP inventory for current average canister DH limits**
- **DH limit is a key variable**

Additional SFP space needed for HBU is a strong function of canister decay heat limit

HBU SFP Inventory Change Simulation Results

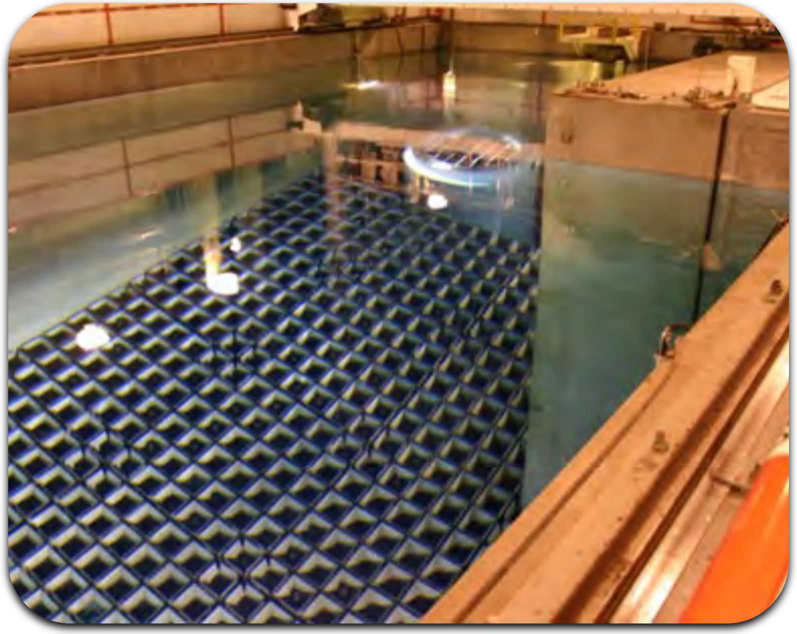
Canister	Loading	18 Month HBU	24 Month HBU
1	Uniform	0.2	0.2
2	Zoned	0.6	0.3
3	Zoned	1.3	0.8

- Units are full core equivalents of additional SFP fuel inventory due to HBU cycles
- Varies by DH limit, DH zoning (particularly low DH zone limit), cycle type, etc.
- Somewhat higher impact on 18-month HBU cycles (higher batch average DBU than 24-month)
- Due to multiple important variables, suggest plant/cycle/canister specific assessment

Results are broadly similar to simple illustration

HBU SFP Maximum Decay Heat* Simulation Results

Case	Canister	HBU increase in maximum DH
HBU (18 month)	1	2.8%
	2	3.9%
	3	4.9%
HBU (24 month)	1	1.9%
	2	2.3%
	3	3.0%



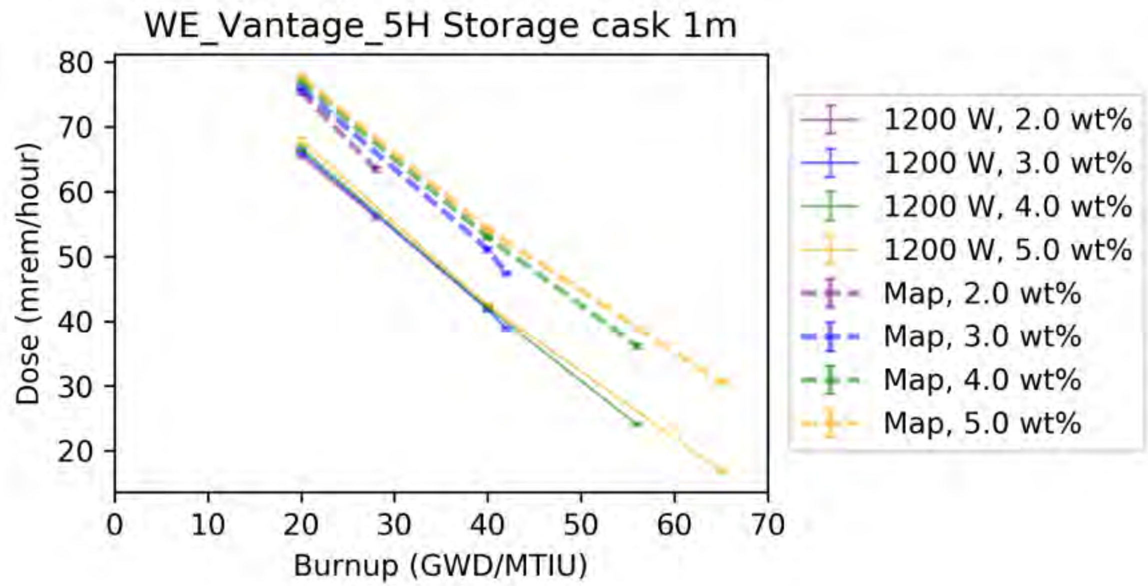
*Maximum decay heat occurs at the end of core offload. Short cooling time (~140 hours) core decay heat is a weak function of burnup. Large majority of peak decay heat is from just-offloaded core. Increase is from increased burnup of offloaded core and increased SFP inventory.

Modest increase in maximum SFP heat load

Dose Rate Trends – ORNL/SPR-2020/1441 Study

ORNL "isocaloric" dry storage canister dose rates

- Uniform loading with all assemblies 1200W decay heat
- Different enrichment / burnup combinations
  - 3.0 wt% / 42 GWd/MTU
  - 4.0 wt% / 56 GWd/MTU
  - 5.0 wt% / 65 GWd/MTU
- Increased decay time required more than offsets increased burnup for gamma dose rate
- Gamma dose rate dominates in normal storage conditions
- Neutron dose rate change a mixed bag
  - Depends on specific enrichment/decay time/burnup



Courtesy of Oak Ridge National Laboratory, U.S. Dept. of Energy

Normal storage condition dose rates decline due to HBU

Dose Rate Trends – EPRI Cask Loader Software Simulation

Dry storage system normal storage

- Comparison of 18-month cycle base case and 24-month cycle HBU case
- Normal dry storage configuration
- Uniform 1100 W/assembly loading
- Neutron and gamma dose rate figure of merit (FOM)
- Case run to confirm conclusions from ORNL data

Combined FOM/canister reduced 8%

- -9% gamma FOM/canister
- +25% neutron FOM/canister

Number of canisters loaded is reduced 16%

Small or beneficial dry storage dose rate changes due to HBU

HBU/SFP Scoping Calculation Conclusions

1) **Modest increase in SFP inventory over time**
   - Offsetting effects
   - Net result depends on multiple variables (canister DH limits, DH zoning, cycle length, etc.)
   - Equilibrium impact takes multiple cycles to build-in, provides time to accommodate

2) **Modest increase in SFP peak decay heat (~2%-5%)**
   - Net result depends on multiple variables (canister DH limits, DH zoning, cycle length, etc.)

3) **Number of dry storage canisters loaded will decrease 15-20%, reducing total dose**

4) **At normal conditions, individual dry storage system dose rates will also likely to decline**
   - Normal storage is gamma dominated
   - Additional cooling time needed for HBU reduces gamma dose rate

5) **Scoping calculations used simple methods and numerous simplifying assumptions**
   - Recommend plant/cycle/canister specific assessment for confirmation and transition planning

SFP inventory impacts appear modest, occur gradually over multiple cycles.
Normal condition dry storage dose rates likely to decline (per canister and total).

Questions/Comments?

Together…Shaping the Future of Energy®

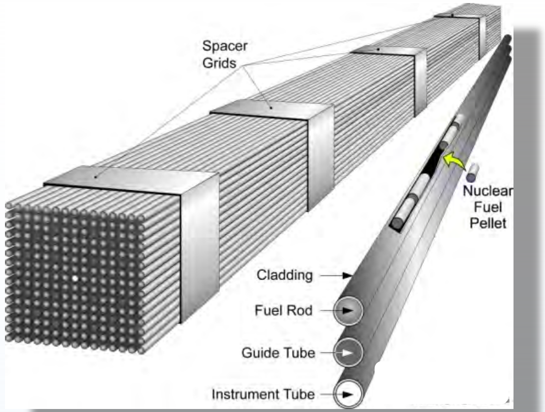
Extended Storage Collaboration Program (ESCP): Collaborative Forum for Addressing Global Technical Challenges Around Used Fuel

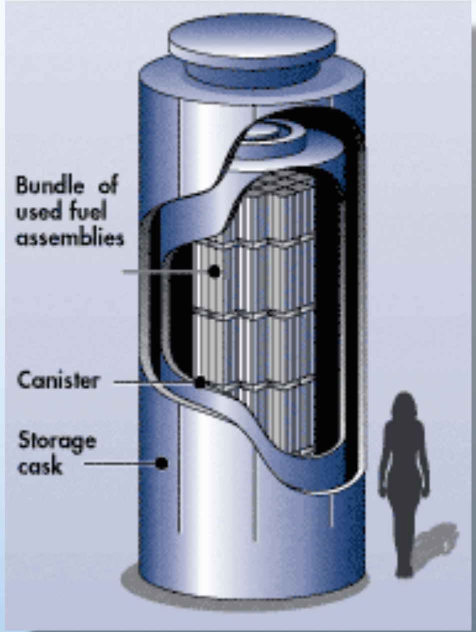
Hatice Akkurt
Used Fuel High Level Waste Management Program

ACRS Meeting

May 18, 2023

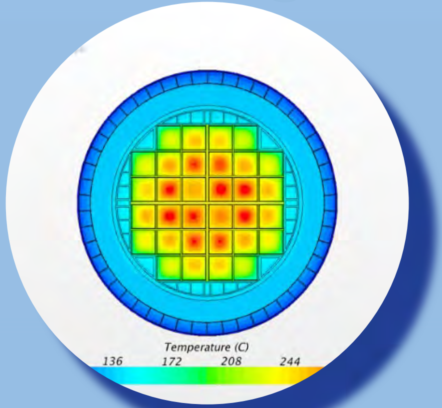
Managing Extended Storage of Used Fuel: Technical Challenges





Fuel Cladding Integrity

Aging Management of Dry Storage Systems



Accuracy of models: Thermal, decay heat, and dose models

Collaborative R&D to Inform and Transform

EPRI formed Extended Storage Collaboration Program (ESCP)

EPRI

Extended Storage Collaboration Program (ESCP)

Mission

- **Enhance the technical bases to ensure continued safe, long term used fuel storage and future transportability**

Goals

- Bring together US and International organizations engaged with active or planned R&D in used fuel area
- Share information
- Identify common goals and needs
- Identify potential areas of "formal" collaborations

Phases

- **Phase 1:** Review current technical bases and conduct gap analysis for storage systems
- **Phase 2:** Conduct experiments, field studies, and additional analyses to address gaps
- **Phase 3:** Long-term performance confirmation

ESCP Structure - Subcommittees (SCs) and Task Groups (TG)

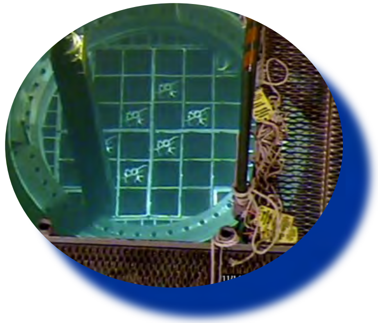
- **ESCP Steering Committee**
  Chair: Hatice Akkurt (EPRI)
  - **Fuel Assembly**
    Chair: Mike Billone (ANL)
    - Advanced Fuels
      (ATF, HBU, HE)
      Sven Bader (Orano)
    - Advanced Reactors
      TBD
  - **Modeling & Benchmarking**
    Chair: Maik Stuke (BGZ)
    - Thermal
      Lead: David Richmond (PNNL)
    - Decay Heat
      Lead: Fredrik Johansson (SKB)
    - Radiation Dose
      Lead: Kaushik Banerjee (PNNL)
  - **Canister Integrity/Aging Management**
    Chair: Rob Kelly (UVA)
    - Aging Mechanisms and Consequence
      Leads: Sam Durbin (SNL) and Jimmy Burns (UVA)
    - Mitigation, Repair, and Inspections
      Lead: John Kessler (JKA)
  - **International Subcommittee**
    Co-chairs: David Hambley (NNL), Woo-seok Choi (KAERI), Brady Hanson (PNNL)

Subcommittees and task groups are formed based on needs and sunset once they complete their objectives

EPRI

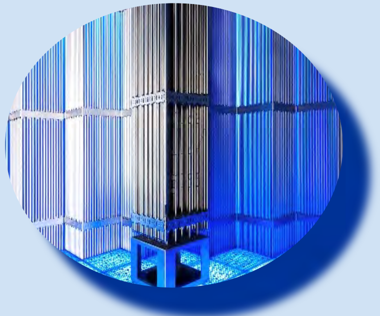
Spent Fuel Integrity R&D

[EPRI HBU Demo video on YouTube:]



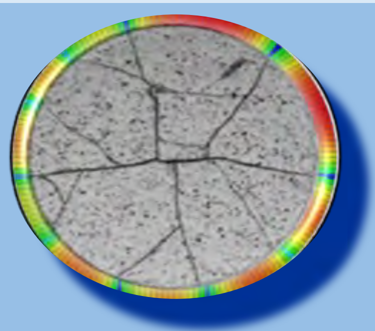
EPRI/DOE High Burnup (HBU) Demonstration Program

- ✓ Demonstrate high burnup fuel performance
- ✓ Supports dry storage license renewals



Improved Performance Margins

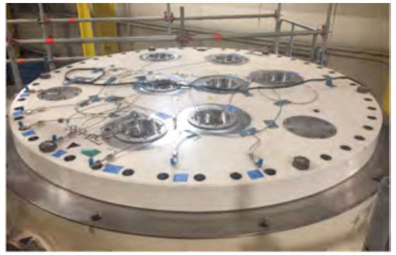
- ✓ Measured temperatures much lower than estimated
- ✓ Identified performance margins exist
  - ✓ Multiple PIRTs since HBU Demo loading

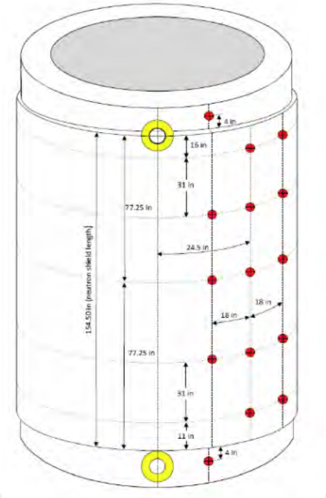


Key High Burnup Fuel R&D Findings

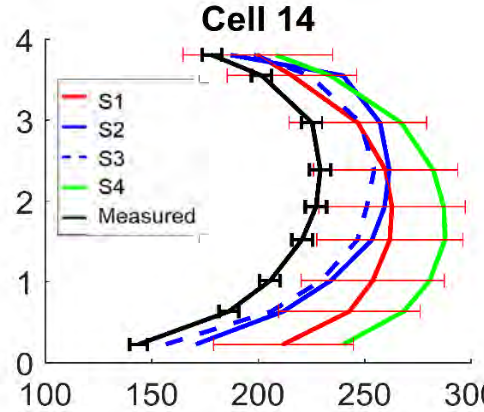
- ✓ High burnup fuel more robust than originally understood
- ✓ Dry storage and transportation are safe

HBU Demo showed measured temperatures are much lower

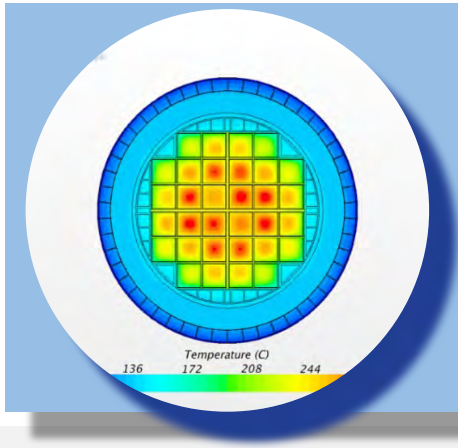




Parameter	FSAR	LAR	Best-Estimate	HBU Cask Meas.
PCT	348°C	318°C	254-288°C	229°C
Total Heat Load	36.96 kW	32.934 kW	30.456 kW	30.456 kW
Ambient Temperature	100°F	93.5°F	75°F	75°F
Design Specifics	Gaps	Gaps	Gaps	No Gaps?



Modeler	Code
S1	ANSYS Fluent
S2	STAR-CCM+
S3	COBRA
S4	ANSYS APDL

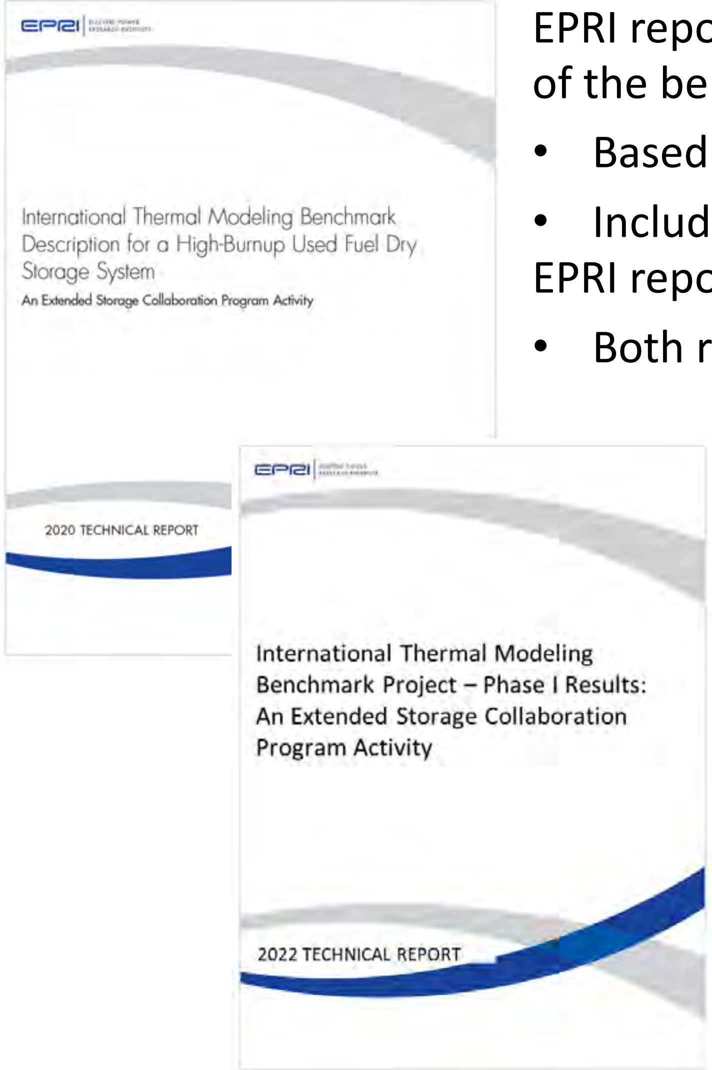


- **HBU Demo Measurement results published in EPRI report 3002015076**
- **HBU Demo Blind Benchmarking Thermal Results published in EPRI report 3002013124**
- **Both reports are publicly available**

EPRI

ESCP Modeling & Benchmarking Activities

ESCP International Thermal Modeling Project



EPRI report, **3002018498**, provides a description of the benchmark:

- Based on publicly available information
- Includes a recording of the description

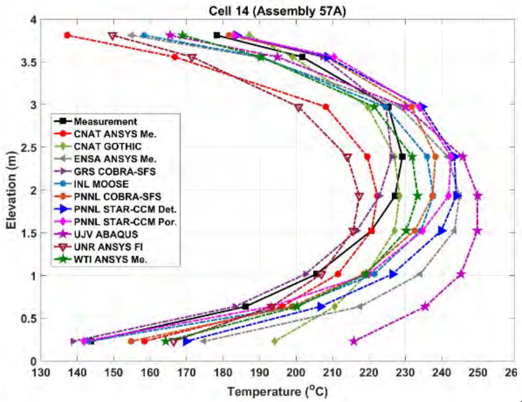
EPRI report, **3002023976,** provides Phase I results

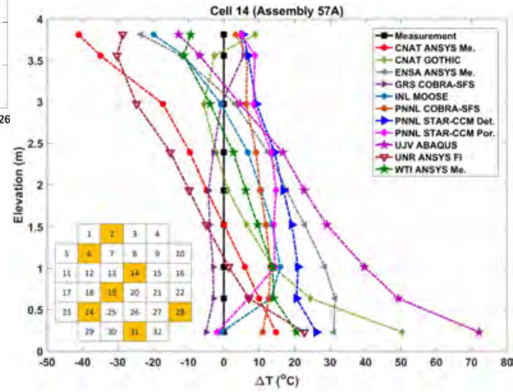
- Both reports are publicly available

Solution Method	Code(s)	Organization(s)
Finite Element Method (FEM)	ANSYS Mechanical	CNAT ENSA WTI
	ABAQUS	UJV
Finite Volume Method (FVM)	ANSYS Fluent	UNR
	GOTHIC	CNAT
	STAR-CCM	PNNL
Finite Difference Method (FDM)	COBRA-SFS	GRS PNNL
FEM & FVM	MOOSE	INL

Observations:

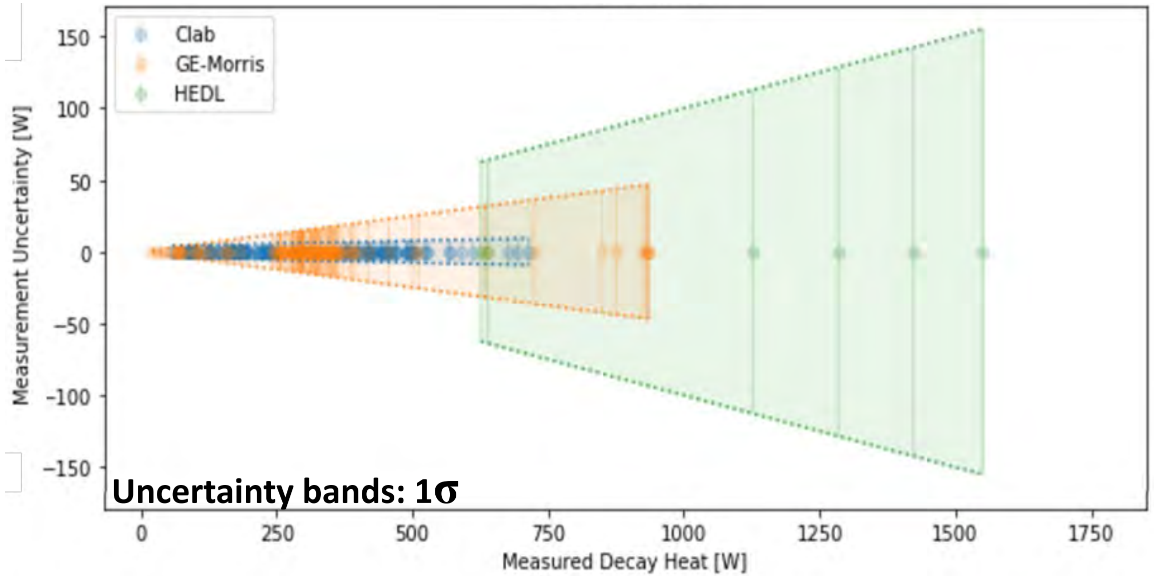
- Wide variation in temperature predictions
  - Between different codes
  - Between different organizations, using the same code
- No correlation between computational time, details of the model and accuracy of the results

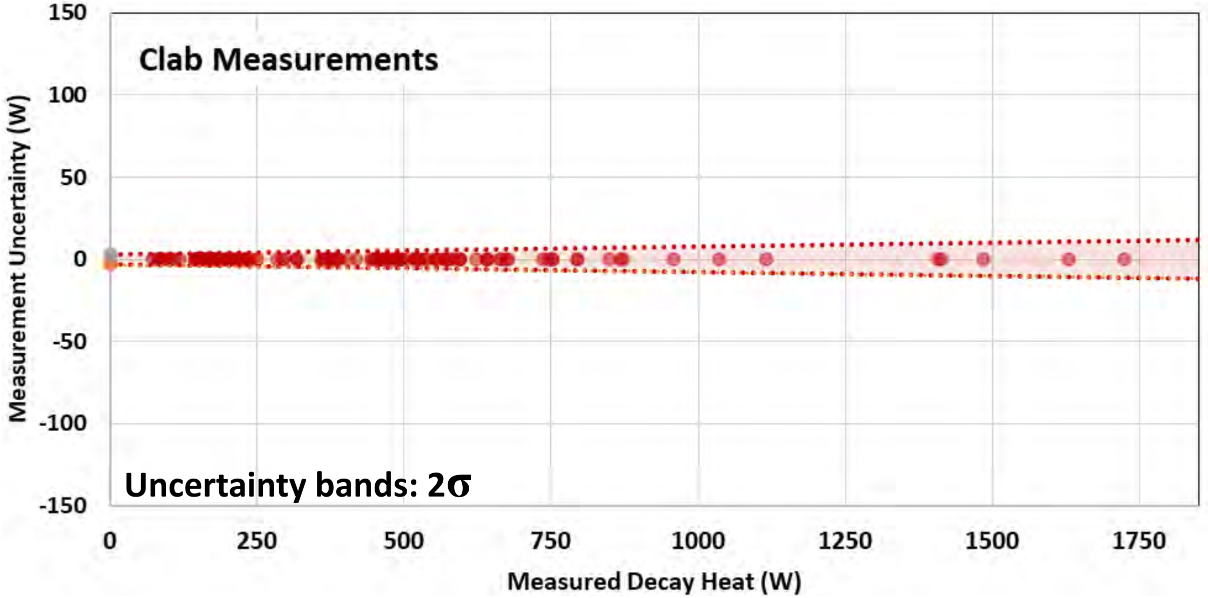




Eight organizations from four countries using seven codes and 11 solutions with different solution approaches. Phase I is complete; Phase II is ongoing

Extending Validation Range for Decay Heat and Reducing Measurement Uncertainty





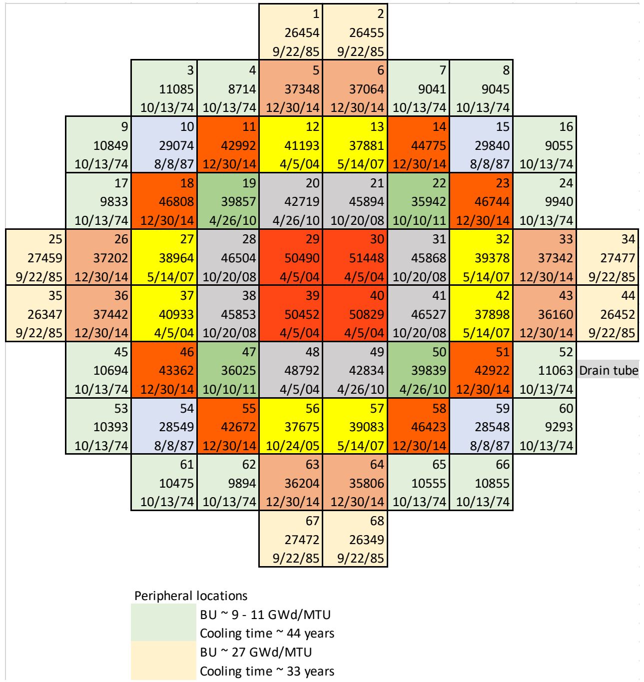
- **HEDL:** Large measurements uncertainty; no other measurements for high DH range ➔ can't be taken out of validation set yet
- **GE-Morris:** Measurement quality issues at higher DH; no other measurements ➔ can't be taken out of validation set yet
- **CLAB:** Low measurement uncertainty; focus on low DH

- Over 120 new DH measurements that are not published yet
- High quality data ➔ better validation set ➔ decrease DH uncertainty and increase margins for global industry

EPRI initiated a collaborative project with SKB to publish unpublished CLAB measurements; perform new measurements to close the gaps - **ESCP Decay Heat Task group members, and other interested collaborators, will perform review and participate in potential blind benchmark for new measurements**

Radiation Dose Benchmarking

Radiation dose measurements from three loaded canisters are available from two sites for modeling



Benchmark description, based on publicly available documents, and assumptions will be provided to participants

Blind Benchmark

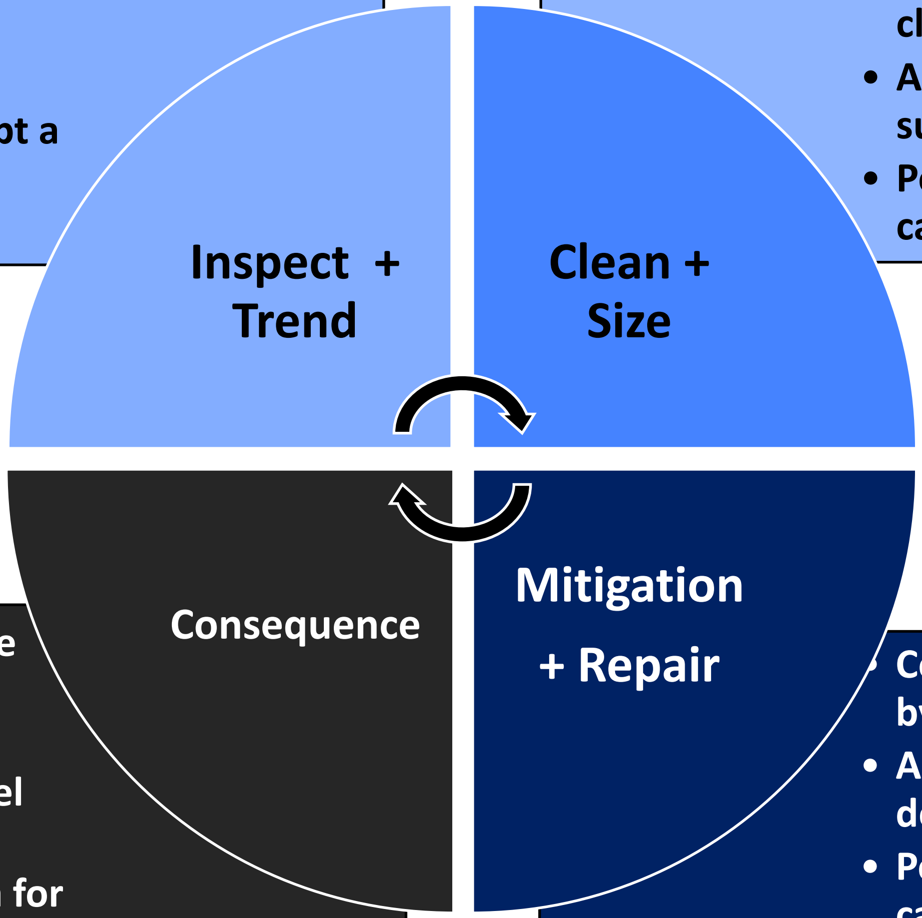
- EPRI will not release the measurement data until the completion of benchmark project
- Actively participating organizations:
  - USA: INL, ORNL, PNNL
  - Sweden: SKB
  - Japan: NMRI
  - Germany: GNS
  - Spain: ENSA
- Project kick-off meeting in February 2023
- Results will be published in a publicly available EPRI ESCP report

ESCP Aging Management and Canister Integrity

Canister Aging Management Research Activities

- Many EPRI/Industry development and demonstration projects
- Robotic visual inspection
- Inspection results could prompt a need to clean, assess, and mitigate/repair

- Active project to demonstrate cleaning
- Active project for flaw sizing NDE support
- Potential use of EPRI/DOE canisters



- Potential SCC dose consequence informs mitigation and repair
- Active projects for GOTHIC/MELCOR canister model internal particle deposition
- Need actual measurement data for validation

- Cold spray and coating evaluations by multiple organizations
- Active project for repair demonstration
- Potential use of EPRI/DOE canisters

Many collaborative research activities to address current and potential future needs

ESCP Focus Areas - Next 2-3 Years

Forward Looking ESCP Focus Areas for Next 2-3 Years

Fuel

- Phase II sister rod testing
- Transport of HBU Demo cask and opening
- Increased focus on ATF/HE/HBU and back-end effects
- Increased focus on Advanced Reactors and back-end issues

Aging Management and Canister Integrity

- Mitigation and repair techniques development
- Demonstration via field tests
- Acceleration of consequence studies

Modeling & Benchmarking

Thermal:

- Completion of international thermal modeling project
- Gathering more benchmark data during inspections

Dose:

- Blind benchmarking activity for dose modeling

Decay Heat:

- Completion of decay heat reports
- New measurements and potential for blind benchmark

Collaborative R&D to Inform and Transform

Summary

- ✓ ESCP is a forum that enables collaborative development of innovative solutions for spent fuel management
- ✓ Recent cooperative **R&D with DOE and NRC reduced dry storage and transportation concerns** of high burnup fuel
  - ✓ Research shows continued long-term storage of commercial spent fuel is **safe with larger performance margins**
- ✓ ESCP is continuing to enable the development of **improved aging management guidelines** with inspection, repair, and mitigation technologies as well as consequence analysis
- ✓ ESCP is increasing its activities in modeling and benchmarking, advanced fuels (ATF/HE/HBU), and advanced reactors areas.

ESCP Winter 2023 Meeting

October 23-26, 2023; Charlotte, NC


Together...Shaping the Future of Energy™