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(202) 234-4433

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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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RADIATION PROTECTION AND NUCLEAR MATERIALS

SUBCOMMITTEE

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THURSDAY, NOVEMBER 16, 2023

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The Subcommittee met via Teleconference, at 1:00 p.m. EST, David A. Petti, Chair, presiding.

COMMITTEE MEMBERS:

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JOY L. REMPE, Member

THOMAS ROBERTS, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANT:

STEPHEN SCHULTZ

DESIGNATED FEDERAL OFFICIAL:

WEIDONG WANG

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

1:01 p.m.

CHAIR PETTI: Good afternoon, the meeting will now come to order. This is a meeting of the Radiation Protection and Nuclear Materials Subcommittee of the Advisory Committee on Reactor Safeguards.

I'm Dave Petti, chairman of the subcommittee. Members in attendance are Charles Brown, Joy Rempe, Matt Sunseri, Ron Ballinger, Walt Kirchner, Vesna Dimitrijevic, Vicki Bier, Greg Halnon, Tom Roberts, Bob Martin, and I believe Jose March-Leuba may be on.

MEMBER MARCH-LEUBA: I am on.

CHAIR PETTI: Great. And Steve Schultz, our consultant, is also with us today. Weidong Wang is the Designated Federal Official for this meeting.

As posted in the agenda and on the ACRS website, the topic for today is to hear information -- an information briefing on Sandia National Laboratory's report, high burnup fuel source term accident sequence analysis.

The subcommittee will hear presentations by and hold discussions with the NRC staff, Sandia, and other interested persons regarding this matter.

The meeting is open to the public. Rules for participation in all ACRS meetings, including today's, were announced in the Federal Register on June 13, 2019.

The ACRS section of the U.S. NRC public website provides our charter, bylaws, agendas, letter reports, and full transcripts of all full and subcommittee meetings, including slides presented there. The meeting notice and agenda for this meeting were posted to there.

We've received no written statements or requests to make an oral statement from the public.

The subcommittee will gather information, analyze all of the issues and facts, and formulate a post positions and actions as appropriate today.

Transcript of the meeting is being kept and will be made available. Today's meeting is being held in person and over Microsoft Teams through ACRS staff and members, NRC staff, and other attendees.

There's also a telephone bridge line and a Microsoft Teams link allowing participation for the public.

When addressing the subcommittee, participants should first identify themselves and speak with sufficient clarity and volume so they may be readily heard. When not speaking, we request that participants mute their computers or microphone by pressing star six.

We will now proceed with the meeting. And before we call up our staff management, I'd like to just put some context in. As you know, we reviewed Reg Guide 1.183 here about a month ago. And questions, as part of that discussion, talked about the calculations that were done by Sandia.

And we thought it would be useful to have a briefing on this so members get a more complete picture of the depth of the technical basis upon which 1.183 relied.

With that, whoever's going to -- you are? Okay, great.

MS. WEBBER: All right, good afternoon, everybody. My name is Kim Webber. I'm the Director of the Division of Systems Analysis in the NRC's Office of Nuclear Regulatory Research.

It's a pleasure to be here today to talk about a topic that is of really broad interest, not only for the NRC, but also for our external stakeholders.

There's a lot of interest as it relates to Reg Guide 1.183 and, you know, a future update to that regulatory guide. So, I'm glad to see that there are a lot of online participants as well, as those who are in the room.

And this is going to be a focus on research that was recently done over the last few years. And it's really not focused on the regulatory application of it, although it's probably a key component of the technical basis that may be used to develop a Reg Guide, a future revision of the Reg Guide.

So, we do also have our regulatory partners here with us, Elijah Dickson and Michelle Hart. And if there are questions related to the regulatory aspects, hopefully they can answer those types of questions.

But also here with me today, I have my staff, Hossein Esmaili, who's the chief of the fuel source terms code branch, along with Shawn Campbell and Mike Salay who are experts in severe accidents.

And then, we also have our colleagues from Sandia National Lab at the table, too, who are also going to do some presentations.

But before we get into that, you know, I just wanted to say that, you know, this analysis that the staff and contractors have been working on for quite some time is really important when it comes to providing technical basis for the use of high burnup fuel and also accident tolerant fuel in the future.

And as many of you know, the MELCOR codes, severe accident code, was used to perform these analyses. And it's a system level code that simulates the entire spectrum of accidents and phenomena from accident initiation to core and fuel degradation and fission product gas release from the fuel and transportation to containment and the environment.

It has a large user base, both domestically and internationally, with about 30 participants or 30 countries participating in our CSARP, Cooperative Severe Accident Research Program, co-chairing program.

And you know, it's critical to have their participation because they identify code bugs. They highlight important aspects of the scenarios that are included in those codes. They contribute, you know, their own studies with using those codes. So, that cooperation is critical.

And then, also, MELCOR uses inputs from our SCALE neutronics code for decay heat and fission product inventories.

And so, because of the flexibilities of

the combination of MELCOR and SCALE, we also use them quite extensively for analysis that will support the non-light water reactor technical bases and confirmatory analysis going forward.

So, over the years, we've used MELCOR for a number of regulatory applications, as many of you know.

Some of the high visibility projects include the state of the art reactor consequence analysis, or SOARCA study, and post-Fukushima analysis such as the containment protection and release reduction documented in a NUREG -- in one of the NUREGs.

And we've also completed earlier MELCOR analysis which formed the technical basis for the Revision 1 to Reg Guide 1.183 which is called the alternative radiologic source term for evaluating design basis accidents at nuclear power reactors.

So, this research benefits not only away from code physical model improvements, but also improvements in best practices in generating code input decks which are publically available representations of plants and accident scenarios.

So, in 2020, we convened a panel of international experts to create the phenomena identification ranking table to address the significant phenomenologic issues impacting core degradation and radiological releases for various accident tolerant and high burnup fuels.

And we compared those phenomena against the traditional large light water reactor fuels.

The aim of the PIRT was to help NRC understand how the ATF concepts and the high burnup fuel may change core degradation and radiological release behavior which provide information that is useful in developing source terms for these designs.

And also, along with that PIRT, we did publish a literature view that provided input to the PIRT panelists and also the results of the panel findings.

And both of those are documented in NUREGs.

So, now, I'd like to turn the presentation over to the staff and our colleagues from Sandia National Lab who, Dave Luxat and Lucas Albright, and they'll present the analysis that was associated with the high burnup source term report and the peer review.

And I think we mentioned that this will be used as the, you know, probably part of the technical basis for the Reg Guide 1.183 Revision 2.

So, let me turn it over to Shawn, I think you're next.

So, thank you for your attention.

MR. CAMPBELL: Yes, thank you very much.

So, hello, everyone. My name is Shawn Campbell. I'm in the Office of Research in the Sandia Branch and with Kim Webber.

And so, I'm actually going to turn it over to Lucas Albright here in just a moment and our colleagues at Sandia National Labs.

They are the ones that we commissioned to do this work and put together this Sandia report.

So, we've asked them to come in and help to explain the work that they've done. And we relish your questions and feedback as we go.

So, Lucas?

MR. ALBRIGHT: Thanks, Shawn.

So, I think we can pull up the slides if those are ready. Is that something we have access to here? Okay.

MEMBER REMPE: Sorry, I've been tied up with other things, so Dave's in charge.

But yes, we thought, unless we're told, we rely on the presenters to pull up their slides.

Weidong, can you do it for them? Is that what you'd prefer?

MR. ALBRIGHT: Yes, I think -- yes.

(Off-microphone comments.)

MEMBER REMPE: Weidong, can you -- yes, can you share your screen? We can do that.

MR. ALBRIGHT: No, sorry about that, that was our misunderstanding. I apologize.

(Off-microphone comments.)

MR. ALBRIGHT: Yes, no problem. If they're not readily available, we can pull them up and --

MEMBER REMPE: He just needs to share his screen and they're coming up now.

MR. ALBRIGHT: All right, thank you all.

Okay, so, my name is Lucas Albright. I work at Sandia National Laboratories in the severe accident analysis and modeling group performing analyses with the MELCOR code and also developing the MELCOR code.

This presentation that I'll be giving today is sort of an overview of the technical details of the high burnup fuel accident source terms that we developed.

This was a multi-year effort to basically extend the NUREG-1465 source terms to higher burnups.

What you'll notice in this presentation is that I'm sticking to an explanation of what we did, how we did it, and why we did it that way, not necessarily, you know, implementation or how these numbers would be used on the regulatory side.

Next slide, please? Thank you.

So, a brief overview of some contents that we have in this presentation.

I'll go into the motivation and background that fed into this work.

Then, I'll give a high level overview of the key takeaways from the work before diving into the technical details during the deep dive.

Then, we'll have a little bit of a summary of what we went over for the high burnup source terms before going into the independent peer review and the upcoming work.

All right, next slide, please? So, the high burnup fuel source term analysis, like I said, this was a multi-year effort published in 2023.

The objective here was to develop alternative source terms that were applicable to higher burnup light water reactor cores with extended enrichment, or HALEU, fuel.

And this, like I said, is extending that NUREG-1465 source term and is in the SAND2011 source terms to this higher level of burnup and extended enrichment.

Next slide, please? Some historically relevant studies that we wanted to just sort of just give a high level overview of were TID-14844, the calculation of distance factors for power and test reactors.

This was published back in 1962 and focused really on some experimental data. This was sort of prior to the introduction of the use of computer codes to inform our source terms.

The next major study was NUREG-1465, the accident source terms for light water nuclear power plants.

This was the first source term to use computer codes. This one used the STCP code, which was the forefather of the MELCOR code.

The next major relevant study that we come across is the SAND2011 study.

This was the first source term study to use the MELCOR code.

And this study actually looked at accident source terms for, again, light water reactors, but using high burnup or MOX fuels.

This was an older version of MELCOR published in 2011. So, many advancements have occurred since that study was published.

Next slide, please? So, this is a little time line that I put together for today's talk just sort of going over the major developments as we march through time from NUREG-1465 to today's SAND2023 report.

What you'll see are that between the NUREG-1465 and SAND 2011 reports, we had a number of major developments, including the MELCOR code becoming sort of -- or coming online.

NUREG-1560, this was the plant examinations that sort of gave us the description of scenarios that we look at in SAND2011 and SAND2023.

Reg Guide 1.183, which we're all familiar with.

Phebus FP occurred in this time frame as well which gave us some insights into severe accident progression that we didn't necessarily have as clear of an understanding of it at the time.

In the time since 2011, we actually see that a lot more work has been done.

We note first, Fukushima Daiichi happened a couple months after SAND2011 was published.

The SOARCA studies, for those of you who are unfamiliar, these are the state of the art reactor consequence analyses.

This is where we took a given plant, we have two volumes for the original body of work which was Surry and Peach Bottom, where we basically ran a set of sequences and sort of advanced the state of practice using the MELCOR code to model these severe accidents.

During this time, the BSAF project, of course, began.

This is where we demonstrated, excuse me, where we demonstrated the MELCOR code in modeling the Fukushima accidents.

A number of improvements were also made to the code during this period of time to incorporate those findings.

The last two that I want to make clear here, but before the SAND2023 report were produced, were the SOARCA UAs which were, essentially, extensions of the original two SOARCA documents to look at uncertainty analysis where they investigated the parametric uncertainties associated with severe accidents to sort of explore that uncertainty space that we see.

The other major point here is the HBU, HALEU, and ATF severe accident PIRT that was mentioned earlier.

This body of work was very important in terms of influencing both this work and the ATF source terms which are ongoing right now.

Next slide, please? Okay, severe accident modeling advancements. So, I mentioned that a number of advancements have occurred in the time since SAND2011 as well as the prior studies.

Two major modeling advancements that we want to highlight today are the heterogeneous integrated reactor core modeling.

What mean is that we have discretized the core.

So, what this does in discretizing the core to multiple nodes instead of a single node, we actually promote a progressive and extended core degradation period.

And this actually sort of has shifted the way we look at accidents. We no longer a distinct gap release phase because we've got different stages of degradation in the various nodes.

We have prolonged core damage progression because we have more efficient transfer of heat out of the core as its degrading.

And this all translates to longer times for lower head failure.

The next key point in terms of modeling advancements that I want to touch on today is the prevalence of low pressure scenarios.

So, basically, this was a finding of the SOARCA analysis that, during the early in-vessel phase, we will reach a larger proportion of low pressure scenarios through either thermally induced safety release valve seizure of hot leg creep rupture for pressurized water reactors.

So, this is, basically, both the BWRs and PWRs are basically more likely to go through a low pressure scenario then reach lower head failure before the depressurize.

MEMBER REMPE: I have a couple of questions.

MR. ALBRIGHT: Yes?

MEMBER REMPE: Don't you think that there may be some additional insights that are needed -- or expected to come as we go -- we learn more from Fukushima and other tests?

For example, maybe we should learn a

little more about vessel failure as we look at the fuel assemblies and control rods that are ex-vessel and the photos coming from Fukushima?

And then, the SRVs, the SRV failure, I don't think that's been accommodated yet.

And so, maybe you might want to make some comments about that, even though you improved MELCOR, there might be more that's coming.

MR. ALBRIGHT: Yes, yes, I think those are very good comments, very good points.

I definitely think that, you know, we're on the march towards progress here. And there's always improvements to be made and sort of further refinements.

And I think the way I would contextualize this body of work was that it was performed according to the current state of practice.

And as that data becomes more available, we absolutely would be interested in taking a look at those and incorporating them into these types of analyses. Thank you.

Next slide? So, just a quick overview for the impact of early depressurization on these two different reactor types.

We have two sort of concept nodalizations of a BWR Mark I containment structure on the left, with the reactor vessel in the center there.

And then, on the right, we have a pressurized water reactor and a large dry containment.

What I want to draw your eyes to are the red Xs in each of these cases.

This is where the early loss of the pressure boundary occurs in each of these reactor types.

And basically, the point here is that, once we open up this flow path, once this break happens, or in the case of the valve seizure occurs, this is a direct release pathway for radionuclides to transfer directly into containment during early in-vessel degradation.

So, this is a key point for why we're seeing the numbers we are today.

Next slide, please? Thank you. So, the next point we want to talk about here are the severe accident data sets that have sort of developed.

In recent years, we have sort of highlighted a few of the data sets here. These -- this is not a comprehensive list.

But the idea is that, as time has gone on, particularly since NUREG-1465, these severe accident data sets from different experiments or different events have given us a clearer idea and a clearer picture of core degradation or core damage progression and the radionuclide releases that occur as a result.

The ones we call out here for the experiments are both Phebus and VERCORS.

These two experiments, both sort of demonstrated early fuel failure.

What I mean by early fuel failure is failure at lower temperatures than the sort of constituent materials would suggest.

And then, the Phebus experiment is where we saw the hypothesized cesium molybdate being the dominant chemical form of cesium.

And then, for cores, we saw -- we actually used that MELCOR as a validation basis for a high burnup fission product release rates model.

The next data set here is the Fukushima Daiichi accident.

And right now, as Joy said, this is ongoing work. We're still learning things about this accident as they sort of continue with the decommissioning process.

But one of the key findings that actually came from one of our peer reviewers with their innate knowledge or intimate knowledge of what's going on over there, was that they're actually seeing confirmation that cesium molybdate is the dominant chemical form of cesium.

So, next slide, please?

DR. SCHULTZ: Lucas, the other elements that Joy mentioned with regard to Fukushima, they were not brought forward in the PIRT evaluation that led to the work that you've done?

MR. ALBRIGHT: No, the PIRT work that sort of fed into this work were phenomenological differences between high burnup fuels and conventional fuels.

So, it wasn't necessarily looking at plant wide behavior so much as the fuel behaviors.

DR. SCHULTZ: Thank you.

MR. ALBRIGHT: Thank you.

So, the next background slide here is severe accident knowledge advancements.

These are some of the high level key insights over the past decade of development, two decades, really.

And the first one we see here is the chemical form of iodine.

The treatment of iodine in our severe

accident codes is different than back in 1995 when we were using STCP to develop the NUREG-1465 study.

They had assumed 95 percent of iodine in the form of cesium iodide.

We, today, according to the current best practice -- published best practices in MELCOR, assume all iodine is bound in cesium iodide.

We still assume that 5 percent of that iodine inventory is present in the gap which is consistent between the two studies.

The next point is the chemical form of cesium.

This is also a new practice relative to NUREG-1465. In NUREG-1465, the assumption was predominant presence of volatile cesium hydroxide.

Today's practice is to assume, essentially, 5 percent of the total cesium inventory is in the gap. And that's made up of both cesium iodide and cesium hydroxide based on the available mass inventory for cesium iodide.

The next change since NUREG-1465 is the predominance of cesium molybdate, which I talked about in the last slide being the dominant chemical species and being confirmed in the current Fukushima efforts.

The next slide -- the next point here,

excuse me, are the molybdenum releases.

We found that, since NUREG-1465, especially in the presence of the cesium molybdate, that molybdenum releases are much larger than the other metallic products such as ruthenium and palladium.

And this is sort of reflected in the new way that we break out our chemical classes which we'll see later in this presentation.

Yes?

MEMBER REMPE: Since you pointed out the peer review, which, by the way, I know you're going to talk about it later, and I thought it was admirable that you not only got their comments and addressed it, you went back to them to see how they did it.

But DDR and Louis, the comments they made, it seems like that they mentioned some of the more recent data from OECD projects than you have said.

It was state of the practice. We did what we could.

Is there a path forward that's clear that NRC will be updating the code MELCOR to take into consideration some of the changes that they've recommended?

MR. ALBRIGHT: I think that's something

that Shawn can answer.

MR. CAMPBELL: Yes, the answer is yes, always.

I mean, we're actively involved in all of these international programs. Right? And that's always our purpose in being involved in these programs is try to wait until they are mature. They've been -- and the testing is complete.

And then, yes, to try and go on and incorporate that.

So, we're always seeking to make MELCOR align with this best practice, right, and to align with what's coming out of the research programs.

So, yes, for sure.

MEMBER REMPE: Then, is going to lead to another question. I was going to wait until later.

But after Fukushima happened, there was the benchmark, right, between MAAP and MELCOR.

And if you had to do that, even with the changes you've made now, plus the future changes, would you come up to some point where things were still progressing along the same path or are you going to have some divergence?

And, again, hopefully, we don't have another accident and we're trying to figure out what's going on, but if it were, you would understand, well, okay, MAAP hasn't incorporated this data or they did incorporate something as we understand the differences?

MR. CAMPBELL: I'd say that's a research project in and of itself.

But yes, I mean, we are -- especially, meaning Dave, has a wonderful experience in MAAP as well and understands the ins and outs of that code.

And so, yes, we do actively try to understand what's involved in MELCOR on that.

So, are you saying that, would we be able to understand the differences between the two codes? Is --

MEMBER REMPE: If you had to do something again, like the benchmark where you had to do a comparison between MAAP and MELCOR, and understand why there's some differences, which differences would you expect and to be able to say, okay, yes, that's because such and such a model was or wasn't incorporated in our two different codes?

Because I know MAAP's going at a different rate on how they incorporate some Fukushima insights.

And I'm just wondering how that that comparison would go if you had to do it?

Which is a bit off the topic here, I know, but I'm interested and I'm curious.

MR. ESMAILI: So, Dave, do you want to -- sorry, Hossein Esmaili.

So, Dave, you've been involved in that benchmarking of MELCOR versus MAAP.

And so, what do you think was the upshot of that and where do we go from there?

MR. LUXAT: So, a fair amount of that.

There are some differences that emerge in terms of how the codes treat in-vessel degradation, particularly in core degradation.

And utilize, particularly for the BWR, paths for relocation downward.

So, MELCOR tends to have a propensity to use the bypass more effectively to allow to relocate down.

MAAP, at times, has tended to form crusts that can hold up debris above it, so to speak, and promotes more of a crucible like geometry.

I can't speak to more recent updates necessarily, but some of the indications from Fukushima did tend to highlight the potential for a, shall we call it, a more incoherent degradation that is the downward relocation of debris that we typically see in MELCOR and a more segmented relocation of debris in to the lower plenum as a result.

And I think we have generally been moving down the direction of making better use across all the codes of a bypass to promote downward relocation, and if you will, a more incoherent release of debris into lower plenum for particularly BWRs.

The challenge is still one where, how do I put this, we still -- when it comes to a PWR, when it comes to events with say less water addition where that the strength of those crusts are not as clear, there are still uncertainties we just don't have the reactor scale data to understand if these sort of lower crusts will form, be strong enough to hold up a crucible like geometry like we saw at TMI-2.

If you recall, the TMI-2 was a very, very different type of event in the perspective of water injection over to Fukushima.

But generally, we -- with both codes, we've been moving in the direction, to wrap this up, of, you know, understanding and promoting the idea of downward relocation towards the core plate as a potentially more dominant relocation mechanism.

MEMBER REMPE: So, where I'm going with what I'm asking is, because of, again, there's international comparisons which MELCOR is involved with as well as other countries codes, is that, as you improve the code which has been recommended, and you're saying your going to keep doing it, keep in mind the other codes in industry.

Because it helped that we kind of keep, as well as we can understand about in-vessel early relocation.

And later on, there's a lot of uncertainties.

And it just seems like different models are being put in and it just doesn't mean that you can influence what industry's doing, but to just kind of keep track is what I was kind of going with the question.

MR. LUXAT: There's overall a general sense that when it comes to source terms, there are going to be difference, obviously, between the codes.

But a lot of the dominant releases occur for both codes during the same period of time where you've got largely, if you will, debris that has surface area through which fission products can be released.

It's generally later in the in-vessel accident progression where you start to either build up relative to realized debris or relatively, shall we say, multi-cool like debris.

But I would say from at least to the perspective of the early in-vessel phase, when it comes to source terms, the codes tend to come close to each other, generally.

And when we look at the Fukushima data via MAAP or MELCOR, the general releases that both codes are predicting or estimating for Fukushima are generally consistent overall.

MEMBER REMPE: The chemistry doesn't change, but the iodine is in all of those things?

MR. LUXAT: Absolutely.

MEMBER REMPE: Okay.

MR. LUXAT: Absolutely.

MEMBER REMPE: Thank you.

MEMBER ROBERTS: Just a question of the detail.

Just looking at your slides with knowledge investment and then the statements that have the word assumes, they all reflect, say, a chemical reaction.

It is assumed, in this case, right, there is not, in other models, that are incorporated in these reactions and reaction rates and energy ins and outs?

MR. LUXAT: The speciation in particular, yes, that is an input, shall we say, that is assumed. It's not based on, shall we say, the execution during the simulation, but we'll talk about the next, say, a thermochemical equilibrium calculation.

The speciation is, if you will, to use some of our lingo, assumed as frozen catastrivia (phonetic), if you will.

MEMBER MARTIN: And you don't feel like, you know, neglecting that has this negative impact on progression in one of the other?

MR. LUXAT: For the conditions that we see, generally, we're in a steam environment, so to speak. Obviously, we would make different considerations if this was an accident with a spent fuel pool with a different atmosphere reducing versus oxidizing conditions, so to speak.

And that will influence it. But for this, and this was sort of a point of discussion that we had with the peer review, the frozen chemistry or the phase of the accident in a steam rich, if you will, reactor vessel, during the phase of release is a reasonable approximation, particularly when it comes to the cesium and iodine releases.

MEMBER ROBERTS: I'm wondering if you

could speak to the assumption of a 100 percent of the iodine and cesium iodide?

It seems like ignoring the gaseous iodine could be significantly nonconservative, depending on the analysis.

And I was just wondering, you know, why that assumption is justified, given that at least the report says it's highly uncertain as to what the behavior of gaseous iodine is?

MR. ALBRIGHT: Yes, that's a good point.

So, the SOARCA uncertainty analyses have actually looked at the impact of some of these speciation uncertainties.

And we sort of point to those reports as being like the place to find that information and being outside of the scope because it's really investigating what kinds of practices one could perform rather than the current state of practice, if that makes sense.

In those analyses, they actually had a, essentially, the iodine -- elemental iodine mass was a function of burnup.

And this was informed by experiments.

And if I remember correctly, the iodine -- the percent of the iodine mass that ended up in this elemental form was single digit percents, so, one, two, three.

And these percentages are very low and well within the uncertainties that are actually being released to containment based on the study -- this current study.

So, while the uncertainty is there, I think we actually have, in this analysis, covered it with our uncertainty bands on these results, if that makes sense.

MEMBER ROBERTS: I don't suppose it would depend on how much credit you're getting in containment.

MR. ALBRIGHT: So, the analysis that we're presenting today actually doesn't account for scrubbing.

We report the total radionuclide inventory reaching containment in the tables that we'll be talking about today.

So, the scrubbing is something that's considered traditionally in downstream codes, if at all. And that's something that, actually, in our follow up presentation we'll be talking about it in much more detail.

MEMBER ROBERTS: Okay, thank you.

So, it might be more of a question for Elijah, then.

It seems like once you start to look at containment, decontamination phenomena, it makes a big difference to, in my past experience, of what the form is as how much credit you take for decontamination.

MR. DICKSON: Yes, that's right.

And we do the transport portion of these analyses utilizing that source, we're taking those credits based off of chemical speciation in the regulatory guidance space.

So, you would see those models in effect in Appendix A of Reg Guide 1.183.

MEMBER ROBERTS: All right.

So, it seems like a couple percent, you know, gaseous iodine can make a big difference once you come into containment response phase.

So, if you get into that later, that's great, but I was just wondering how that played out because I'd have to have .15 percent that's currently in the Reg Guide, I've seen that dominate, depending on what the analysis is because it's not scrubbed.

MR. ALBRIGHT: I think the last point, I don't think we've covered this one yet, are the tellurium releases.

Based on current practices, we see much more extensive tellurium releases during the early in-vessel phase.

This was a finding from the Phebus experiments that basically we've got a fission transportation of the tellurium because it's not being bound up with the zirconium as was previously assumed.

Next slide, please?

CHAIR PETTI: Do they know why?

MR. ALBRIGHT: I think this is an area of investigation still, is my understanding.

CHAIR PETTI: Because, in the day, when we measured it, we were pretty convinced there was tellurium in the cloud. And that's something that doesn't move around that easily. And yet --

MR. SALAY: This is Mike Salay. Yes, the -- in both of our cores in Phebus, I think they both observed that once the cloud was oxidized, the --

CHAIR PETTI: Okay, so these were -- yes.

We predicted that if you got to full oxidation, these are highly oxidic melts as opposed to more metallic melts which were, probably some of the earlier testing done in the U.S. was much more.

Yes, you relocated a lot of metal because you've got a liquefaction which is, you know, taking the highly metallic.

MR. SALAY: Yes, they tried to stop the experiment at the Phebus FP, but that's practically like on the level of a PWR when they start getting significant melting.

So, they were getting the tellurium releases before and then, the core ones are just the fuel pellets in the furnace.

And also, they stop before it melted.

MEMBER REMPE: So, Mike, move your mic closer to you.

People like me, I have to shout across the room, but it'll help.

MR. ALBRIGHT: Next slide, please? Thank you.

This is a quick overview of the findings of that HBU, HALEU, ATF, PIRT.

So, this was an investigation into the severe accident behavior for these different fuel types.

And the findings -- major findings from this report were that there were no significant differences between HUB and HBU HALEU fuels. So, those are going to perform more or less similarly under severe accident conditions.

There are some different thermophysical properties that could be expects, thermal conductivity of the fuel being one of them.

Fuel fragmentation and centering is expected to impact core degradation and the sort of rate at which that occurs.

The next point here being that the fission part of chemistry may change followed by a possible cladding embrittlement.

And then, -- and the cladding embrittlement, just for some context here, was more related to the impacts of a reflooding scenario than necessarily leading to different fuel failure behaviors.

And the last point here being the potential for recriticality if there were to reflood without unborated water.

So, that sort of covers the high level findings of that HBU, HALEU, PIRT.

Some of these findings made them -- made their way into our report through sensitivity analyses or sensitivity calculations that we'll be covering later in this presentation.

Next slide, please? All right, key findings. So, this is going to be a quick overview of the main findings of our analysis. In our report, we sort of have three key findings that we claim at the beginning of this report.

The first being that the increased burnup and enrichment is not strongly impacting our in containment source terms and that the most significant variation that we see in source terms is due to sequence variations.

The next finding is that the larger early releases to containment are the result of early pressure boundary failures, primary pressure boundary failures.

In our analysis, we had a higher predominance of those low pressure accident sequences that were sort of being mechanistically predicated by the MELCOR code. And this is in contrast to the NUREG-1465 document that had high pressure sequences.

The third finding here is that releases to containment are significantly reduced if you can keep that primary pressure boundary intact.

So, if we prevent that low pressure scenario from evolving, we're going to have smaller in containment source terms.

And this is sort of related to some of those findings from the SOARCA analysis, and we'll touch on that again in a couple of slides.

MEMBER MARCH-LEUBA: This is Jose March-Leuba. So, the key findings -- sorry, I've got something in my throat -- key findings two or three are not underlying. I mean, I don't need MELCOR to tell me that.

But key finding one is counterintuitive, right? If you have high burnup, you have a higher amount of inventory inside the fuel.

Could you -- I assume you're going to expand on this, can you give me a high level explanation of why increased burnup doesn't affect the source term?

MR. ALBRIGHT: Yes. So, when we talk about the in containment source term, we're not looking at the magnitude of mass that's being released to the containment. We're looking at the release fraction.

So, we'll go into that in more details in the next few slides. But I think that's the clearest cut way at this stage of the presentation to clarify that.

MEMBER REMPE: So, that's actually a comment I was going to have. And I hate to nitpick on words, but I think that you -- what you said is what you mean there is release fractions, not magnitude.

And you might, I guess, this is a final report, but you might think about not having that wording next time.

MEMBER MARCH-LEUBA: Yes, being a little facetious here, if this is the release fraction, it's another -- I mean, it's a no, never mind.

If those are your three key findings, you should be proud of yourselves. You didn't find any surprises, I don't know. Okay, keep going.

MR. ALBRIGHT: Thank you.

MEMBER ROBERTS: Because I don't want to nitpick, either, but I will.

The key finding three, releases to containment is going to be reduced. Isn't that true only for the early in-vessel phase? Because when I read the report, it seems like the total release to containment is probably the same in any of these scenarios.

MR. ALBRIGHT: Yes, yes, yes, that's a good qualifier for this statement is that we are focusing on the gap release and early in-vessel phases and we'll actually touch on the clarification of the phases later in this presentation.

MEMBER ROBERTS: Okay, thank you.

MR. LUXAT: I'll just very quickly say, when it comes to severe accident progression and burnup, what's primarily driving us the containment and what the heat level is.

For the fractional releases, you know, they're driven by decay heat. And what we saw from the Oak Ridge work on the decay heat is that, for the early phase, or essentially, the early times post accident, early cooling times, there isn't a significant difference in decay heat.

MR. CAMPBELL: Well, we can go to the next slide and you can see exactly that.

MR. LUXAT: Yes, thanks, Shawn.

MR. CAMPBELL: Spoiler alert. Thank you, Dave.

Yes, so, this next slide sort of highlights the SCALE analyses that were used as the initial conditions for our reactor core inventories.

And what we see on the left are the decay heats in terms of relative percent to the reference core here.

And I've highlighted in the black box there the time region of interest. And we see that we're always within 5 percent. In fact, less than 5 percent of the same decay heat during the reference period of interest for our analyses.

And so, the main point here is that our burnup and enrichment aren't changing the decay heat, which is one of the major drivers of the accident progression.

On the second half of this slide, on the right side, we actually are looking at the bootstrapped release fractions for the different nuclide classes of each of the core types, being 60 and 80 gigawatt low enriched, and 60 and 80 gigawatt high enriched, or HALEU.

And what this slide is showing, or what this figure is showing us, is that the differences in the source term across these different cores are actually very small so that the increased burnup and enrichment does not strongly impact the in containment source term.

Now, this is taken -- or this statement is really much more clearly sort of re-emphasized or reinforced by the next slide, if we can go there, where we look at the actual source terms based on the sequences.

Next slide, please? So, on the left side here, what we see are the same source terms for BWRs and PWRs looking at the different accident sequences.

And what we see are that we're actually seeing differences in those bar charts now.

So, the main point here is that accident progression and the in containment source terms are actually being -- the numbers we're getting are driven by the accident sequences themselves, not necessarily the reactor cores or the different burnups that we can have.

On the right half of this slide, we then look at the impact of that early depressurization of the primary pressure boundary.

So, the purple lines in both of these plots are going to be our reference cases where the reference -- or where the hot leg creep rupture is enabled.

And the orange dashed lines are going to be the cases where we disabled hot leg creep rupture.

And what we see is that that prevention of early depressurization of the primary pressure boundary or early loss of the primary pressure boundary, during these critical early phases of the accident are actually decreasing the source term, the in containment source term significantly.

Next slide, please? All right, so now, we get into sort of the high level tables or the main sort of cuts through our source term tables that we've developed here.

The first point that I want to make is we talked a little bit about the late in-vessel and the ex-vessel phases, you know, after lower head failure.

And those are reported in this analysis in keeping with the current state of practice and giving us the ability to compare to previous source terms like SAND2011.

But the NRC has actually determined that the design basis source terms won't include these two phases.

So, we're going to focus in our presentation today on the gap and early in-vessel phase values.

Now, the first point that I want to make here with these yellow highlighted boxes are that we have significantly longer in-vessel phase durations due to that progressive core degradation that I mentioned earlier.

This is a MELCOR advancement in the way we've modeled these accidents.

Next slide, please? Now, the next point here is looking --

MEMBER ROBERTS: Quick question.

Acknowledging what Kim said at the outset, I'm just curious, maybe for logic, are you reconsidering that 1994 cutoff?

MR. DICKSON: That's a Commission policy.

MEMBER ROBERTS: Oh yes, I recognize that, but it seems to me that going from 1.5 to 6.7 hours seems like a pretty major change in terms of what the overall outlook is of the progression of the plant transient and that.

I was wondering if that's something you're looking at?

Because the principle from that '94 SECY seemed to be, A, the release is about the same as the TID; and B, that's about the time it take before the operators could do anything where the casualty would become so bad that they can't do anything about it.

I think those are the two main preventions that were in there.

And again, it just seems like we go from times like 1.5 hours to 6.7 hours, that -- and this range now is roughly double in TID.

And it seems like the basis for that 1994 judgment might be something worth revisiting.

And I was just wondered if you're thinking about that?

MR. DICKSON: Yes, we can think about.

MEMBER MARTIN: I don't know what the right answer is, because clearly, it's a 30-year-old judgment that's been out there for a while.

But it just seems like it's something worth thinking about.

Thank you.

MR. ALBRIGHT: Okay, this current slide here, what we're looking at are the highlighted gap releases or the gap release phase values.

And what we're seeing is that the enhanced reactor coolant system modeling that we have in MELCOR today allows for the progressive releases to containment.

And this is where start to see that the gap release phase, essentially, now we're seeing those fission parts actually transporting through the reactor coolant system out to containment.

And this is, again, related to that gap release phase sort of no longer being distinct from the early in-vessel phase.

Next slide, please? Finally, I want to highlight the larger release magnitudes, release fraction magnitudes that we're seeing in the current study. We've highlighted them here.

Now, primary reasons that we've already discussed for these are the progressive core degradation that we're seeing in MELCOR and the longer durations to lower head failure.

So, essentially, we've got our fuel and our debris are being held at longer time periods at high temperatures during these early in-vessel phases in the current MELCOR calculations.

And that's driving basically larger releases that are then captured in containment based on this early loss of that primary pressure valve we're seeing in our simulations today.

So, that sort of gives a quick overview of some of the major differences between the 2023 source terms and the NUREG-1465 in containment source terms.

Next slide? Oh?

MEMBER BIER: Sorry. I have a very high level question but on this kind of modeling at all.

When the new analysis was done, was it done kind of from, I don't want to say post principles like basic physics, but, you know, modeling the whole scenario from scratch?

Or was it done by kind of looking at the previous analysis and figuring out where you can take advantage of improved fuel characteristics?

MR. ALBRIGHT: Yes, that's a great question. So, the current analysis, what we used as our starting point were the actual reactor models and practices from the SAND2011 report.

And the first thing that we did was bring those inputs up to modern MELCOR best practices.

And then, we actually used the same scenario models, except for where best practices have evolved, so these early pressure failures -- or early primary pressure boundary failures.

And we maintained as much consistency there as we could with the 2011 values so that we could actually compare apples to apples.

MEMBER BIER: Okay. I guess the reason I'm asking is, in my area of PRA, I know there's always a tendency to look for like where you can sharpen your pencil to get better numbers and not, you know, is there some unexpected phenomenon that gives you worse numbers.

But it sounds like you tried to do it pretty even handed.

CHAIR PETTI: So, but like you talked about how you modeled -- how you incorporated line from Phebus and like.

So, the release rates are fundamentally

different, right, in these calculations and in the 1465?

MR. ALBRIGHT: Yes, so, in terms of release rates, we actually use a different release rate correlation for high burnup fuels.

And that is incorporated and based on the validation of the RT-6 VERCORS experiment using the MELCOR code.

CHAIR PETTI: Yes, so, it's a combination of better accident progression and release of what's --

MR. ALBRIGHT: Absolutely, yes, yes, yes.

What we've tried to do is maintain consistency where we could and advance the previous practices to modern practices where anything has evolved.

CHAIR PETTI: Kind of just scary that these numbers are heading back towards the TID source terms. We spent billions of dollars on this.

MEMBER REMPE: That's why I was asking about how MAAP would compare if --

CHAIR PETTI: Right.

MEMBER REMPE: -- they have a different failure time. It's earlier, they could get back down again.

CHAIR PETTI: Yes, that's another question about how you decided what sequences to look at in terms of establishing this amalgamated source terms, if you will, given there are thousands of severe accident scenarios, right, that you could have?

MR. ALBRIGHT: Sure.

And specifically in the selection of scenarios, you know, we fall back on that NUREG-1560 and I think we talk about it later in a later slide where we talk about that.

What we're trying -- that's where we try to stay with consistency. Right? We tried to stay consistent with the 2011 practice of scenario selection. And we have a NUREG basis for those choices.

MEMBER MARTIN: The numbers that get reported here, they reflect, of course, the calculation will talk about -- mention, of course, in certain analysis.

And it's been a cold month since I've looked at the 2023 Sandia report.

But I believe uncertainties were, as I recall in that document, uncertainties were discussed, addressed.

When we look at numbers like this, I mean, these are kind of mean. Whether they -- that is the -- it's implied or stated, correct?

MR. ALBRIGHT: Yes.

So, we can -- we'll go into the more detail on that statistical process that was implemented in this study.

But these numbers are the median of the distributions that we selected. And that's based on the accepted practice from the -- I believe it was the peer review for the 2011, actually, that sort of was the first assertion I'm aware of that said that that was the most representative so that we weren't unequally weighting certain scenarios.

MEMBER MARTIN: Will you be discussing what all went into the uncertainty analysis part?

MR. ALBRIGHT: Yes, we'll have a number of slides, I'm forgetting off the of my head right now, but I think it's a handful of slides on the process that we used to arrive at the distributions that informed the numbers that we present.

MEMBER MARTIN: And maybe you can say it ahead of time, I mean, is the biggest uncertainty really the event itself, to Dave's point, that --

MR. ALBRIGHT: Yes, yes, the largest uncertainty that we have in this study is the sequence.

The variation in sequences is larger than any other variation that we observed.

MR. LUXAT: Yes.

So, it's sort of common, if you look at a Level 2 PRA. Your end states and your releases are typically a distribution, but it's really going from one branch to the next. One end state to the next that really causes the big changes in release.

And any phenomenological variation about that particular sort of branch or sequence is typically gives you a Gaussian, gives you distribution, but it isn't enough to necessarily push you from one end state release category into a completely different release category for another state.

And so, it's -- what we found is typical of what we always find in Level 2 PRAs.

MEMBER MARTIN: If you played around that space, I mean, we talk about cusp events in different context a bit earlier today, I mean, if you had a survey to look at, you know, event outcomes and with an eye towards, you know, those events that you would otherwise classify as like cusp events, I mean, do you think we've crossed a line, a cusp of such?

Of course, a severe accident by nature, we've crossed the line.

But when it comes to releases, are there tiers where you see kind of clustering of, you know, these events, you know, kind of land in this cluster and the more and more severe as you look at worse and worse conditions?

MR. LUXAT: I think for this particular set of scenarios, we're dealing with unmitigated scenarios.

And so, really, the big -- the main changes are, A, the initiating event, is it a LOCA or is it an SBO?

And also, the other main issue is the integrity of the RCS or the nuclear steam's ply system pressure valve.

And it's those two features that typically give you, if you will, the underlying bifurcations that push you in a direction of one cluster versus another.

And obviously, if we were to go down further and further into event trees with mitigating actions, you would see other types of bifurcations.

But within the scope of essentially unmitigated events early in-vessel source terms, it's really those two characteristics that are principally influencing the nature of the source term.

MEMBER MARTIN: Thanks.

MR. ALBRIGHT: Okay, I think we've covered this slide, so next slide, please?

This next set of -- couple of slides is going to be related to the release rates.

And we just wanted to sort of highlight here that, with the longer phase durations, the release rates, when we assume uniform release across the phase duration are decreasing quite significantly relative to NUREG-1465 for many key radionuclides.

Next slide, please? So, these are release fractions per hour. So, this is a very simple calculation. Take the total release fraction, divide it by the phase duration.

And again, what we're seeing is that, in general, these are much smaller.

We see -- next slide, please? Thank you.

We see that, for the tellurium group, these numbers are increasing. And the main reason here is because we didn't used to assume what tellurium was going to transport efficiently to containment.

So, this is that advancement in our

understanding of severe accidents that's resulting in this change or larger release rate tellurium.

Next slide, please? Thank you.

Okay, so -- yes?

CHAIR PETTI: Just a question.

So, there's no reaction with any of the still surfaces in the primary system highly reactive for metal?

MR. ALBRIGHT: So, our tellurium release rates are being informed based on the validation matrix for radionuclide transport.

So, this includes, in particular, that VERCORS RT-6 experiment.

And what we do is we're actually, how do I put this, the tellurium release rate is not going to look at any downstream chemistry. Right?

We have frozen chemistry in MELCOR. So, the only way for the tellurium to be release is direct release. Right?

We're not actually looking at any chemistry on its way out of the fuel, if that makes sense. We've assumed tellurium is going to transport out of the fuel at this rate and it will have these transport properties.

And that goes back to our frozen chemistry assumption in MELCOR.

CHAIR PETTI: But physically, one of the most reactive metals, you would think that it would probably react with steel as it's, you know, coming between the core and the break.

MR. SALAY: Mike Salay.

It was before my time, in Phebus, they actually -- and if they'd observed a chemical reaction in Phebus, they would have done it.

And Phebus had a model steam generator and RCS and containment. So, they would have seen something, seen retention, specific retention there. And if they'd seen it, they would have accounted for it.

CHAIR PETTI: So, all these old discussions I can remember with Dana on fission product revaporization, because it's on the surface of the primary system and it's going to self-heat. That's all gone, we don't do that anymore?

MR. LUXAT: We do.

CHAIR PETTI: Okay, we do it for some fission products but not for others? I'm confused.

But Phebus probably didn't show that, did it?

MR. LUXAT: We have a set -- we do model

revaporization and we also do the chem absorption on surfaces and tellurium --

CHAIR PETTI: So, you do model a tellurium chem absorption?

MR. LUXAT: There is tellurium chem absorption and the coefficiency in the validation data.

And we do actually have them as a --

CHAIR PETTI: Okay.

MR. LUXAT: -- show on surfaces.

Now, tellurium is probably not -- it's not a very dominant one from what I recall in terms of chem absorption. I think some of the cesium that, obviously, cesium hydroxide is a more dominant one that we typically consider.

But we do have chem absorption models and --

CHAIR PETTI: But since your position now assuming it's all cesium molybdate instead of cesium hydroxide, there's not a lot of cesium chem absorption.

MR. LUXAT: And cesium -- there is some cesium iodide.

CHAIR PETTI: Yes, cesium iodide, sure.

MR. ALBRIGHT: Okay, I think that covers

this slide.

Next slide, please? Going into the deep dive, so just getting everyone, including our members in the virtual audience on the same page, what do we mean by in containment source terms?

This is a little late in the presentation, but what we're talking about is the total radioactive inventory in containment.

So, what that means is, we combine all of the different sort of phases of radionuclides, airborne, liquid, or anything that's escaped the containment in the case of the later vessel phases.

We combine that all into one single value, the in containment source term.

What you see here on the figure in the right is some of these different values, deposited airborne escaped. And then, total, which is that top black line at the very top here.

For one example case, the halogens. And it just goes to show, you know, the many processes that MELCOR is tracking and modeling that we then have collapsed into the numbers that we're presenting in the tables for this analysis.

The reason we do this is because we have downstream codes in the process that are meant to handle these mechanisms.

So, we have to actually, you know, post process them out of our MELCOR simulations for those codes to do what they need to do.

MR. SALAY: Just to clarify, you don't mean Sandia has downstream codes?

MR. ALBRIGHT: Yes, I apologize.

Yes, yes, particularly the RADTRAD code that's mentioned on this slide which is done later in the regulatory process for these regulatory source terms.

Next slide, please? So, what is an alternative source term? Basically, the concept and requirements of alternative source terms were defined in Reg Guide 1.183.

We've sort of boiled down the five criteria here that it has to be based on major accidents involving substantial meltdown of the reactor core.

It needs to be represented in terms of quantities, times, rates, and speciation of a fission product release.

It needs to be based on a representative set of accident scenarios.

It needs to have a defensible technical

basis.

And it must be peer reviewed.

These criteria were all met according to the peer review process that was completed in 2023 and published in like at the same time as the SAND2023 source term report.

It's listed on the left there for anyone who wants that reference.

Next slide, please? So, how do we develop these source terms? We're looking at light water reactors, so we go and we find the accident sequences that are relevant for a BWR or PWR.

We develop a radionuclide inventory and decay heat for the particular reactors of interest using the SCALE code package.

Then, we perform our accident progression and source term analysis using MELCOR.

That's the part that we do, the SCALE code package was completed by another team.

And then, finally, we develop the statistically representative source terms based on that data. And that's another part of our job.

Next slide, please? So, as I mentioned, we were in this process starting from SAND2011 inputs and trying to maintain consistency so that the two reports could be sort of compared reasonably.

In that SAND2011 report, we -- sorry, excuse me -- in our 2023 report, we focused on extending the SAND2011 source terms to look at higher burnups and HALEU fuel.

The main sort of overlap that we maintained between these two studies were the power plants that we modeled, the scenarios -- the accident scenarios that we simulated, the chemical classes represented, and the sort of phase criteria.

So, when does a given phase start and when does that given phase end?

And finally, the sort of statistical process that was used to develop the final values was maintained across these two studies.

Next slide, please? Okay --

DR. SCHULTZ: Lucas, while you're --

MR. ALBRIGHT: Yes.

DR. SCHULTZ: Was there any reason to make changes in those elements of analysis?

MR. ALBRIGHT: Any reason to make changes to the methodology?

DR. SCHULTZ: As you described them, you maintained those from first analysis to the second.

Was there anything pulling on you to make changes to those?

You indicated you had maintained them, so you had comparative.

MR. ALBRIGHT: Yes, yes.

DR. SCHULTZ: Is there any reason why you would, but I just wondered because you did the analyses, or the team did, and it didn't work.

Did anything pull you in a direction to change those fundamentals?

MR. ALBRIGHT: So, I think at the outset of this project, the goal was to extend those source terms with the current state of practice.

So, we weren't necessarily drawn to evolve any of the practices because we were trying to sort of do things according to the current state of practice, not necessarily develop new practices as part of this process, if that makes sense.

I think -- oh, go ahead.

DR. SCHULTZ: The PIRT basically validated that? The evaluations that were performed?

MR. ALBRIGHT: Yes, the PIRT findings really informed the way we interrogated the uncertainties with high burnup fuels here.

And in terms of the methodology that was used, that SAND2011 report was also peer reviewed.

And in that peer review itself, that was where sort of that state of practice was accepted and that we tried to sort of maintain for that comparison of apples to apples for this set of data.

MR. CAMPBELL: And just to piggyback off of that.

And then, confirmed by our own peer review. Right?

So then, for the 2023 one of the things that they stated was methodology was acceptable.

So, we were jumping off of the basis of the 2011 peer review and then, confirmed by the 2023 peer review.

MR. ALBRIGHT: Thank you, Shawn.

Let's see, okay. So, in this slide, we're sort of giving a summary of some of those details that we just talked about.

So, we looked at BWRs and PWRs, looked at four different containment types for these two reactors, a Mark I containment with the Peach Bottom reactor and a Mark III containment with Grand Gulf.

For the PWRs, the pressurized water reactors, we looked at an ice condenser containment looking at Sequoyah, and a large dry containment looking at Surry.

In terms of accident scenarios that we used in this analysis, we focused on the same that were used in 2011, the small break LOCAs, the large break LOCAs, the short-term station blackouts, and then, two additional scenarios, the long-term station blackout and anticipated transient without scrams for the BWRs.

The table that we're looking at here is a summary of the release phase criteria or the accident phase criteria that were used in the analysis. These were refined back in 2011 and used here so that our numbers were comparable.

Again, I think the sort of main points here are the gap release phase starts when the RPV water level reaches the top of active fuel and it ends after 5 percent of the total xenon inventory release from the fuel.

The early in-vessel phase releases 5 percent -- starts when we've released that 5 percent of xenon and it ends at lower head failure.

The two other phases that are not considered in these design basis source terms, the ex-vessel and late in-vessel both start when that lower head fails.

And then, they have their own criteria

based on the cesium releases pertaining to those different bases.

Now, the peer review findings in regards to these details were that the ex-vessel and late in-vessel phase criteria have limited technical justification.

Again, these two phases are not really used for the design basis source terms. And that's sort, I guess, something for our discussion that you brought up earlier.

So, I think that covers the details for this slide, if we want to move to --

Oh, yes? Sorry, no, go ahead.

MEMBER REMPE: I have a question on that, is it unfair to ask you, but I think maybe the staff could comment on it.

A couple slides back, you talk about what the AST should be based on Reg Guide 1.183.

And you talk about not a single accident scenario.

Risk guidance insights can be used, but there's a phrase you didn't -- or a sentence you didn't mention, however, risk insights alone are not an acceptable basis for excluding a particular event.

And do you think, I mean, just saying,

well, we did what the folks in 2011 did and they did it based on the IPE.

Have we met this last point that's still appeared in Reg Guide 1.183 Rev 1? Have you gone back and thought about, did we exclude some things because of risk assessment?

I don't quite understand why that sentence is still in Reg Guide 1.183, but have we done that?

MR. CAMPBELL: There wasn't even a consideration of trying to look at scenarios. People have mentioned that perhaps we should look at scenarios more, but yes, so --

MEMBER REMPE: Containment bypass, for example, well, you've kind of said, well, we don't think we're going to have it because the hot leg fails earlier.

But --

MR. CAMPBELL: Well, the containment bypass, I think we're -- I think explicitly excluded or from the set of scenarios because the whole purpose is to develop a source term to test your equipment in containment.

And so, it literally excluded those.

MEMBER REMPE: Yes.

MR. CAMPBELL: And so, they tried to get

the -- based on the IDs, as big a fraction of the core damage frequency as possible for both Bs and Ps.

MEMBER REMPE: Good.

I'm just wondering if maybe either that sentence ought to go away in Rev 2 or maybe we'd better have a reason why we didn't think about things that might have been excluded Because of risk insights.

MR. ALBRIGHT: Are you suggesting a scenario that needs to be added?

MEMBER REMPE: I just didn't want it -- we need to think about a scenario that should be added. It's just an interesting thing that has caught my attention doing the one letter on Reg Guide 1.183 and here it is again.

MR. ALBRIGHT: Sure.

And like we said, that was kind of beyond the scope of this work. We were trying to be consistent. We were trying to have an apples to apples comparison and extend the basis. Right?

It was an extension exercise at this point. Right?

If there's a need or a desire or we see the potential need to go and re-evaluate scenarios, again, that's another research project that we can --

MEMBER REMPE: Yes, maybe change the Reg Guide.

MR. ALBRIGHT: Right, there you go.

MEMBER REMPE: It's just a comment.

MR. ALBRIGHT: But at this stage, we haven't seen a need, I guess that's what I went back to.

MR. ESMAILI: This is Hossein, can I say something, Joy?

Okay, so, we think that these scenarios, right, what's driving this things? What happens during the scenario? Not what is the initiating event, right?

And so, there are cases that, you know, start at high pressure and they have shown that, you know, the hot leg -- either the hot leg fails or they have started.

But those have, by far, the biggest influence as you have seen in distortion, how things get released from the, you know, RCS into the containment than whether you are considering a, you know, whether a two-inch LOCA or whether it's a four-inch LOCA, et cetera. Right?

So, starting from the core damage, what happens during the core damage? And you know this even better than I do, is what's driving this source?

And I'm just looking at Dave and Lucas just, you know, to confirm it.

So, we can throw in other accident sequences. You know, another LOCA, maybe another thing. And I'm not suggesting we shouldn't, I'm just saying that we know that what the answer would be.

They're going to fall somewhere in the -- between here. Right?

And so, we have captured the most important phenomena that occurs during the accident progression.

I want to make one thing clear, is that the input deck, the other thing I wanted to make clear, is that the input decks, as we use for in 2011, we did SOARCA analysis, right, we did the uncertainty analysis.

And so, the input decks actually evolved. It was not just the cold experiment and everything that we learned. We learned how these plants are built, you know, what is the insulated, what's not insulated.

And in some cases, we understood, you know, that they make a lot of difference.

After all is said and done, so we have

improved over these past ten years or so, we have also improved our input deck.

But that finding three says that what's driving these things is the failure of the pressure valve.

If you keep that intact, you are going to be where you were, you know, in NUREG-1465.

Does that -- okay.

MR. LUXAT: I just wanted to say, like Hossein said, the primary driver here is the release of fission products from the fuel. And it's really what pathways exist to get those fission products out of the primary system into the containment that's sort of driving the magnitude and timing of the buildup of fission products in the containment.

And as we'll see a little bit later, one of the discussion that we got in with respect to the peer review was related to, you know, okay, there's the core damage, but then, there's also looking more deeply into the transport of fission products from the core through the primary system and ultimately into the containment.

And Shawn is going to be talking a little bit later about some of the follow on work that came out of the peer review looking at the transport pathways in more detail to better understand where fission products have migrated into different regions of the containment or the primary system.

And then specifically, do different regions that have or can influence different types of release pathways be it through an MSID or through a containment back area.

And we'll be discussing that in more detail.

But it's really from the perspective of the accident scenarios, we got core damage.

And then, it's really some of the additional failures that are influencing the transport pathways or transport of fission products to key release pathways that can get them out of the containment that are dominant.

And Shawn is going to be going into that in a little more detail soon.

MR. ALBRIGHT: Thank you.

Next slide? Okay, so the accident selection. The accidents that we used in this analysis were informed by that NUREG-1560 that I mentioned earlier, the IPEs, or the individual plant examination program.

And basically, in 2011, they took that IPE to develop the accident sequences that they looked at so that they were consistent.

And then, we have followed those through.

And these are actually similar to what was selected for NUREG-1465. So, doing this, it gives us apples to apples comparison with NUREG-1465 as well.

And this provides coverage of all of our major severe accident -- unmitigated severe accidents.

It incorporates, again, the station blackouts, the LOCAs, and the ATWS scenarios.

And the peer review during that process acknowledged that there are more recent PRAs that would potentially show different core damage contributors.

But overall, for the intended application, this set of scenarios is appropriate with regard to the progression of these unmitigated severe accidents for evaluation of the in containment source term.

Next slide, please? So --

MEMBER ROBERTS: Can you provide any insight as why they said that?

So, they concluded that the set of reactions you had were sufficient?

MR. ALBRIGHT: Say that one more time?

MEMBER ROBERTS: So, why couldn't -- why

did the say the reactions you have were sufficient given they may not be the right ones?

MR. ALBRIGHT: Well, I think -- yes, yes.

So, I think their characterization wasn't that they were the wrong accident scenarios, but that there may be more recent plant proprietary PRA studies that have different contributors to that core damage.

So, again, I think the big picture is that we're covering the spectrum of severe accidents here and that, for the development of this representative unmitigated severe accident source term, that the current set of scenarios provides that coverage.

MEMBER ROBERTS: So, if something that you didn't look at was worse, that's okay because of the nature of why you do the studies now? Is that the way to translate that?

MR. ALBRIGHT: Say that one more time?

MEMBER ROBERTS: Yes, what I think you said is that they -- for the intended use, this set of accidents is sufficient.

But I -- is having your why? And one reason would be that, you know, they're bounding.

I don't think you said.

Another reason would be, they're not bounding, but because of the way they're used, it doesn't matter that they're not bounding.

MR. ALBRIGHT: Okay.

MEMBER ROBERTS: I guess I'm trying to figure out which it is. And you get the latter, you know, what is the thought of what the intended use is?

Because there's a lot of intended uses.

MR. ALBRIGHT: I think I understand your question now.

And you know, responding on behalf of the peer review committee, I'm speculating here.

The peer review committee did not identify scenarios that we missed. They identified that there may be different contribution percentages to the core damage frequencies.

So, it wasn't -- they didn't say that you missed a scenario, they said, the relative percentage of the different scenarios might be different than what was done in NUREG-1560.

MEMBER ROBERTS: Okay, I see.

So, if you had a more complete set, they don't think you would have a larger answer?

MR. ALBRIGHT: I don't think -- I don't know that the peer review committee made any statements regarding how the numbers would change.

I think the only statements that come to mind are that the current spectrum of accident sequences provided the coverage of the potential accident sequences that were expected to be covered.

CHAIR PETTI: But there's other PRAs that show that the frequency of those events may be quite different.

So, when you stack them all up into a CDF, their percentage contributions, depending on how you weight the sequences, are different.

MR. ALBRIGHT: Yes, I believe that was the intention of that first statement and I guess the final statement as well.

DR. ESMAILI: May I say something? This is Hossein Esmaili, again.

So, yes, you're absolutely right, except that they are not looking at the frequencies, you know, when they're coming that they have representative source terms.

And the other thing I'm speculating is that, you know, the peer reviewers, you know, they have been doing -- a lot of them have been doing this for a long time, as I mentioned earlier.

The accident signature, the core damage signature, right, it does not change as long as they have representative station blackouts or representative LOCAs, et cetera.

And even in PRA, you know, we combine, we don't look at every accident sequence, we just combine a lot of them into a single plant damage state because we know the accident progression itself does not change.

You know, we are still -- it's an unmitigated accident scenario. We're still melting the core. You're still looking at, you know, oxidation energy.

So, a lot of these accident signatures that we have, they are captured.

And again, as I said, you know, what changes here is, you know, during this progression, what happens to the pressure boundary and how do you get these things from the reactor into the containment itself.

So, those are the important main drivers rather than, you know, what is the initiating event? And how do you combine them? Et cetera.

So, I think they believe, and this is my person opinion, that we are capturing, in terms of, you know, accident signatures, et cetera within this.

And you are absolutely right, you know, we have done SOARCA uncertainty analysis. Even when you do SOARCA uncertainty for a single accident, you're seeing a rate and, you know, what parametric do you use.

But our systems are not unbounded. Right? I mean, we have a system that can produce this much hydrogen, you know, you have this much metal, you've got this much water, et cetera.

So, we understand, you know, what that range of uncertainty is and whether you are doing uncertainties with accident scenarios or parametric, et cetera, so we know where we end up.

And I'm just, again, I'm just speculating that that's what they are.

Thank you.

MEMBER REMPE: So, my question that was the unfair question about the Reg Guide 1.183 actually was motivated by some of the peer reviewer comments where they were suggesting, did you look to see if these scenarios that you've selected are really capturing the dominant ones at this time from a risk perspective?

And my thought at the time as well, even if you had something that wasn't important, you should leave it in because of that one sentence or if you've missed something, well, you probably should.

And I think it might not -- it wouldn't be a bad idea to kind of look at that at some point, not this version, but I think you're even planning to do another updated AST in the future, somewhere I read or someplace to consider more things. And the next time, maybe think about that.

MR. CAMPBELL: We're going to have a new AST coming, I'll just say that. We're not redoing 2023 at this point, but we're doing really additional follow on work to explore additional sensitivity to that type of thing.

MEMBER REMPE: Yes, it might be something to look at.

MR. ALBRIGHT: Thank you.

So, the next few slides will sort of give that overview of the accident scenarios that were covered here.

We break them out into sort of four categories.

We've got initiating events, coolant injection, RPV status, and containment status.

In terms of Peach Bottom, we looked at seven SBOs, four of them were short-term and three of them were long-term SBOs with that prolonged DC power.

And then, we had two LOCA scenarios.

For coolant injection, we had RCIC operation in three of the scenarios, the long-term station blackouts.

We had no coolant injection in the short-term station blackouts.

And we had actually no coolant of any kind -- no coolant injection of any kind in six of those scenarios.

For the RPV status, we had the potential for, I will say, high pressure scenarios in that we didn't prescribe failure of that primary pressure boundary.

The accidents were allowed to evolve according to the MELCOR calculations. And if that SRV reached its failure criteria, then it would seize.

In terms of containment status, we had early failures and late failures. We'll see this in the following slides as well.

We looked at drywell liner melt through, torus overpressure, drywell head flames leakage for those early failures.

And then, for the late failure, we saw a high temperature failure -- or I'm sorry, that should say overpressure failure. I'm not sure why that one got mixed up.

Next slide, please? For Grand Gulf, we see much the same in sort of the overview of the scenarios here.

The main difference is that we considered at ATWS scenario for the Grand Gulf plant.

And I'll just jump to the end here where the main difference is in terms of the containment failures.

Again, we had those early failures including at a high containment pressure failure prior to core degradation in the ATWS case.

And then, the same high containment pressure failure for the late failure.

So, Grand Gulf and Peach Bottom scenarios, SBOs, LOCAs and ATWS.

Next slide, please? In terms of the Surry accident scenarios, we considered two station blackouts and three LOCAs.

So, you'll notice that, in this case, we've actually shifted the PWRs are showing more LOCAs than station blackouts in this analysis.

We credited cooling in one of the scenarios and didn't have any coolant injection in four of them.

We, again, have the possibility of high

pressure scenarios for the Surry analysis where, if the model would predict hot leg creep rupture, then we would allow it to occur and it actually ended up happening in all of those scenarios.

So, all scenarios were low pressure scenarios.

In terms of the containment failures, we see the same containment failure possibilities here with hydrogen deflagration at head failure and high containment pressure as a late failure mechanism.

Next slide, please? So, Sequoyah accident scenarios, this is our last plant here.

Again, I'll point out, we've got more LOCAs in this model as well than station blackouts.

Five of our scenarios credit coolant injection, while two of them are not considering coolant injection.

Sequoyah also had the possibility of the high pressure scenarios but ended up predicting hot leg creep rupture in all cases.

And then, we had the same containment failures.

I will make a general note about the different models. The containment failures occurred at or after lower head failure for folks who are maybe wondering about that.

The only case where that wasn't true was that ATWS that I mentioned earlier where we -- an overpressure during the ATWS scenario before core degradation.

Next slide, please? Okay, so, the next couple slides, we'll be going into the radionuclide inventories. I know this is a wall of numbers.

MEMBER ROBERTS: Yes, Lucas, is that a new insight that you could get containment failure before a lower head failure?

MR. ALBRIGHT: No, that was actually prescribed in SAND2011 as well.

MEMBER ROBERTS: How about the NUREG-1465?

MR. ALBRIGHT: Off the top of my head, I do not know, but we could definitely go look at that.

MEMBER ROBERTS: Yes, it's very, again, half my question for Elijah, is it -- if you've got scenarios where lower head failure happens after containment failure, that's yet another reason to question that SECY-1994, you know, philosophical or principle, or whatever assumption.

Because if it's based on that's the point at which you could reverse the transient and make things better and if you had containment failure before lower head failure, that doesn't seem to make a whole lot of sense.

MR. ALBRIGHT: Thank you. So, as I was saying, this is a bit of a wall of numbers here, and I recognize that. I think it's actually more important that we consider the relative findings from the peer review committee in that process.

So, what these tables on this slide and the next slide are showing us are large changes in mass for the different radionuclide groups.

And you know, up to 51 percent in this particular slide.

What I want to make clear is that the radionuclide mass differences are not equal to the differences in activity that result from those changes.

So, a 50 percent increase in mass does not correlate to a 50 percent increase in activity.

And this is important because what the downstream codes are going to look at is the dose based on the activity that's released.

MELCOR is looking at the release fraction according to the current state of practice. And that release fraction is reported in terms of the mass release.

So, one of the peer review committee members had a note that it's actually unlikely -- based on the actual changes in activity that are being observed here, that it's unlikely that the siting calculations will be impacted by these changes in burnup because key radionuclides like iodine are not actually changing in activity across these burnups, if that makes sense.

Next slide, please? So, this is the same table for the pressurized water reactors. I think I've probably covered this in enough detail at this point that it's important these changes in mass are not equal to changes in activity.

And the impact of the larger masses is essentially going to reveal itself in these downstream calculations that are looking at the actual change in activity as the result of the change in burnup an enrichment.

Next slide, please? This slide covers our iodine and cesium chemical forms. So, this can get a little bit confusing with all of the numbers we've got here, so I'll try and be clear.

The NUREG-1465 analysis assumed 5 percent of the iodine inventory was gaseous, either elemental iodine or organic iodines.

Ninety-five percent of that inventory was assumed to be in the form of cesium iodide and the remaining cesium inventory was assumed to be involatile cesium hydroxide.

This is a very large difference between current practices which are reported in the I think it's the NUREG-CR-7008 or something like that that's the MELCOR best practices for iodine and cesium chemical forms.

We've reported them here. They're consistent with those base case SOARCAs.

And in this analysis, 100 percent of the iodine inventory is assumed to react with cesium to form cesium iodide.

Five percent of the iodine inventory, so 5 percent of cesium iodide is placed into the gap. And then, the remaining 5 percent of cesium hydroxide is made up of -- sorry, of cesium in the gap is made up of cesium hydroxide.

So, we've got cesium iodide and cesium hydroxide in the gap in our current MELCOR calculations.

And then, 95 percent of the cesium is represented as cesium molybdenum.

So, big picture, uncertainty in iodine

speciation is persists to this day despite several experimental studies.

We had several discussions with our peer review committee regarding the FPT3, DF-4, and BECARRE experiments that highlight some of the remaining uncertainties that's here.

Fukushima Daiichi, as I indicated in the post accident analyses is confirming our assumption that cesium molybdate is the dominant chemical form of cesium.

And finally, the peer review committee, Joy, for your information, I think you mentioned this earlier, has recommended that we do go and look at some of the other experiments like the VERDON experiments in terms of how we look at these aging analyses.

MEMBER REMPE: So, I think I heard yes, we're going to today. But it wasn't clear then in the actual report. So, it's still not clear when, but I just wanted to bring it up.

MR. CAMPBELL: We'll get back to you. Like I said, we're always trying to adopt the best practices. But at the time of these calculations, that was the best practice. But we're always evolving.

MR. ALBRIGHT: Next slide, please? So, the next few slides are going to cover other analysis assumptions. I'll try to get through these quick.

Basically, we aren't looking at any variations in gap inventory at the start of the accident. We're assuming in every one of these scenarios the same speciation and gap inventories that I mentioned a minute ago.

We're also not looking at the fraction of aerosolized iodine in containment or any radionuclide retention or removal mechanism.

So, if you'll remember, MELCOR is calculating these numbers, but we are clapping them all back together to the total inventory in containment.

The source terms that we're presenting are consistent with the state of the art or current state of practice.

Many of these practices were established under the SOARCA project and have been published under that NUREG that I mentioned a little while ago, the MELCOR best practices.

We did use the latest code version at the time labeled MELCOR 2.2. And that version was released back in 2021.

Some modeling practices have evolved since SOARCA that we've incorporated into the analysis that we're presenting today, namely three items.

The time at temperature fuel rod failure model. So, basically, it's a lifetime model that defines how long a fuel rod can stand at a given temperature has been modified to use our default model that was informed by the VERCORS experiments.

We have changed the liquefaction and oxidized fuel failure temperatures for UO2 and ZrO2 to 2479 Kelvin. This is the mean value from the SOARCA analyses -- the SOARCA uncertainty analyses.

So, we took this as a more representative value than the default MELCOR value.

Next slide, please? We have assumed in this analysis that the relative contribution of different accident sequences for both PWRs and BWRs is not being changed by the different core types that we've got that we're looking at. So, the high burnup and the HALEU cores.

We've looked at the -- sorry, we have analyzed the aleatory uncertainty or the range of accidents here through our analysis.

But we're not really looking at any parametric uncertainties except for the sensitivity calculations where we investigate key phenomenological uncertainties that were identified as part of the HBU, HALEU, ATF, PIRT as being different or as being distinct for high burnup or high burnup HALEU fuels from conventional fuels.

So, we're not looking at uncertainties that exist in conventional fuels. We're looking at uncertainties due to the differences of the fuels that are being analyzed here.

We are not looking at the containment removal mechanisms, as I mentioned. So, containment sprays, any sort of deposition, suppression pool scrubbing, we're not crediting that in the numbers we're reporting today.

But again, we will be talking about that in the follow up presentation with the follow on calculations that we've done since this release.

Finally, the release fractions, anything below 1E to the negative 6 was considered negligible and it was truncated to and reported at 1E to the negative 6 in this analysis.

Next slide, please? So, here we are at the non-parametric statistical analysis. I'm sure some of you have been looking forward to this slide based on previous comments.

So, this is a non-parametric bootstrap methodology. And what we're trying to do is we're trying to explore the uncertainty across these scenarios.

And give an idea of the uncertainty distribution that we might expect based on the spectrum of scenarios that we have considered.

So, the strengths of this analysis or of this method is that it can be applied to data that follow any distribution.

We're not assuming normality or anything like that in this analysis.

It's a bootstrap methodology. So, we're repeatedly resampling the existing distribution of results.

And then, we're using that sampling to actually develop a mean empirical cumulative distribution reduction.

And then, we have this ECDF for each quantity of interest that we're interested in. So, phase duration, cesium release fraction during a given phase, et cetera.

Then, we look at that distribution and we actually report the 50th percentile of the ECDF. This has the bonus of equally weighting our simulations.

We're not sort of, you know, biasing our results towards one end of the spectrum here.

On the plot -- in the plot on the right here, you see the early in-vessel phase duration results for SAND2023 relative to the NUREG results.

So, what we're looking at with the points along the solid line are the actual percentiles that we've calculated for each of these distributions through this bootstrap process.

And then, we select the 50th percentile as our reported representative source term, in containment source term.

And the dashed lines that you see around those solid lines are the uncertainty that's described here as spanning plus or minus standard deviation at each percentile. And the lines have been smoothed out here so that it looks continuous.

Yes?

MEMBER BIER: So, I am far from a bootstrapping expert, so if I spent more time on this, I might have a different opinion.

But personally, I would really rather see things based on mean values rather than medians.

Basically, I mean, I understand what you're saying that you're weighting all cases equally.

But it matters whether the times you're above the median, are you above by a little bit or by a huge amount? And mean captures that.

So, you know, I don't want to, you know, send you guys back to redo everything, but in future, I would sort of suggest doing things both ways and presenting both because it does give different information.

My two cents.

MR. ALBRIGHT: Thank you.

Yes, this was definitely a point of discussion with the peer review committee as well.

And ultimately, they determined that not biasing the results using this 50th percentile was what they found acceptable.

But thank you for, yes, I definitely agree with the points that you're making about the differences between choosing a median or a mean.

Oh, there's the point right there, actually, at the bottom of the slide.

The peer review did find that those median values are appropriate because they're not introducing that bias that we were talking about a minute ago from potential outliers.

Next slide, please?

MEMBER MARTIN: Just real quick. On the previous slide, please, looking at the plot and, it's probably just confusion or maybe not, if they're a cumulative distribution ruptures and the PWR1 has a shape that you kind of expect, nice S curve there.

And the BWR isn't. That does suggest, you know, I asked the question about a cusp effect, how should I interpret that shape there?

MR. ALBRIGHT: Yes, I think the -- if I remember your comment earlier correctly, you were asking, do we see clustering for certain accident sequences of source terms, is that a correct summary?

MEMBER MARTIN: That's correct, yes.

MR. ALBRIGHT: Yes. So, and the answer is yes, we see discrete clusters of source terms based on the accident scenarios.

MEMBER MARTIN: Maybe more so for BWRs?

MR. ALBRIGHT: I would have to go look again at what those clusters looked like, but in this case, with the different shapes of the cumulative distribution functions, yes, it does look like there is a different weighting across the spread here.

MEMBER MARTIN: Okay.

MR. ALBRIGHT: Next slide, please? So, this is the high level overview of this bootstrapping procedure. Again, this was maintained from prior studies and deployed in this study as well.

So, if we think about this, the first step here is we run case simulations. Let's say we run 200 simulations, so that's about the order that we can get here.

So, we have 200 values for the early in-vessel phase duration. And then, we actually generate N samples of that many simulations from the original distribution.

So, let's say we take a 1,000 samples from that distribution of phase durations, right, and then, we actually calculate the percentile for each of these N samples.

So, we calculate the 5th up to the 95th percentile. And now, we have a distribution of 1,000 50th percentiles. The next step is to compute the mean value of that distribution so that it's representative and the standard deviation of each of those percentiles.

So, this goes back to the dots and the dashed lines on the last plot. And then, we actually will interpolate to obtain what the ECDF for that particular quantity of interest looks like.

So, I hope that was a clear example for

folks on what this process would like for an example quantity.

The advantage here is that we can actually look at the variability from the different plants and accident scenarios inside of the representative source term as a single sort of statistical method.

We're not looking at things differently for the different plants or the different scenarios unless we want to break out the blocks of data in that way.

And a general note here is that, with this process, basically, you're maximum and your minimum values are going to be determined by the maximum and minimum source terms that you observe in your data set.

So, we're not trying to extrapolate out beyond the observed data here.

Next slide, please? Okay, so results and discussion. So, I don't know if we need to go over this again. I wasn't sure when our break would be, so I wanted to sort of restate some key aspects of the analysis here.

CHAIR PETTI: So, our break is normally in seven minutes.

MEMBER REMPE: It's up to the subcommittee chairman's discretion.

CHAIR PETTI: Well, I have an agenda, so I thought I had to live to the agenda because I've been --

MEMBER REMPE: It's at your discretion.

CHAIR PETTI: So why don't we take a break now, then? And we'll be back at ten after 3:00.

(Whereupon, the above-entitled matter went off the record at 2:53 p.m. and resumed at 3:09 p.m.)

CHAIR PETTI: Okay, it's time to begin again.

MR. ALBRIGHT: All right, so, now that we're back from our break, I'll restate some of the key aspects of our analysis.

So, again, our objective here was to extend the NUREG-1465 for these higher burnup and HALEU fuels.

And we looked at four accident -- or four, sorry, nuclear power plants. We looked at two BWRs and two PWRs with Mark I and Mark III containment for the Bs and a ice condenser in large dry containment for the Ps.

We had four reactor cores analyzed with varying degrees of burnup enrichment.

And we considered accident scenarios,

including small break LOCAs, large break LOCAs, short-term station blackout, long-term station blackout, and ATWS, or anticipated transient without scram.

The criteria for the different accident phases are listed below. Again, I think we've discussed these in a good deal of detail.

We're going to be focusing on gap release and early vessel release for the remainder of this presentation.

Next slide, please? So, we talked earlier about the impact of the reactor core or the burnup and enrichment on our in containment source term. This is sort of summarizing those results again.

And basically, what we're seeing from our analysis is that the different reactor cores are really not significantly different in terms of the in containment source term.

So, the conclusion here is that the increase in burnup and enrichment does not strongly impact the in containment source term because the accident sequences are driving variability here.

Next slide, please? So, in terms of the BWR in containment source term evolution, this is an extended version of the table that we showed earlier today.

What I want to highlight here is that both SAND2023 and SAND2011 are using MELCOR to predict these results. Whereas NUREG-1465 is using that STCP code which was the forefather of MELCOR so to speak.

We've advanced our understanding of the accident progression through the knowledge advancements, modeling advancements, and input deck advancements over the course of the SOARCA project.

And our results are actually consistent with the findings of the different SOARCA projects.

So, what I'll highlight in this table here are, again, that our in-vessel phase durations for BWRs are quite a bit longer than NUREG-1465.

This is, again, because of the progressive core damage that we -- or the prolonged core damage progression, excuse me, that we see in the MELCOR code due to greater discretization of the core region.

And we're seeing larger releases for several key radionuclide groups, including the halogens, the alkaloids metals tellurium as well as molybdenum.

We've talked about several reasons why these releases are longer -- or larger, excuse me. Some of the primary reasons are the longer duration until lower head failure as well as the early failure of that primary pressure boundary.

One thing I want to highlight before we move on to the next slide is that, again, these results are total inventory in containment.

So, MELCOR is calculating deposition. It's calculating the scrubbing in the suppression pool. It's calculating all of the different mechanisms that will remove or retain different fission products at different locations in the power plant.

And then, we are, in our analysis, taking all of those values and summing them up into our total inventory in containment. So --

MEMBER BIER: Quick question.

MR. ALBRIGHT: Yes?

MEMBER BIER: In terms of things we care about like offsite consequences or whatever, is it important to know the total in-vessel inventory? What does it tell us in the end?

MR. ALBRIGHT: Yes, thank you for that question.

So, the -- in the process of how these numbers are used in regulatory practice, basically, we calculate the total inventories and then, the downstream codes will calculate the transport of those inventories to the environment.

And the short answer is, yes, the processes of retention and removal are very important.

These numbers here are much larger than what you would expect because we have taken those aspects of what MELCOR is calculating out of the numbers that we're reporting, essentially, so that those downstream codes can do what they're designed to do.

MEMBER BIER: Yes, I guess I'm wondering whether having different numbers for in containment inventory is actually meaningful if they're parceled out differently between the different streams like in the old study versus the newer study.

You know, like if certain things are going to be, you know, released faster in one study than another, wouldn't we want to separate those out rather than have them lumped?

MR. CAMPBELL: So, it's kind of -- not kind of, it's a legal requirement like the downstream codes, and so they need -- they have models for some of these processes, but they're simplified and aggregated.

And usually they have conservative models that are chosen.

So, even though this best estimate, sometimes the -- for the purpose of licensing, maybe and our guys can elaborate on that, they need more conservative models for these processes.

MEMBER MARTIN: Maybe related to what Vicki was saying, and probably related to what I might have said to Dave during our break.

I'd say the BWRs seem to get a penalty here with that view. And I know the underlying assumption and the whole mitigating features credit, I can't help but feel that, you know, passive mitigating features.

I mean, we spend a lot of time talking, you know, about the non-LWRs and crediting passive features for everything and we're not looking at our old plants in the same light.

Do you have a feel for if you eliminated the inventory that MELCOR predicts in the pool?

You're already ready, Shawn.

MR. CAMPBELL: I'm ready to answer, I'm so sorry.

First of all, I'll say, wait until my presentation.

MEMBER MARTIN: Okay.

MR. CAMPBELL: I've got a whole

presentation on that and it's in light of some of the peer reviewer comments where they were talking about, you know, there's a lot of retention there in the suppression pool, it seems, and so, can you guys try to explain that and help us to understand what's going on. Right?

So, we have some work that we've done post the 2023 report, and that's what I'm going to be presenting on later.

To try -- and right now, we're not trying to speak of anything in the regulatory sense or what's being done downstream.

We're just trying to better understand, what are the concentrations? What impact does the suppression pool have? Is there places where you may be bypassing the suppression pool and not fully capturing the -- what's available for release if you just let it all go into the suppression pool?

So, that's what I'm going to hope to explain a little bit later.

And I'm hoping that it will also answer a little bit of Vicki's questions as well on understanding, you know, how are these things then being used kind of downstream in these codes as well.

So, we try to explain that as well.

MEMBER MARTIN: I'm anxious to see that. Maybe I'll be quiet up until then.

MR. ALBRIGHT: Thank you.

Next slide, please? So, I mentioned before that the results that we have found in this study are consistent with SOARCA.

And the major result that we want to highlight here is that SOARCA identified limited in-vessel halogen retention during that early in-vessel phase.

And we see it here in this plot. It's a little bit busy with all of the different places that we're seeing the radionuclides.

But essentially, everything that's being captured in that suppression pool, everything that's being released airborne into the drywell, or released into the environment during the early in-vessel phase which is towards the front of this plot, around that, let's see, probably around that six-hour mark, if I remember correctly.

What we're seeing is that you've got an enormous amount of the halogen fraction inventory release to containment. And that's what we've observed in this study and it's because of that that failure of the primary pressure boundary.

Next slide, please? Again, the same type of data we're presenting for the PWRs here, so the main points here are very similar to what we talked about with the BWRs.

We've updated these practices. We started with the 2011 decks. We modernized all of those practices. We implemented the new best practices from SOARCA. And then, we revised any other practices that I mentioned earlier like the time and temperature model, et cetera.

These practices represent an improvement over SOARCA in many cases. And actually, are still consistent with the SOARCA results in terms of the way these releases are occurring and the timing and accident phase.

I do want to highlight that the accident phases are, again, longer than NUREG-1465 and that we're seeing larger releases for the halogens, the alkaloids metals, the tellurium group, and the molybdenum group.

So, we're seeing much the same pattern of changes here and that's because the changes are rooted in those advancements and our understanding and our modeling practices here.

So, we should be seeing the same changes across these two reactor types.

Next slide, please? So, in this slide, again, highlighting the overlap or agreement with SOARCA. For PWRs, SOARCA also found limited halogen release or limited halogen retention, excuse me, during the early in-vessel phase. And this is due to the hot leg creep rupture.

So, different failure mechanism of the primary pressure boundary, but the same result. We open up that pressure boundary and things start to move into containment earlier and in larger quantities than observed previously.

Next slide, please? In terms of the release rate evolution, we talked a little bit about this earlier.

The highlight that I want to make here is that when we assume a uniform release rate, the release rates from the 2023 report are actually lower than NUREG-1465. And this is because of those longer phase durations that we have for our early in-vessel phase.

So, yes, larger in magnitude, but over the duration assuming that uniform distribution, smaller values.

Next slide, please? So, this next group of slides will go over some of our sensitivity cases or sensitivity calculations that were performed as part of this report, very high level. For anyone interested in the further details about some of the different key radionuclide groups, we provide that in the report.

But we've summarized here the main findings. The first sensitivity calculation we'll look at is the fuel thermal conductivity sensitivity. The idea here being that increased burnup will tend to decrease the fuel thermal conductivity.

So, what we did was we ran actually a fuel conductivity sensitivity with a lower fuel conductivity that was informed by the FAST code.

What we found was that there was no impact from variation of the fuel thermal conductivity.

Next slide, please? In the next sensitivity calculation, we looked at the in-vessel particulate debris porosity.

The idea here is that the higher burnups will promote disintegration of the fuel material.

We ran three sensitivity cases here with a reference porosity, an increased porosity, and a decreased porosity to sort of show the spread that might result from different porosities.

What we found here is that there is no major impact from variation of in-vessel particulate debris porosity.

Next slide, please? This third sensitivity calculation was looking at the diameter of the in-vessel particulate debris.

The idea being that higher burnups will promote breakup of the fuel resulting in smaller fuel particulate debris.

So, what we did in this analysis, because of actually, you know, the range of sizes that you could see inside of a core was we ran a high and low case scenario to get an idea of what the spread was.

And what we found was that the variation in particulate debris diameter is impacting that in containment source term, but that those impacts are actually smaller than the variation we're seeing across the scenarios.

And we'll highlight this particular point in just a few slides.

Next slide, please? I think we're on four. Particulate debris following velocity.

In this case, due to the change in size of the particulate debris, you might expect a different velocity of that debris as it falls in the lower plenum to the lower head.

What we found in our analysis, even decreasing that velocity by a significant amount, is that there no impact on source term due to variation in the particulate debris falling velocity.

Next slide, please? In terms of the fuel relocation temperature sensitivity, this is a particularly interesting sensitivity for those of you familiar with the SOARCA uncertainty analyses, this was a primary parameter that was varied in those analyses.

The idea here is that material interactions can cause early failure of fuel assemblies as well as other components.

So, in MELCOR, we have two modeling options for these types of material interactions.

We have the interactive materials model and we have the eutectics model.

The interactive materials model is the practice that was used in previous analyses and was explored through that SOARCA uncertainty analysis.

And the eutectics model is sort of a newer model that's been explored through other uncertainty analyses.

In this calculation, we basically looked at the SOARCA uncertainty analysis distribution of fuel relocation temperatures. And we chose an uncertainty range that spanned the SOARCA evaluation.

So, we've got a low value, a high value, a reference value that I mentioned earlier.

And then, we also ran a case with the eutectics model.

So, the importance of this sensitivity, and the reason I've spent a little bit of time here, is that material interactions are causing different fuel failure timings.

And that's actually impacting the accident progression. That's changing the timing of relocation of fuel and it's actually moving the progression of the accident around to an extent.

And in the SOARCA uncertainty studies, this was found to have an impact on those in containment source terms.

However, this uncertainty is also present for conventional fuels.

For those of you who are familiar with those SOARCA analyses, they were looking at conventional fuels.

And we're actually looking at the same distribution of uncertainty here for these high burnup fuels in this case.

Next slide, please? Oh --

MEMBER REMPE: There's a lot of other events like doesn't relocation also affect hydrogen generation in core? And again, I think this peer review committee emphasized the correlated variables that I just wanted to bring that up.

MR. ALBRIGHT: Yes, no, that's a good point. I'm a little tunnel-visioned in on in containment source terms today. But that's a very good point, that the change to the accident progression that's occurring as part of this fuel relocation, it's more widespread than just the in containment source term for sure. Next slide, please.

So this next model if the fuel rod lifetime model. And I mentioned this briefly earlier. So this is a model that basically we allow fuel components to stand for a certain amount of time at different temperatures, and it accumulates damage over time.

So through the temperature history, we accumulate damage until the fuel fails. This is one of the many fuel failure models that we have in MELCOR. What we did in this analysis was we looked at the reference which was the default time at temperature model, an increased lifetime, reduced lifetime, and then the SOARCA lifetime.

And basically what this is, is it's timing where -- a lifetime model where at lower temperatures your fuel is going to stay in for a really long time, 22 hours in the case of the increased lifetime. And at really high temperatures, around 2,600, we're going to decrease the amount of time that it can stand. And in our most -- in our smallest lifetime case, we only allow the fuel to stand for three minutes if it reached that high temperature.

What you'll notice in the plot is that we're not actually seeing major significant differences in the releases to containment. And this is because the fuel rod lifetime model just doesn't drive fuel rod failure. What we're seeing in our simulations is that the fuel relocation temperature or the failure of oxidized fuel assemblies is generally dominating our fuel failure.

So this is sort of competing models in MELCOR interacting, and in this case, this one didn't have a strong impact on the containment source term. Next slide, please. This analysis shows the hot leg creep rupture. We talked about this a little bit earlier.

This was a key insight from SOARCA that early failure of the hot leg due to this creep rupture will result in larger releases to containment. What we did in our analysis was we actually just disabled the hot leg creep rupture model for our sensitivity calculation. And this allowed the model to actually persist at high pressures for the simulation to persist at high pressures until that vessel failure occurred.

And what we see is that there is a significant -- I think it's on the order of, like, 40 percent or something -- difference in the mass fraction released to the containment when we allow this pressure boundary to remain intact. So this is particularly important in terms of that in containment source term. Next slide, please. I really like this slide. For other folks, I hope you do too.

So this is showing our source term variability from an orange, the sequences, and in purple -- sorry, in orange, the sensitivities and in purple, the sequences. This really highlights sort of the idea that we've been talking about all day today, that the sequences are driving our source term variability, not parametric uncertainties in our models. So what you'll see and the easiest way to explain this is the uncertainty around the purple lines is a larger span than then the uncertainty around the orange lines.

And basically what that means is we've got more variability in outcomes from sequences than we do in terms of individual model sensitivities. Now in relation to this finding or this observation, the peer reviewers did not the potential for the combined effects of the sensitivities. We ran separate effects sensitivity calculations.

The peer reviewers were concerned that if you combined all of these sensitivities, you might see larger differences than what we observed. From our experience with these models and our understanding of these codes, the non-linear processes in our models tend to limit the amplification of any combined sensitivities so that when we combine several sensitivities, we're not going to see additive movement in a single direction. We've got non-linear movement here such that to explore the impact of any of these sensitivities, a single scenario with a single parameter sensitivity is representative of what kind of variation we might observe from that uncertainty.

DR. SCHULTZ: Did you happen to do that?

I mean, did you happen to take a few sensitivities and combine them, see what would happen?

MR. ALBRIGHT: We did not include any combined sensitivities in the final report, no.

DR. SCHULTZ: And based upon what we've seen, I think you're correct. You could test it out pretty quickly. Well, quickly is perhaps an exaggeration. I'm sorry.

MR. LUXAT: From past uncertainty analysis students, certainly for an international uncertainty analysis project, we typically see is that you can collapse to a smaller set of sort of uncertain parameters and still realize, if you will, the same output uncertainty, if you will. There are sometimes a few kind of parameters or output parameters that are very sensitive and important. And those tend to have, if you will, some amplifying effects, and it's usually around hydrogen generation.

But generally what we do is that one way to think about it is you're inducing variability across a very complex network of calculations, calculational steps in a code. So think of it, just a myriad of floating point operations that you're walking down this computational tree. And you put an uncertainty up here or a variability up here and it kind of propagates through this not quite a butterfly effect or a tree falling in Brazil or the Amazon, but in some ways, very similar.

Because of the complexity of these calculations, you wind up with a very small curvation up here, propagating and realizing this vast sort of space of realizations. And to a certain extent, you lose a certain amount of correlation ultimately to that starting variability as you go through this complicated set of calculations. But it's just enough to kind of, if you will, push you down and realize the same sort of space of outcomes and variability. I know that's probably a very abstract statement we always see with these codes, but --

DR. SCHULTZ: It's a good description. A nice thought experiment. Appreciate it.

MR. ALBRIGHT: So before I move on from these sensitivity calculations, I want to take a minute to sort of reemphasize that the point of these sensitivity calculations was to investigate the identified uncertainties that were distinct between high burn-up fuels and conventional fuels. We're not looking at all uncertainties. We're looking at what's special for high burn-up or HALEU fuels.

With that, that's move to the next slide.

So summary of this report, I think these are fairly familiar concepts at this point to everyone. But basically, what we're seeing that increased burn-up and extended enrichment are not significantly impacting the source term based on our analysis and that sequences are the most significant contributor to variability in our data set.

The status of the RPV or of when that lower head failure occurs is basically going to be an important factor in terms of our early in-vessel releases. That low pressures are exhibiting more significant releases to the containment than high pressure that were considered in previous analyses. And finally, that those early in-vessel source terms are greatly reduced if that pressure boundary remains intact.

MEMBER REMPE: Well, I guess at this time I want to ask my question about the status of reactor pressure vessel. Not worried about the pressure within the vessel but vessel failure. And I know we did those tests -- or you did those test out at Sandia years ago.

And we didn't think about things that we're seeing nowadays, Fukushima. But maybe vessel failure isn't at a distinct time. And it's really --

is it really what you're interested in is when you have a large mass of material ex-vessel on a containment floor? And if that's the case, maybe it doesn't happen automatically. And I was wondering if maybe some additional thought is needed in that area.

MR. ALBRIGHT: So I think in terms of the way we model severe accidents and in terms of the current state of practice, lower hedge failure is sort of a discrete bifurcation in terms of what we see in our analysis results. And I think you were alluding to that earlier, right? We see the release of significant quantities of material, debris to the containment.

In terms of what we're seeing from Fukushima, there may be reason to be that failure could occur more progressively than the lower head failure tests at Sandia would suggest where you have sort of a progressive relocation of debris into the containment. In terms of how we split out our accident phases, I think that's probably at the end of the day we can do different things with our calculations. But in terms of how it feeds into a regulatory source term, I would have to defer to my NRC colleagues to give me direction on what they need for my calculations.

MR. CAMPBELL: Yeah, just going back to what we said earlier that we're already going out to seven hours here. And you'd have to have some sort of start and stopping point. And so there's a bifurcation of -- at that point of vessel failure. And so we're trying to find a stopping point. When else would we end, I guess.

MEMBER REMPE: So the question is what you're going to do with it. And then should it be a discrete point is what I'm asking. And maybe, again, if you're looking at the effectiveness of the ECCS systems, I'm not sure that a discrete point is -- this is a larger question.

MR. CAMPBELL: It's a larger question that is --

(Simultaneous speaking.)

MEMBER REMPE: -- I just think of something that ought to be thought about because as you get more information -- I mean, the Sandia test didn't have fuel assemblies drop out of the -- and they were -- again, that's what everybody wanted.

We used to worry about a penetration versus global vessel failure and what does it mean. And just again, now it's so important. At that time is when we went from, like, one and half hours to eight hours and then six point whatever hours. It seems like it's going all over the board. And maybe it's not all at once.

MR. SALAY: This is Mike Salay. We didn't even consider trying to redefine that because that's another thing that we'd have to defend. And so we stuck with the definition which is a point that you can get, that they can -- that NRR and the regulators can use.

MEMBER REMPE: But maybe the regulators ought to -- is it a good point? Yes. Anyway, I've made the question a long one.

MR. ALBRIGHT: Next slide, please. So now we come to the end of a peer review. What I'll give here is an overview of this peer review process and some of the main findings, definitely direct people to the peer review report which is highlighted here on the left for any further details they may be interested in. The idea behind this external peer review was to review the technical basis of the SAND2023 document that we just finished covering today.

Recommend improvements to the draft form of the document prior to its publication. And then any additional recommendations after its publication as well as assess the suitability of the source terms for the regulatory applications that were intended. Next slide, please. The overview of the organization of this panel, there were, let's see, six panel members here from different organizations.

We had Dr. Mohsen Khatib-Rahbar, excuse me, from ERI. We had Dr. Richard Denning from -- he was a consultant. We had Mr. Jeff Gabor from Jensen Hughes, Dr. Didier Jacquemain from the OECD NEA, Dr. Luis Herranze from CIEMAT, and Yu Maruyama from JAEA.

So an international group of severe accident experts who are reviewing this document. Their objectives were to assess those qualities of an alternative source term that we mentioned earlier today. So what was the technical adequacy of the approach and the specific applications of the MELCOR code to developing these source terms?

How appropriate were the sequences selected? How the assumptions and applied models line up with our current understanding of severe accidents and source terms, and how adequate are these approaches given current experimental and other existing data sets. Finally, to assess whether our source terms were representative rather than conservative and bounding and then to basically make a conclusion on the overall technical basis of the approach. So next slide, please.

In terms of this review process, high level, we prepared a draft high burnup fuel source term that was completed in 2021. And then we had a group of virtual meetings that began in 2022. The first meeting was a briefing of the draft report that was followed up by discussion and essentially comments from the peer review committee.

During the second meeting, we provided initial responses to the peer review committee's comments, resolving some comments but not all of them. So we revised the report. And during the third meeting, we provided them with a final report that had responses to all of their different comments which are detailed in that peer review report for anyone who's interesting in sort of following this track through of comment and discussion. Next slide, please.

In terms of the acceptability of our source term, I've taken some direct quotes here that the peer review panel endorsed the approach. They've stated that we've provided a defendable technical basis for our source terms. We reasonably represent the U.S. nuclear fleet, and we have a spectrum of accidents that is sufficient to satisfy the RG 1.183

requirements. Next slide, please.

In terms of some qualities of our source term, the peer review committee had a number of comments that I thought relevant to bring to our attention today. The first was the study's significant technical improvement using state of the art methods implemented in MELCOR. In containment source terms for high burnup in HALEU fuels are representative MELCOR estimates rather than conservative and bounding estimates.

The peer review committee did not identify any biases that would overestimate in containment source terms in our analysis. And the sensitivity studies were valuable in supporting this application, particularly looking at the impact of the depressurization of that primary pressure boundary or the failure of the primary pressure boundary, excuse me, so the HALEU pre-pressure sensitivity analysis. Next slide, please.

In terms of recommendations, the peer review committee had two major recommendations that I have on this slide. The first is that the gap release phase be incorporated into the early in-vessel phase. I mentioned earlier that the advancements in how we model severe accidents has led to the loss of distinct gap and early in-vessel phases.

We now have sort of overlap between those two phases depending on where we are in the core. The second point that they made is that it would be more appropriate to represent the impact of burn-up on core inventories through expression of radiological activities. And we have some details on this particular note in the next presentation where we look at this one in more detail. Next slide, please.

This list here is basically the compilation of some of the peer review comments that we've made throughout previous slides here. The first one here is basically that we didn't consider bypass or (audio interference) regression scenarios in the development of these tabular source terms and that we did not include any fission product removal and retention mechanisms in the final reported results. Again, MELCOR is capturing these, but we are providing source terms in terms of the total inventory.

We talked about how the peer reviewers acknowledged more recent PRAs that might give us a different distribution of core damage contributors but that the current analysis practice was suitable for its intended purpose. There was the important note that for most radionuclides there is not increase in activity with burn-up such that it would impact siting calculations. Then we had comments on the uncertainty in iodine speciation that we touched on earlier.

And then the last two bullets here are the confirmation of the assumption that cesium molybdate is the primary chemical form or dominate chemical form of cesium coming out of Fukushima again. And finally, that the use of the median estimate is appropriate for avoiding bias in our final results here. Next slide, please. This last comment from the peer review committee sort of is driving some of our follow-up calculations.

So I just wanted to put it up here and quickly summarize it. The tabular source terms provide a simplified tool for regulatory applications but that there are limits to how these tabular source terms can be used and the information that can be provided in them. And the peer review committee encouraged direct application of state of the art severe accident codes like MELCOR to specific issues when appropriate.

The issue that we will be talking about in the next slide and the next presentation with Shawn is the suppression pool scrubbing or the impact of suppression pool and the radionuclide concentrations in the steam line. Next slide, please. So this table, I don't want to spend too much time on the details because Shawn is going to go into this in much greater detail that I could in a single slide here. But what we're looking at are for the gap release phase and the early in-vessel phase, the total inventory including the suppression pool, and the total inventory excluding the suppression pool.

So what you'll notice is that when we pull the suppression -- the radionuclide inventory that's in the suppression pool out so the right column for each of these accident phases, the source term decreases significantly. So the main point here is that the suppression pool like some of you have already mentioned today has a significant effect in terms of radionuclide retention for key radionuclide groups. It is basically immobilizing some of these fission products while they are retained in the suppression pool.

Two peer review findings are particularly important for us to note here. The first -- and these are quoted -- I think they're direct quotes, not summaries. But the in-containment source term should consider the impact of retention in the suppression pools, especially for SBO scenarios that discharge directly into the suppression pool.

And estimates of retention and suppression pools provided in SAND2023 could be used in regulatory guidance to establish suppression pool decontamination factors. So this is something that we're still looking at. And we've done some follow-on calculations again that Shawn will be looking at in the next presentation which I think we'll get to in just a few minutes. Next slide, please.

The next two slides are going to be very quick on upcoming work. The first is the chromium coated ATF concept. We have been working on an chromium coated accident tolerant fuel concept source term that follows the same practices that we've outlined for you all today looking a chromium coated fuels.

And this analysis is also being informed by that ATF part that we mentioned earlier today. Next slide, please. We are also currently working on a source term report for FeCrAl fuels. Again, this analysis is being informed by that ATF severe accident part. And these are following the same practices that we've outlined today.

CHAIR PETTI: What's your timeline for those?

MR. CAMPBELL: That's for me. We're very close to these. So I'd say in the next couple of months we're planning on having a draft complete of these. And then we'll have another period of time of just internal review and discussion. But in the next coming months, we plan to have these finished.

CHAIR PETTI: We might be interested.

MR. ALBRIGHT: Next slide, please. Thank you.

CHAIR PETTI: Okay. Watch the time. Let's keep going.

MR. CAMPBELL: Okay. So that's the end of that presentation. And if we don't mind switching to the other presentation, I can get started on mine. All right. So hello, my name is Shawn Campbell, and I'm in the Office of Research.

So I want to provide you all with some follow-on work that we've been doing in light of some of those latter peer review comments that we were talking about. And I think Sandia has done a great job teeing this up a little bit. So this is follow-on work that we've been doing, just to try to explore the impact of suppression pool and its retention. So next slide, please.

So overall background here, like I said, the peer review panel has commented on potential impact that a suppression pool could have on what's getting into containment obviously. And that table, it's Table 516 if anyone is interested. It's in the SAND2023 report.

So it's there, and it provides the containment release fractions both including and then excluding the suppression pool. So we did some supplemental investigations following those peer review comments to try to investigate the impact of the suppression pool. And in particular, try to look at scenarios and pathways that might bypass the suppression pool because that's really what's important here, right?

So to that end, we modified the two BWR input decks, Peach Bottom and Grand Gulf, try to better capture the behavior that could be going on, particularly in the steam line. And then we performed a set of BWR source term calculations. So it's the same scenarios that we did with 2023 that were performed for this analysis. So next slide, please.

So before I go forward, there's been a lot of confusion about what a source term versus inventories versus all that. So I thought that this could be kind of useful to make sure that we're all on the same page. We all understand what is a containment source term and it's being used.

This becomes really important later on in my presentation. So up here at the top is kind of a representative MELCOR calculation if you will where, for example, if you're trying to do a SOARCA analysis or a Fukushima analysis, we would use MELCOR to do the full accident scenario. This means cladding oxidation, relocation, transport of the fission products out of the core, into the reactor vessel, out into containment, containment failure and vessel breach.

And then after vessel breach, you've got MCCI and over-pressure and so on. So MELCOR is calculating all of these things, right? And so not to say that when we talk about the source term, MELCOR is calculating all aspects of retention and deposition and everything.

It's just what is then being reported as the containment source term. So the containment source term then is the cumulative amount of fission products that enter the containment during these phases. So in the gap phase, early and vessel and so on.

It's the cumulative amount of fission

products that have entered into that volume. And that's what we report down here in the green box. That's our containment source term.

And it's a fraction of the overall inventory in the core that has made its way into the containment. This is a fraction, not -- because MELCOR deals in overall mass, not inactivities. So in order to get to activities as you were saying, what we really care about here is the dose, right?

Or at least from a regulatory perspective, what we care about is the dose in the end, right? And so how that's done downstream in the regulatory space then is that containment source term is then tried to make by an applicant or whoever, make it more reactor specific, right, because this is a representative source term that we have here in the green that's supposed to be representative of the fleet. And so to make it more reactor specific, we want the concentration of that source term.

So that's where you would divide by the volume of your containment, your individual containment, to get your concentration of the source term into that containment structure. So it's a simple ST divided by volume here. That concentration is then used in this simplified approach using removal mechanisms, whether that may be sprays or a natural deposition or whatever that might be.

In a simplified approach, those removal mechanisms are accounted for here. And then you have a leak rate. And then this is when your inventory or your activity comes into play, right? And so now we're seeing how much activity goes from containment and now into your people space. I'll pause there and see if there's any questions or needed additional clarifications.

MEMBER REMPE: Jose always says that he has an evil mind to try and think of something again.

(Laughter.)

MEMBER REMPE: -- and go against the system. Had a vessel that purposely would fail early.

And that's what I was saying really if the vessel would fail earlier. And then you got a smaller source term. And so to try and avoid that would be a good motivator to rethink the question I asked earlier. Again, because there are a lot of design developers and it's a way to make sure things --

MR. CAMPBELL: I'll state the obvious. Our aim is not to game the system.

MEMBER REMPE: It is --

(Simultaneous speaking.)

MEMBER REMPE: But somebody else's would be.

MR. CAMPBELL: Right. But that's why we provide the source term or that's written into the Reg Guide is that we provide a source term that we think is representative, that's peer reviewed, that we think is defensible. And then that's what's used then by in the regulatory space.

MEMBER REMPE: It's just a reason to think about.

MR. CAMPBELL: Right. Okay. Any other questions?

MEMBER ROBERTS: Yeah, just for clarification. When you talked about suppression pool scrubbing, I don't see that on this page. It's not on the removal mechanisms, or --

MR. CAMPBELL: It is not part of the removal mechanisms.

MEMBER ROBERTS: Okay. So if you don't account for it, then --

MR. CAMPBELL: It is not accounted for in this bottom part. And then up on the top, obviously MELCOR is calculating it. But it's part of that whole bookkeeping. It still gets lumped into that green space and brought down.

MEMBER ROBERTS: Right. So it could be part of that mechanism modeling. Mechanism modeling, if you chose to do it that --

(Simultaneous speaking.)

MR. CAMPBELL: And that's -- yes and no. And that's one of the things I want to kind of tease out here because one of the concerns is, is there anything that's bypassing. And that's one of the aspects that you have to kind of tease out in this whole aspect.

So you don't want to just simply say, oh, let's just scrub and take 80 percent, 95 percent of it is just gone. Are there pathways -- release pathways to the people space. Let me bypass this whole thing. And so that's what we're trying to investigate here.

MEMBER ROBERTS: Okay, thanks.

MR. CAMPBELL: Yeah. All right. Next slide, please. So this is just trying to show where we've tried to refine our models. So over here on the left is a representative -- this is Peach Bottom. So this is our original from 2023.

This is the nodalization that we have within MELCOR. So we wanted more refined nodalization of the steam line in particular because this is what we're talking about when I'm saying particularly through the MSIVs we're trying to understand what's going on up there, what's happening within the steam line, what's the concentration within the steam line. And so we refined our modeling.

And I'm afraid I can't zoom in here. If I had my mouse, I would. But in this purple space here, you can see that top bar. It's two volumes. You can see two boxes.

That was our steam line A. Thank you very much. That's our steam line A. And so that's the steam line that has the lowest pressure SRV. And so that had a slightly more -- thank you so much.

That has a slightly more refined modeling, I guess, with two CVs. But then everything else was lumped together into a single CV, all three of the other lines. So we wanted to break this out a little bit so we could better capture the physics of what's going on in the steam line.

So this is where we have our more refined modeling. And the reason we need more refined modeling if you look on the right-hand side over here, you can see it's a rather long -- the steam line is rather long. It's got bends and turns and everything, not to mention the SRV cycling on it, HPCI, RCIC pulling off and so on.

So we wanted to be able to better capture what's going on if we're going to ask questions about what is the fraction of inventory that's making its way into the steam line. So next slide, please. Thank you. So here's our more refined steam line modeling. So for each of the BWRs, each steam line has now been broken up into this nodalization.

So the first volume that I have here in blue, this is going from the steam dome down to and including the first SRV. And in our model, that first SRV is the lowest pressure SRV. So that's the one that's going to be cycling.

And then the next volume is this green volume. This is everything downstream of that first SRV up to the first MSIV. RCIC and HPCI pull off of this line depending on the steam line obviously.

And so we have a much finer nodalization. We also have as a separate volume then in between the MSIVs downstream to the MSIV down to the stop valve and condenser. So all of that has much more refined modeling. And then so now going forward --

MEMBER HALNON: Is the only vent to the environment the condenser because you have so many others?

MR. CAMPBELL: We do. And we're not going out to that level of detail to try to model everything that can get to the environment. That's really don't by those downstream codes, right? We're not trying to capture -- you might have leakage through the valve stems and so on. We're not trying to capture overall --

(Simultaneous speaking.)

MEMBER HALNON: That's usually a significant loss, megawatts.

MR. CAMPBELL: Sure.

MEMBER HALNON: So that we -- we're always chasing. It could be one -- three percent. Getting close though.

MR. CAMPBELL: Sure.

MEMBER HALNON: Quite a bit.

MR. CAMPBELL: Yeah, and as I'm about to say, we kind of stopped our investigation at the first MSIV. And so I'll explain why here in a moment. But most of our investigation is here in this green portion.

So in the rest of my presentation, I'm going to be talking about source term fraction kind of in the same way that we talk about our containment source term. I'm going to be talking about source term fraction in the steam line because one release pathway for BWRs obviously is the MSIVs, right? And this is a release path that has the potential for bypassing the suppression pool.

And that's why we thought that it was important to better characterize and understand what's going on in the steam line. So as we go forward, a few things that I need to talk about here about this green portion. First of all, when I report source term in the future, I'm reporting what's in the green portion.

We thought this was representative kind of in the same way that the containment source term is what's available for release through a leakage pathway in containment. This is kind of what's available for release through an MSIV. Also distinct from how the source term is reported in containment as a cumulative release into containment, that's harder to do here in this green portion because you don't have fission products entering into this volume and then staying there indefinitely due to cycling of the SRV, through RCIC and HPCI operation, through leakage through the MSIV.

This is a dynamic volume. It's not quite as dynamic obviously as that blue portion. But this is a dynamic volume. And so you're going to have a lot of in and out.

So we needed a better way to come up with a value for source term. And what we came up with is a time averaged source term. And I'll try to flush that out a little bit more as we go.

But we talked about what we're going to capture here is a time averaged airborne fission product source term. And so emphasis on airborne because we're already taking into account all of those deposition mechanisms. This is another thing that's distinct from how we report in containment where those downstream codes then look at whatever might deposit through sprays or through whatever.

We've already taken that into account here in this green portion in order to try to capture all of that physics, the ins and the outs. We're reporting it all as an overall fraction of airborne time averaged within that phase. So I'll pause there. That was a lot.

Okay. Well, then I'll move on from there. And if we need to cycle back to this, I can. So this is what we're reporting here. We've seen most of these values already many times.

You've got your Reg Guide 1.183, Rev. 0, Rev. 1, and then 2023 values that my colleagues have been talking a lot about. And then these next two, I need to point out. These are just the Table 5-16 values.

These are not doing a bunch of additional calculations. This is a bookkeeping exercise here of what is or is not in the suppression pool. The only thing that's truly unique on this slide versus what we did in the 2023 report is in the last column, and this is this fraction that is sitting in that green portion of the steam line.

Now I want to point out here this is fraction. So you may look at that and say it's incredibly small, e to the -5. But you have to remember this is a different volume.

Concentration is what's really important here, not fraction. And so what we need to really do is look at what's the overall concentration comparison between what's in containment versus what may be sitting in the steam line. And that's why the next slide is going to be really important where now we have to look at an example and divide by the volume, then do a concentration comparison between what's in containment versus what's in the steam line.

And that's why the next slide is going to be really important where now we have to look at an example and divide by the volume, then do a concentration comparison between what's in containment versus what's in the steam line. So that's what I want to do in the next slide if you don't mind going to the next slide. So here we are.

We got Grand Gulf on the left, Peach Bottom on the right. All I've done here is taken those values from the previous slide and divided by the containment volume. So this is fraction of core inventory per meter cubed. That's the units here.

And so you can see 1465 and 2011 and 2023 all listed here. And then off here in my brackets, this is what's made it into containment minus what's in the suppression pool. That's what's in that first column of the brackets and then what's in the steam line.

So all this is, is the SAND2023 values, the first column. All it is, is the SAND2023 taking away the suppression. That's all I've done here. But then on the steam line side, I just need to point out a few things.

Once again, you may look at this and think that steam line containment are comparable. But be careful. Remember these aren't apples and apples source terms, right? The one on the right has already taken into account any deposition mechanisms within that phase. So this is the concentration throughout that phase, if that makes sense.

MEMBER BIER: So I'm trying to understand the whole picture. So the third set of bars is the 2023 results --

MR. CAMPBELL: Yes.

MEMBER BIER: -- that these guys just talked about.

MR. CAMPBELL: Correct.

MEMBER BIER: And you parcel it out further and get those bars to the right?

MR. CAMPBELL: That's right.

MEMBER BIER: Do you have a figure that does that to the 2011 numbers, or --

MR. CAMPBELL: This has 2011 numbers on there as well.

MEMBER BIER: No, but that parcel out process on the right.

MR. CAMPBELL: Oh, no, no. I'm actually separating what's in the suppression pool versus -- no, we don't have those values. I would assume it's going to be kind of analogous here.

MEMBER BIER: Got it. That's what I wanted.

MR. CAMPBELL: It's a fair assumption. I don't have that in front of me.

MEMBER BIER: Okay.

MR. CAMPBELL: A couple things I'll point out. Size of containment matters obviously. Grand Gulf is a much larger containment. So your concentration is diluted here.

Also, I'll point out that in the steam line, you notice it really doesn't change too much between the two plants. And that's really a consequence of your steam lines aren't all that different between the plants. It's a relative -- the steam line is relatively similar between these two plants. All right. Yes.

So then here the only purpose of this slide was trying to do a comparison of BWR to PWR then. So I'm not talking about the steam line in this slide. So the purpose here is just to say how does this compare to what's going on with Ps.

And so this is representative volumes for a couple Mark I, Mark II, Mark III and then for PWRs and ice condensers, sub-atmospheric and a large dry. So I've just divided by some volumes here. And you can see that time progression essentially from 1465 to 2011 to 2023, and then if you don't account for what's in the suppression pool.

And you'll see that at least for the halogens, then it's more similar to what you're seeing in a PWR. All right. So the purpose of this one also was -- and this was a recent addition. The purpose of this one really was to find some kind of representative example fission product activities for a few case studies that we've done for high burnup and high enrichment PWRs and BWRs.

So we're trying to calculate end of cycle activities for that inventory, a piece of this, right? Remember we're going back to that first slide. Inventory matters in activities.

So that's what we're trying to get at here versus the kilograms that we've been talking about in the 2023 report. We're trying to better understand what would it be in terms of activity for some of these reactors going up to higher burn-ups. So here we have for BWRs and PWRs, we have a reference case on the first column.

So we've got a core average end of cycle burnup for the reference case for a B of 36.2. We have an average assembly discharge burnup of 52.6 with an enrichment of 4.45. And then we're going to a more higher burnup here.

And we're estimating about 58 average assembly discharge burnup. And this is going from an enrichment of 4.45 to 5.3. So we're trying to come to something that's, like, a representative loading pattern for a BWR that might want to go to a higher burnup and tease out what kind of values they could get.

What you see here, I find it kind of interesting is that the iodine concentration trends with the power. And since the power hasn't increased here, your iodine concentration hasn't increased. Now the alkalis, however, that increases with your burnup.

So you do see a proportional increase of your cesium going to these higher burnups. And this is the same for Bs and for Ps. And then same thing for the tellurium.

There's really a weak dependency on burnup enrichment for tellurium as well. Same thing with the Ps. We went to a higher cycling for the Ps and had a higher enrichment and higher burnup there as well.

So you're seeing that we're not getting -- maybe it's a little deceptive when you're looking at some of the 2023 values and you're seeing a 40 percent increase in cesium and iodine mass. That's not what we're talking about as far as activity. And we're not trying to say that this is the activity that you're going to see from a licensee.

We're just saying this is possibly a little bit more representative of something going to a higher burnup. Okay. Next slide, please. So conclusions, we did some refined modeling, and it provided some better estimates of fission product distribution in the steam line and then in containment. And we found that concentration in the steam line is distinct from that of containment when you're not looking at the suppression pool.

MEMBER HALNON: On the steam line, I guess I'm still befuddled a little bit.

MR. CAMPBELL: Please.

MEMBER HALNON: And you're probably just going to say that's somebody else's job. But we've been looking at license amendments that are trying to take credit for the downstream sections of piping or scrubbing --

MR. CAMPBELL: Yeah.

MEMBER HALNON: -- and other items. How is that factoring in? Or is that somebody else's issue?

MR. CAMPBELL: The answer is it is a regulatory issue of how it's going to be applied.

That being said, one of the things we were trying tease out here is can you apply -- could you apply the same source term that's in containment sans suppression pool? Would you apply that same source term then to MSIV leakage? And I think that's what I'm trying to tease out here is that wouldn't be quite right. You really need to have a -- there is a distinct and distinctly higher concentration in the steam line than there is in containment minus suppression.

MEMBER HALNON: Even if you add in all that extra piping just because --

MR. CAMPBELL: And that's the thing, if you look at all the downstream effects. But that's typically teased out by the individual applicant or licensees, right, because they're using their downstream codes to look at all of that.

MEMBER HALNON: This is the verifying, validating, whatever you want to call it, confirmed starting point. If they can justify some other type of scrubbing suppression pool or downstream, then they might be able to bring it back to a point where their design works.

MR. CAMPBELL: Right. And that's what we're trying to tease out here.

MEMBER HALNON: Okay.

CHAIR PETTI: So longer term, bigger picture is this stuff will inform and update to 1.183 on MSIV?

MR. CAMPBELL: I would have to defer to my NRR colleagues on that. Our purpose here first of all was just to be responsive to the peer review, right? That was the purpose here was to say the peer review said to go look at this. Let's be responsive to that because they're saying that this suppression pool is going to have an impact.

But we wanted to make sure that there has been talk of suppression pool, suppression pool. We wanted to make sure that we teased out can you just apply the suppression pool or just do a DF completely off of that same containment source term without looking up the steam line separately. And hopefully we've teased out that, no, you do need to look at that steam line as a separate entity.

MR. DICKSON: Hey Shawn?

MR. CAMPBELL: Yeah.

MR. DICKSON: Elijah Dickson with the staff. I'd like to connect some of the other, like, regulatory initiatives that are going on right now. So just a few weeks ago, I spoke in regards to the increased enrichment. And we're looking at the control and design criteria in that case.

And part of those alternatives is looking at these mechanistic transport models in support of development of Regulatory Guidance 1.183, Rev. 2. So we are considering all of this based off of the current work, based off comments received from ACRS, individuals in industry. So it's all part of that work right now. And we're still in the process of collecting public comments. Again, I just wanted to kind of connect the dots in regards to other regulatory initiatives that are being done right now.

MR. CAMPBELL: Great.

MR. DICKSON: Yeah, no problem.

MR. KORTGE: Can I ask a question?

MR. CAMPBELL: Yeah, please.

CHAIR PETTI: Who is --

MR. KORTGE: What is the mode of force for the suppression pool?

CHAIR PETTI: Sorry, who's speaking?

MR. KORTGE: This is David Kortge.

CHAIR PETTI: From? Organization?

MR. KORTGE: Constellation, Safety Analysis.

CHAIR PETTI: Yeah, no, you can't ask.

Only during public comments.

MR. KORTGE: Apologies.

MEMBER BROWN: As an uninitiated, may I ask a question?

CHAIR PETTI: Maybe.

MEMBER BROWN: In excruciating detail. I'm trying to figure out what the bottom line is. It's wonderful analysis. It's not answering my question.

MR. CAMPBELL: Please.

MEMBER BROWN: If I read this and did you analysis, all the green pipes are longer than the other pipes which means to me you've got higher source terms you have to deal with under a severe accident. Is that going to affect now the EPZs or the zones that we have to deal with? Is that -- I mean, they're bigger, a lot bigger.

MR. CAMPBELL: Which one is? Can we go up one slide, please?

MEMBER BROWN: I'm looking at the one with the bars, the little green, red, and gold, previous slide.

MR. CAMPBELL: Previous, let's go up one. Up another one. This one right here?

MEMBER BROWN: Yeah, that's the first

question. Does that affect --

MR. CAMPBELL: So I'll say that the orange right there is Reg Guide 1.183, Rev 1 basis. Is that correct? So the 2011 values there --

MEMBER BROWN: Yeah, with the little --

MR. CAMPBELL: -- in the orange?

MEMBER BROWN: Yeah.

MR. CAMPBELL: So then that is used by the downstream codes as the regulatory basis.

MEMBER BROWN: So now it's going to get bigger.

MR. CAMPBELL: If 2023 was the basis for a reg guide, then --

MEMBER BROWN: But you would argue -- I'm trying to be contrary a little bit here. But you would argue that you've proved that the basis -- you've had peer review of the basis, an impressive list of people with qualifications to do that review which would seem to indicate that there's a significant increase in what the source term you have to deal with for computing EPZs which would apply to present day plants. Now that's the takeaway that somebody that's not -- I mean, don't ask me to do the calculations. Okay? My mind was exploding while you were doing that. The second question I would have --

the answer is yes, if you adopt it.

MR. CAMPBELL: That's where I would have to defer to my friend at NRR.

MEMBER BROWN: I understand you're deferring. That's fine. The second is this is high burnup. People want to go to high burnup. Changes your refueling, all that good stuff.

But then once you've taken the fuel out, you now have a spent fuel. This would imply that you have a bigger -- something you have to deal with in the spent fuel pools as well, as well as in storage casks once they've cooled down and you've got them in storage casks. Does that compute?

You've got more source term you start with. You've got more residual. It's not the instantaneous stuff. But I'm trying to come to a conclusion. How does this affect other things? Like, now do I have to have different casks? I haven't heard any of that in any of the previous high burnup fuel discussions.

MR. CAMPBELL: Go back to the point of Lucas's source term. But it's not the high burnup but actually the change in how you're doing sequences and going from high pressure to low pressure that causes that increase primarily, not the increase burnup.

MR. CAMPBELL: In other words, we don't anticipate there being a significant change going to high burnup in the overall activity.

(Simultaneous speaking.)

MEMBER BIER: But there is a significant change due to the analysis.

MR. CAMPBELL: In our source term. In our source term.

(Simultaneous speaking.)

MEMBER BROWN: Existing low burnup then would have the same results. And so that you're still impacted in terms of the EPZs and/or (audio interference). So if it's not a result of the burnup, you just figured out that you didn't have the right answers before for EPZ and for --

MR. CAMPBELL: Elijah is jumping at the bit.

CHAIR PETTI: This is a research presentation, then there is a licensing part.

MEMBER BROWN: I understand, okay.

CHAIR PETTI: This doesn't affect any of the existing plans.

MR. CAMPBELL: That's right.

MEMBER BROWN: Well, you've got results that has to be evaluated at some point that says, the analysis that was used is more comprehensive, has a higher result, and it's independent. That's what he just said, whether you use high burnup fuel or the regular fuel we've got now which implies to me that somebody is going to have to sit down with you all, thrash out another -- you have to go address this in the existing plans in terms of an analysis of their occupations at speakeasies and our spent fuel pools and/or past storage and/or transportation of those casks.

More shielding is -- whatever. I'm just a poor electrical guy. I do understand having numbers come out significantly different regardless of how you got there.

MR. LUXAT: So let me just quickly make a comment that from the peer review, again, the method, they appreciated the advance in the methods. But one of the important comments from the peer review panel was an RD contamination measures related to the suppression pool that were not being --

(Simultaneous speaking.)

MR. LUXAT: -- for BWRs that were not directly being addressed. And what Shawn has been talking about is if we were to look at the effect of those decontamination or those removable mechanisms in the suppression pool, A, it would have an impact on what the containment source term is. But it would also importantly have an effect on what the concentration of fission products are nearby other release pathways like the MSIV.

And that's what this is teasing out is that the key comment from the peer review was there's an important removal mechanism, a passive removal or a inherent removal mechanism for BWRs. And what we've been doing is we've been looking to expand the technical basis and understand better the transport of fission products out of the reactor system and how it potentially -- how they could potentially be removed by the suppression pool, what that effect would be on the containment source term. But also importantly, what the effect of that removal could be on other release pathways that are not, if you will, interfacing directly with the suppression pool.

MEMBER BROWN: There's other mechanisms that remove some of this. That's fine under --

(Simultaneous speaking.)

MR. LUXAT: That was -- yeah, but that was --

MEMBER BROWN: -- standpoint. But what about the non-severe accident standpoint? I have to deal with removal. And you're still telling me that I've got more stuff that I have to deal with that's got source term in it when you pull it out regardless if you have a severe accident or not.

That's the other takeaway I -- that's the uninitiated, bottom line, takeaway for somebody like me. And yet somewhere -- forget the severe accident. I understand a PWR is a PWR.

They have other ways of reducing the activity so it doesn't get spread all over the place. But I still have to deal with regular fuel, regular high burn-up, put it into the spent fuel, then put it in a cask when it gets heated. And now I've got larger source terms that I have to deal with in those circumstances.

MR. DICKSON: This is Elijah Dickson. With an increased enrichment, the rule making efforts, they are looking at transportation and several other rules as well. In regards to EPZ sizing, there are several regulatory source terms that we use for different purposes.

So this is an in containment source term used to size mitigative systems to reduce radioactive release to the environment and inside a facility, certain distance away from a population center. For

EPZ sizing, they do use this in containment source term as one of the analysis that went into the Ten Mile EPZ rule, right? They also used source terms that were derived from the PRAs, WASH-1400 as well.

And then also looked at different figures of merit at that time to justify that Ten Mile EPZ. So between the work that's been done in SOARCA with the severe accident work and the more realistic consequence analyses and this in containment source term used as a design tool, right, to size safety-related mitigative equipment. There's no a whole lot of difference that we're seeing between now and what was done back in the late '60s when we did the EPZ sizing.

MEMBER BROWN: If you take into these other considerations and other things. I'm trying to transition away from the severe accident. We haven't had, quote, severe accidents.

MR. DICKSON: Three Mile Island was.

MEMBER BROWN: That was not as severe as a real severe accident.

MR. DICKSON: Right.

MEMBER BROWN: That was like spitting in the ocean.

MR. DICKSON: Right.

MEMBER BROWN: And a lot of dumb stuff. I mean, my point being is what this tells me in your results regardless you've got analyses, methodologies in terms of routine operations. You still built up a different spectrum of stuff that you had --

MR. DICKSON: Understood.

MEMBER BROWN: -- to deal with on a routine refueling storage, then cask storage, and then transportation issues aside from this severe accident. That's all. I'm just trying to make sure that thought process is in place as well or should be in place.

MR. DICKSON: It is. It is. We're thinking a lot about --

MEMBER BROWN: There's --

MR. DICKSON: -- all of this.

MEMBER BROWN: -- routine operations, no accidents. We haven't been getting the right answers. If everybody accepts this as proper analyses --

MR. DICKSON: Right.

MEMBER BROWN: -- to get the right results. Thank you. I think I've exhausted my brain power.

MEMBER BIER: I have one more question that's kind of a follow-up. Charlie, you were saying that, okay, there are other mitigations that are not reflected in the results here. Would those be reflected in the results of a Level 3 PRA? Or are Level 3s overestimated because they're not modeling all of the (audio interference)?

MR. DICKSON: So I'm a practitioner of this source term. I utilize this source term. These are the experts that develop the source term, right?

Again, this is Elijah Dickson with the staff. The way Reg Guide 1.183 is set up, Rev. 1 has these tables of fractions of the reactor core source term. In the appendices, Appendix A specifically is the MHA LOCA appendix that tells you how to transport this source term through all the different systems, how to credit different type of removal mechanisms. And that's where Shawn has been discussing how we can make improvements in these particular areas, these mechanistic transport models, in the appendices specifically understood.

MR. CAMPBELL: Can we INSPECTOR BOTH: back to that final slide? No, that was it. That was it.

MEMBER BROWN: Sorry to disrupt.

MR. CAMPBELL: No, thank you, no. So I think the second bullet here is pretty obvious. There's a lot of retention in the suppression pool.

Third bullet, preliminary investigation of the fission product inventory showed limited effect for high burnup and high enriched uranium. So this was that whole inventory aspect that you have marginal increase in your actual activity. So inventory matters.

And then finally, this is something that we're investigating right now, we're looking into. There's potential to apply MELCOR to better inform some of those removal mechanisms that I was talking about in that first slide, those lambdas and to inform those for the simplified tools. And I just wanted to point out here we're in the process of drafting a document that summarizes this work that we've been doing on investigating concentrations within the steam line. So this is preliminary in that we're looking for early feedback from ACRS on our process and so on as we're drafting this report.

MEMBER BROWN: Well, don't take my questions as being negative or critical. It's nice to see we're not sitting on our past accomplishments, but yet we're trying to make sure we're doing it right for both today and the future. So don't take my comments as being majority that are critical because it's good to see somebody doing good research that comes up with some results we can deal with, at least reduce it to the understanding of the common man.

MR. CAMPBELL: We take all feedback as constructive.

MEMBER REMPE: Okay. What are you going to do about the existing source term and the differences between? I mean, you've got this new model of higher release fractions because of the way that the sequences modeled. And can't do back fits probably. And so what will you do with this --

MR. DICKSON: We're still in the review process. We're just now kicking off increased enrichment. And with that is the work in developing Reg Guide, Rev. 2.

MEMBER REMPE: It's a different problem because it's --

MR. DICKSON: It is.

MEMBER REMPE: -- a penalty for the high burnup, high enriched fuels because of our increase in knowledge. And how does one deal with that?

MR. DICKSON: This is the beginning of that process looking at this report and doing these additional analyses. At this point, I can only share so much and how we want to look at this and what we want to be doing in Rev. 2, Reg Guide 1.183.

MEMBER REMPE: This is not going to come

up in the next month or two.

MR. DICKSON: Timeline, it's matching up with increased enrichment. So this is under the umbrella of increased enrichment. So I think we said beginning or end of the school year, calendar year 2024, beginning of 2025 we'll be seeing something.

MEMBER ROBERTS: I'm not sure you answered Vicki's questions. If you can go back to Slide 8. No, the one after that, the one with the colors. That one. No. Thank you.

Yeah, I think what I've got out of these last couple hours is there's really no difference in ultimate consequence between the blue, the orange, and the green because they're driven by changes in modeling. And the fission products are going to come out from the core. At some point, the progression, just those somewhat arbitrary division of lower vessel head rupture.

That defines when you stop making that plot. But what Vicki asked I think is interesting. If you have a Level 3 PRA with these different models, would the end result be significantly different? Because the timing, right, is not all that different.

It's the timing of the events that are happening, not so much the timing of the degradation of the fission product release. So if you did a Level 3 PRA with these different models, would you have a different answer? Or would it just all come out in the wash? I mean, there's nobody here who can answer that question.

DR. ESMAILI: And this is Hossein Esmaili. Can I just say so we're not going to be using this or any Level 3 PRA. Remember what Shawn showed in the first or second slide that a Level 3 PRA, that's the thing that we are actually doing right now is that we are going to be doing a very mechanistic accident progression source term calculation, going all the way to lower head failure, MCCI, et cetera.

So whatever is going to come out of that, whether it's a containment failure, what is the release to the containment. So we are not looking at the release to the containment. And don't forget, this release to the containment, these bars that you see, it's everything.

It's airborne, deposit that in the suppression pool, et cetera. In a consequence analysis, we are not going to be looking at what's in the deposit, et cetera. So for a Level 3 PRA type of analysis, we are just going to go that path, the one that Shawn showed at the beginning.

We're going to go all the way, do a very integrated analysis and look at what's going to come out of the containment, whenever it's leaking or containment failure, et cetera. This one is just for the purpose of doing that bottom one which means that we are doing a simplified approach right? I mean, that simplified approach, somebody is going to take care of the position, et cetera, those lambdas in the real -- up one in the integrated analysis, that's all part of the calculation.

MELCOR will calculate the position and the structure, the position on the pool, in the pool surface itself. So in that respect, you know, we are just being very mechanistic. What we come down to is that we specify what that lambda is from this calculation, right?

MEMBER ROBERTS: Okay. Thanks, Hossein. I think what I got out of that is the blue that's there would be NUREG-1150. So whatever models produce the blue also produce results at NUREG-1150. The green is whatever is coming out of your current Level 3 PRA project.

And so you would see if there's an effect between the blue and the green if you can find it with all the other different things that have changed over the last 30 years. But yeah, thanks for the answer. I think that answered my question. So there's no -- in the severe accident world or Level 3 PRA world, there's really no distinction to be drawn by this blue-yellow or orange-green.

(Simultaneous speaking.)

DR. ESMAILI: Yes, in Level 3 PRA, we would be looking at the accident sequences and just combining them, whatever the plant damage say, what is the release category, et cetera. This thing is only when we come up with that green box that Shawn showed. And then we do that path of a simplified model.

MEMBER ROBERTS: I didn't get an answer to Charlie's question then. Is the blue versus orange versus green have a real meaning in the reactor space?

MR. CAMPBELL: Yeah, that's sort of what I want to say.

MEMBER BROWN: I walked away with that. I transitioned back to the regular stuff. There's still the differences you have to deal with.

MR. CAMPBELL: Yeah, so some of these are artifacts because it's these regulatory applications. Like, Level 3 PRA, you'd actually -- you want to consider taking out the effects of the suppression pool. You have to do that for the downstream licensing calcs only. So that's --

CHAIR PETTI: I just think it shows how difficult it is to take what's done and let's call it the state of the art. And to try to boil something down that you can put into a licensing approach that's simple because you don't want something complex because there's so much artificiality. And you can get misled as much as you can get an answer that's good. So good luck, Elijah.

MEMBER REMPE: Okay. So back in the NUREG-1465 days, they thought an hour and a half was enough for a source term, right? If you did an hour and a half with your 2023 source term, you'd probably get a lower amount, not a higher amount, right, or something comparable? If you want to be consistent, why not just say an hour and a half and cut it off because vessel failure is irrelevant.

DR. ESMAILI: So this is Hossein Esmaili. So Joy, I'm just going to defer to NRR, right, because we are -- no, I'm just trying to be very careful because what we are doing is that we are showing everything. We are showing you everything.

This is what happens in the containment. This is what happens if you just take out the suppression pool. You can see in terms of the concentration. And concentration is what's really driving this. You go from that green, that big green. You go to that red which is --

(Simultaneous speaking.)

DR. ESMAILI: And we are going to have meetings. We're going to have public workshops in the next few months starting in January going to April. And whatever regulatory decision they're going to make, it's going to come at the end of that.

MEMBER REMPE: So the question really wasn't for you or for --

(Simultaneous speaking.)

MEMBER HALNON: Okay.

MEMBER REMPE: It's more for Elijah. Elijah, that's what was done back in the days of 1465. And that was enough in order to be a consistent predictable regulator. It seems to me that that was enough back then to say, okay.

DR. ESMAILI: But talking about the vessel and lower head failure, we have learned over the years that there was the calculations we used to do, like, 20, 30 years ago, as soon as the debris would melt and come down. It would reach with the lower head, whether it's a drain plug, whether it's an instrument tube, and fail it and go out. So that was that.

Nowadays --

MEMBER REMPE: We used to worry about high pressure injection too and a lot of things we don't worry about anymore. But anyway, it's just a suggestion that might be --

DR. ESMAILI: Yeah, yeah. No, that's a very good suggestion. All I'm suggesting is that some of the -- this durations, and this came out of SOARCA, is because we're doing a better modeling of heat transfer to the water, you know, that it has to boil. There's a massive amount of structures in the lower plenum of a BWR against the duration of the (audio interference). So all of those things factor into the fact that now it's in vessel phase. Actually, this was a SOARCA insight.

It's going to take a long time, right? And I hear you. I'm not going to make any judgments. Leave it up to NRR, what they want to do. But we have the data. We have the analysis. We can go data mining. We have other choices. It's just that decision has not been --

MEMBER REMPE: I get it. I just was trying to figure out how one would get out of this mess. And so it's not a question.

DR. ESMAILI: Yeah, the other thing is

that you can see that we talk about the suppression pool. It's really not -- we should not be comparing the source term. We should really comparing the concentrations because when you compare, like, in a BWR, we have the suppression pool.

So you automatically see that the concentration goes down with the different Mark I or Mark III design. But in the PWR case, you don't have a suppression pool. But you have a huge containment volume, right?

So the concentration you can see, it's just like comparable even to a -- the Mark III. So that was the purpose of showing this. Sorry, thank you.

CHAIR PETTI: I guess we should probably go out for public comment. Any members of the public that would wish to make a comment, please unmute yourself, name, organization, and comment.

Hearing nobody online, we do have someone in the room. Please go ahead.

MR. CSONTOS: So I just wanted to say thank you very much for the presentations. The first time -- oh, Al Csontos, NEI, Director of Fuels. Thank you very much for the presentation and discussion. I think it's very helpful. We had a lot of comments and questions and looking forward to having those workshops that you talked about.

I think that will be really important to count as dialogue on how the impact of the new modeling that's here goes out to the actual implementation steps. Charlie, to your question, a comment you made. EPRI did do a scoping study looking at the impact of high enrichment, higher burnup on the back end. Okay. And --

MEMBER BROWN: Without severe accident.

MR. CSONTOS: Yes, this is more -- you had mentioned dry cask spent fuel, things like that. That report number is 3002027535. I'll pass it on to Larry and folks to provide you.

But the bottom line there was that there's very little impact to the back end. There's a small -- longer time that you might have to put it in the pool, but it's manageable. And actually dose rates for the workers actually goes down because you're loading less canisters. So that's the -- but you can read the study.

MEMBER BROWN: The good news is you've looked at it.

MR. CSONTOS: Yes, we've looked at it. But thank you very much and more on this later. And also during the rulemaking reg basis comments that the industries would like.

CHAIR PETTI: Thank you. With that, I don't see any more comments. So I think we -- oh, I want to -- yes.

MEMBER HALNON: Yeah, you guys did a great job. I mean, you kept energy through this whole thing, and that kept us going. For those of us like my young friend here who are not modelers, it was understandable. Appreciate it.

MEMBER BROWN: I couldn't have asked the question without absorbing a little bit of what you said for the last two and a half hours, three hours, whatever it is now.

MEMBER REMPE: It's good to hear things. I mean, one, we were curious about this because of the Reg Guide 1.183 but also the new calculations and understanding this will help others when they're trying to deal with Reg Guide 1.183, Rev. 2.

CHAIR PETTI: Very important when we hear the next go around of (audio interference). Technical basis here was just very informative. Okay. Thank you everyone, and we are adjourned.

(Whereupon, the above-entitled matter went off the record at 4:44 p.m.)

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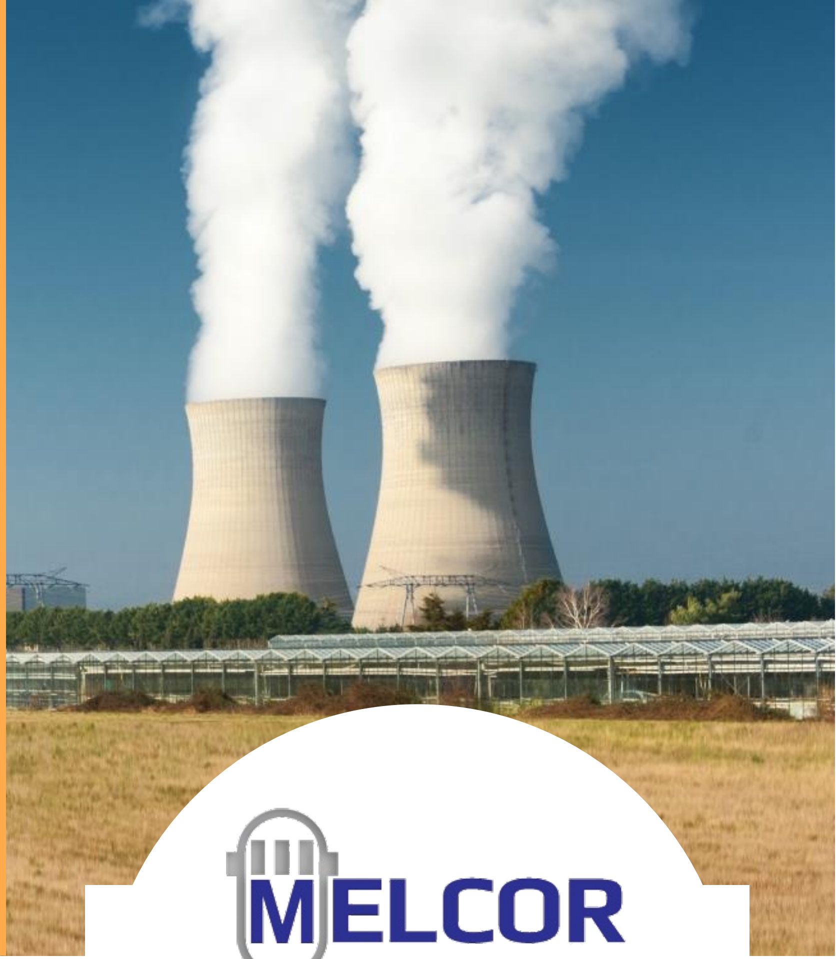

MELCOR

High Burnup Fuel Accident Source Terms

ACRS Briefing Nov 16, 2023

Presented by Lucas I. Albright and David L. Luxat

U.S. DEPARTMENT OF ENERGY NNSA

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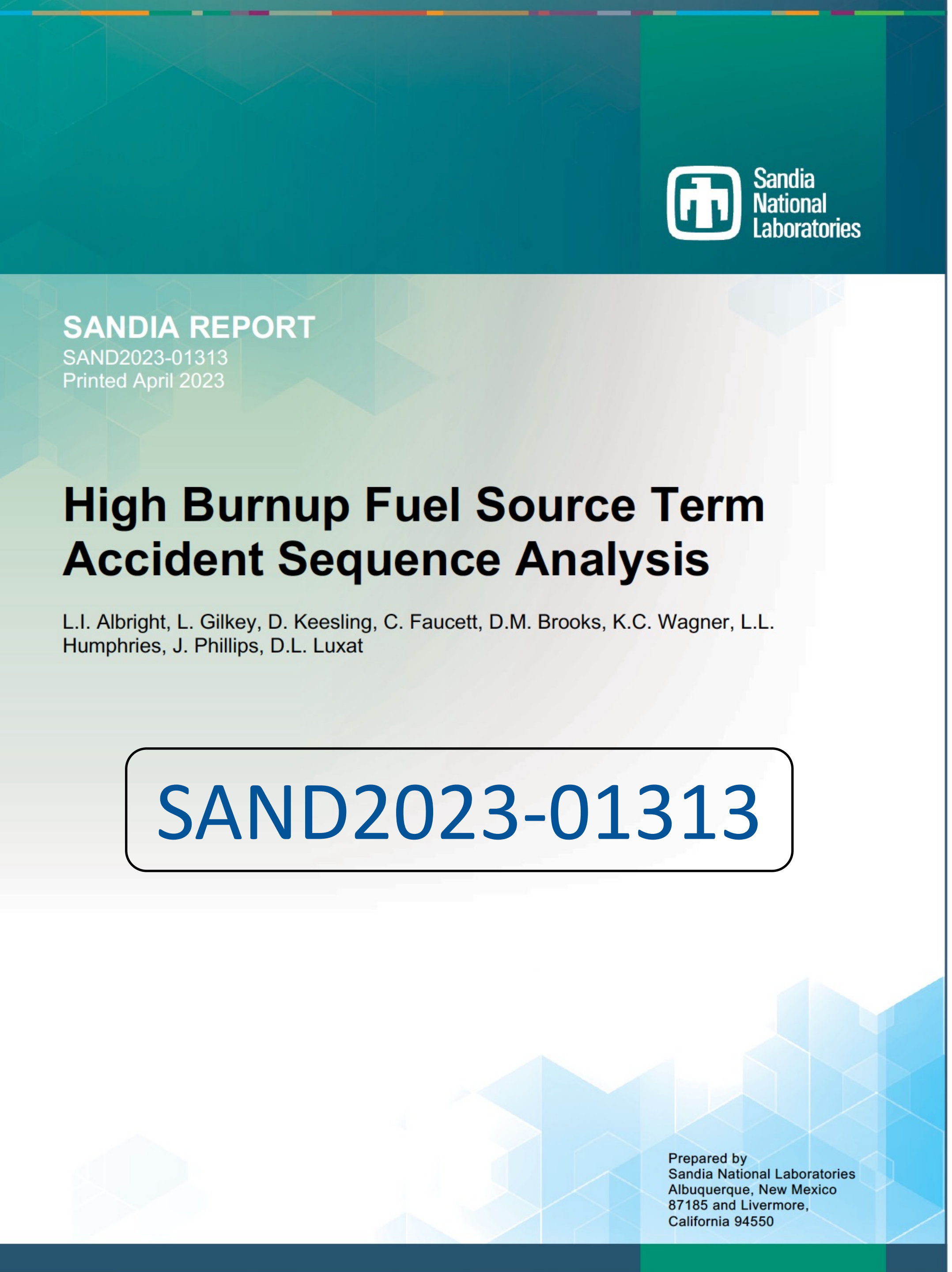
Contents

Motivation and Background


SANDIA REPORT
SAND2023-01313
Printed April 2023

Sandia National Laboratories

High Burnup Fuel Source Term Accident Sequence Analysis

L.I. Albright, L. Gilkey, D. Keesling, C. Faucett, D.M. Brooks, K.C. Wagner, L.L. Humphries, J. Phillips, D.L. Luxat

SAND2023-01313

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550

High Burnup Fuel Source Term Analysis Motivation

- Develop alternative source term applicable to LWR cores with HBU/HALEU fuel
  - Different burnup levels and enrichments considered
- Extends NUREG-1465 and SAND2011-0128 alternative source terms

Historically Relevant Studies

- **TID-14844**: "Calculation of Distance Factors for Power and Test Reactors," – USAEC 1962

- **NUREG-1465** – "Accident Source Terms for Light-Water Nuclear Power Plants," – USNRC 1995 (code: STCP)

- **SAND2011-0128** – "Accident Source Terms for Light- Water Nuclear Power Plants Using High-Burnup or MOX Fuel" (code: MELCOR 1.8.5)

Source Term Timeline

Major Developments

- TID-14844
- MELCOR
- NUREG-1560
- RG 1.183
- Phébus FP
- Fukushima Daiichi
- SOARCA V1 Peach Bottom
- SOARCA V2 Surry
- BSAF
- HBU/HALEU/ATF SA PIRT
- SOARCA UAs
- ATF Source Term

‘95 ‘11 ‘23

Accident Source Terms

NUREG-1465

NUREG–1465

Accident Source Terms for Light-Water Nuclear Power Plants

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

L. Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, J. N. Ridgely

→

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SANDIA REPORT
SAND2011-0128
Unlimited Release
January 2011

Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel

D.A. Powers, M.T. Leonard, R.O. Gauntt, R.Y. Lee, and M. Salay

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Participation in OECD/NEA International Programs

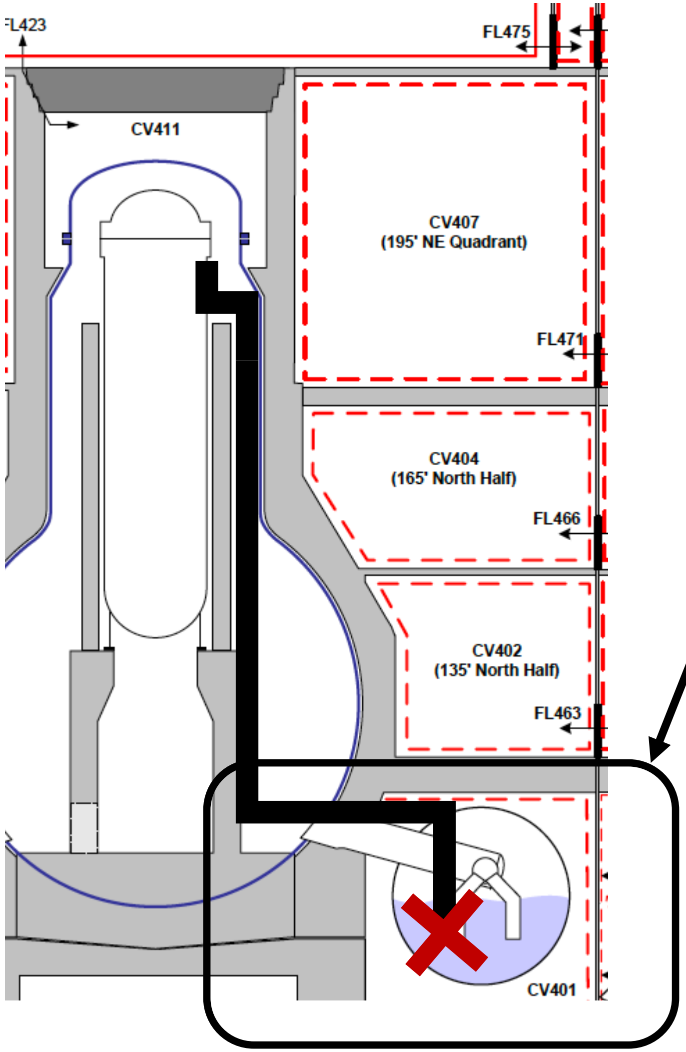
Severe Accident Modeling Advancements

- **Heterogeneous, integrated reactor core modeling** tends to promote to progressive and extended core degradation.
  - 2D discretization of the reactor core
  - No more distinct “gap release phase”
  - Prolonged core damage progression
  - Longer times to lower head failure

- **Prevalence of accident-induced low-pressure scenarios – SOARCA**
  - Thermally induced SRV seizure for majority of BWR sequences
  - Hot leg creep rupture for majority of PWR sequences

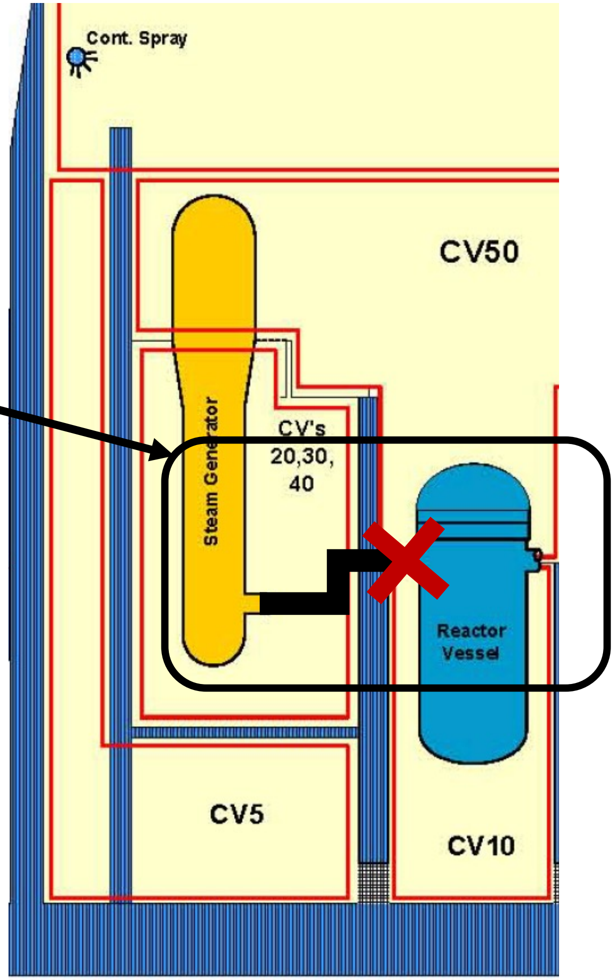
Impact of Early Depressurization

BWR: Thermally induced SRV seizure



Early loss of the primary pressure boundary induces depressurization of the reactor coolant system and opens a release pathway for radionuclides to transport directly to containment during early in-vessel core degradation

PWR: Hot Leg Creep Rupture



*Diagrams are for illustration purposes only

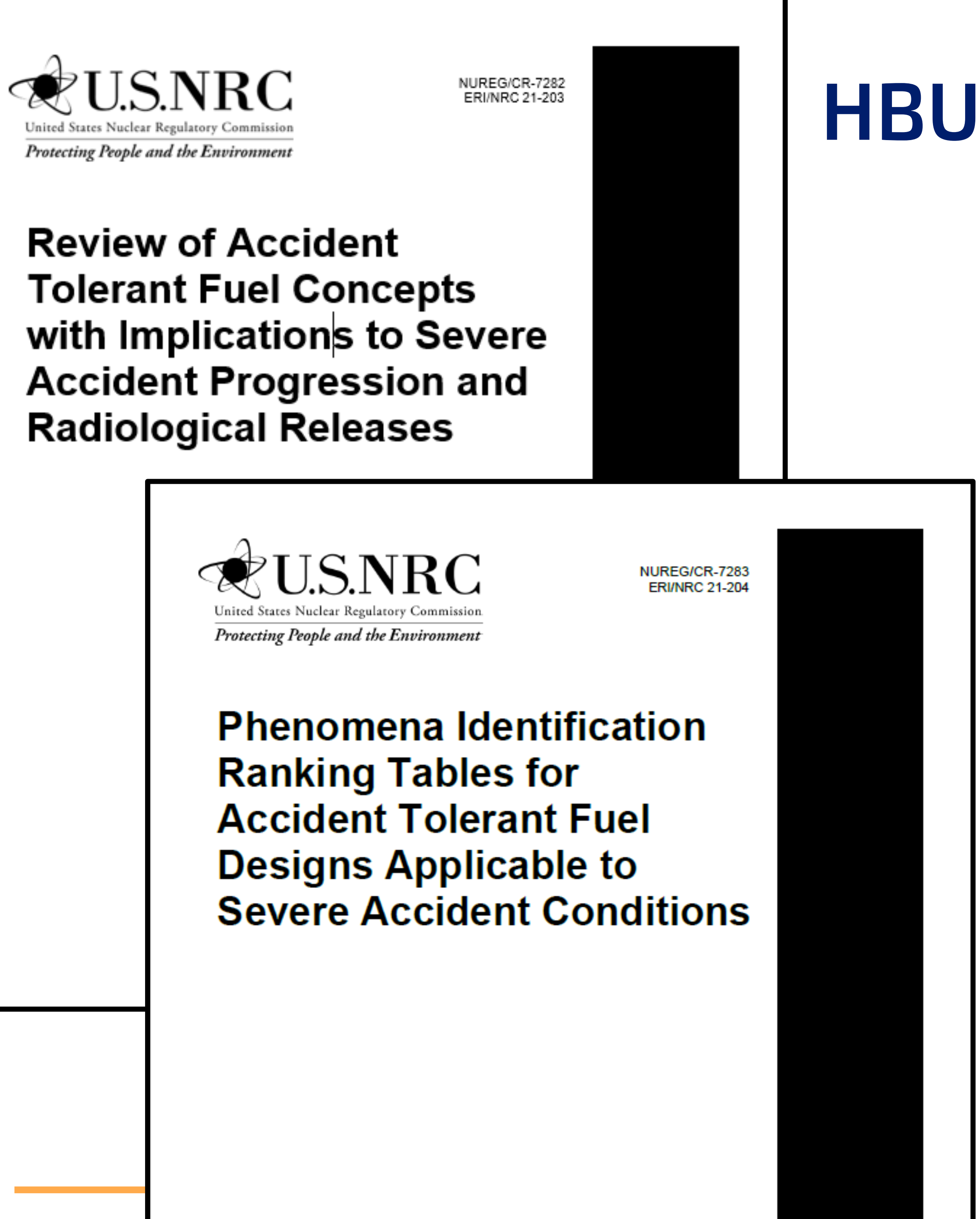
Selected Severe Accident Datasets

More recent severe accident datasets have improved characterization of core damage progression and subsequent radionuclide releases since NUREG-1465

- Severe accident experiments used to validate severe accident codes
  - Phébus FP
    - Early fuel failure
    - Hypothesized $CsMoO_4$ as the dominant chemical form of Cs
  - VERCORS
    - Early fuel failure
    - High burnup fission product release rates
- Severe accidents are a primary data source for severe accident code validation
  - Fukushima Daiichi
    - Existing data confirms that $CsMoO_4$ is the dominant chemical form of Cs

Severe Accident Knowledge Advancements

- **Chemical form of iodine:**
  - NUREG-1465 assumed 95% of iodine in the form of CsI
  - Current practice assumes all Iodine to be CsI
  - Still assume 5% of the total iodine inventory is present in the gap inventory
- **Chemical form of cesium:**
  - NUREG-1465 assumed Cs predominantly in the form of volatile CsOH
  - Current best-practice assumes 5% of cesium present in the gap inventory as both CsI and CsOH
  - All remaining cesium assumed to react with Mo to form Cs_2MoO_4
- **Mo release:**
  - Mo releases are now higher than other metallic fission products such as Ru and Pd.
- **Te release:**
  - Current best practice is more extensive Te release than reported in NUREG-1465
  - Due to change in chemical form with more efficient transport of Te to containment


U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment
NUREG/CR-7282
ERI/NRC 21-203

Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment
NUREG/CR-7283
ERI/NRC 21-204

Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions

HBU/HALEU/ATF PIRT

- **HBU/HALEU fuel severe accident behavior**
  - No significant differences between HBU and HBU/HALEU fuels
  - Thermophysical property differences expected
  - Fuel fragmentation and sintering can impact core degradation
  - Fission product chemistry may change
  - Possibility of cladding embrittlement
  - Greater potential for recriticality during reflood using unborated water for HALEU

Key Findings

Study Highlights

Key Finding 1: Increased burnup and enrichment does not strongly impact in-containment source term

- Most significant variation in source term arises due to differences between accident scenarios

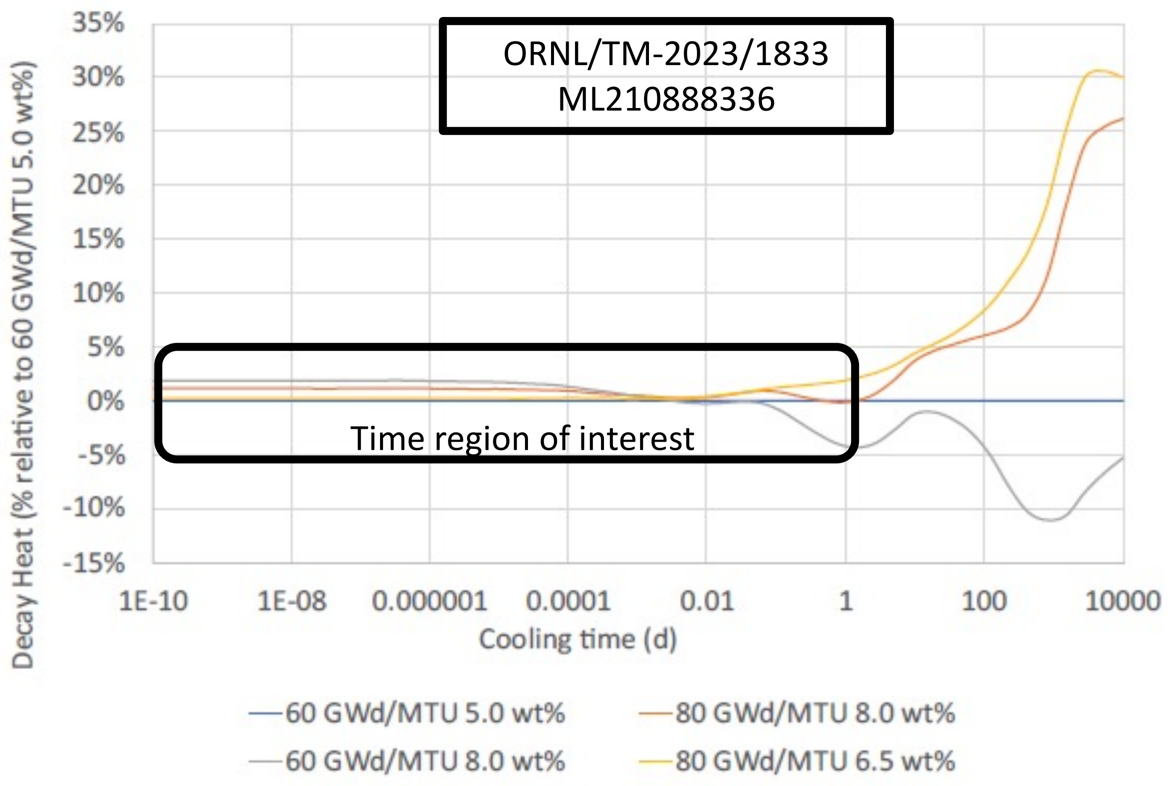
Key Finding 2: Larger early releases to containment result from early primary pressure boundary failure

- Set of accident scenarios dominated by low pressure accident sequences
- NUREG-1465 prescribed a larger number of high pressure scenarios

Key Finding 3: Releases to containment significantly reduced if primary pressure boundary remains intact

- Low pressure scenarios lead to more significant releases to containment
- Evolution of severe accident modeling state-of-art since NUREG-1465 (e.g., SOARCA)

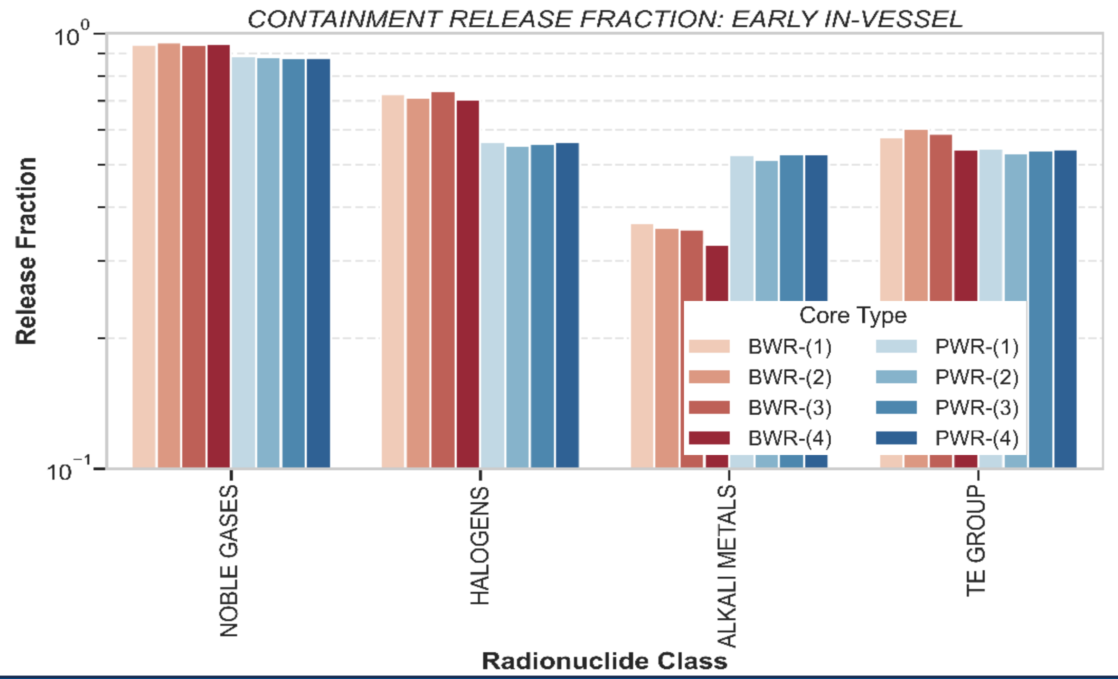
High Burnup and Extended Enrichment Impact on Source Term



Core Types:

(1) 60 GWd/MTU LEU (2) 80 GWd/MTU LEU

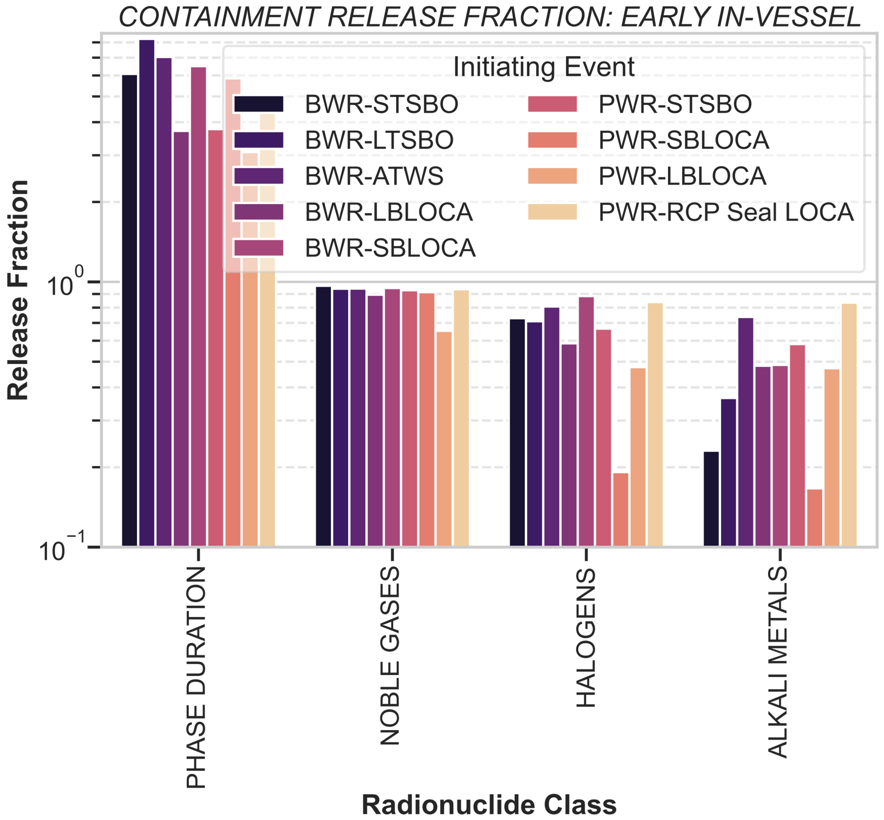
(3) 60 GWd/MTU HALEU (4) 80 GWd/MTU HALEU



Burnup and enrichment do not significantly change decay heat after reactor shutdown

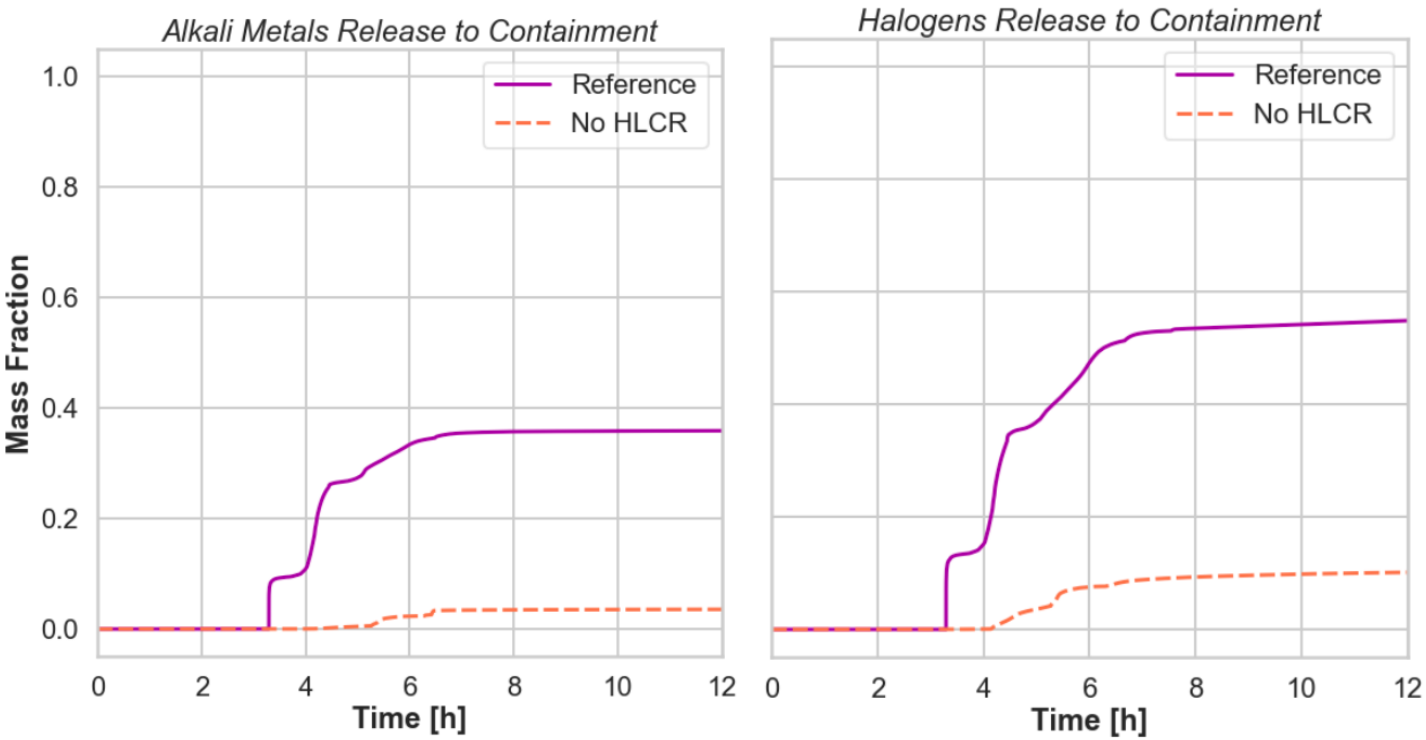
Increased burnup and enrichment does not strongly impact in-containment source term

Impact of Accident Scenarios on In-containment Source Term



Accident progression and in-containment source terms different across accident sequences

Reference	Hot leg creep rupture enabled
No HLCR	Hot leg creep rupture disabled



Primary pressure boundary failure during critical accident phases is a significant factor in accident progression and in-containment source term

In-Containment Source Term Differences

MELCOR

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
BWR	Report	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	Phase Duration	0.70	0.50	6.7	1.5	44.6	10.0	3.1	3.0
	Noble Gases	0.016	0.050	0.95	0.95	0.005	0.0	0.011	0.0
	Halogens	0.005	0.050	0.71	0.25	0.16	[illegible]	[illegible]	.30
	Alkali Metals	0.005	0.050	0.32	0.20	0.021	[illegible]	[illegible]	.35
	Te Group	0.003	0.0	0.56	0.050	0.19	[illegible]	[illegible]	.25
PWR		Gap Release		Early In-vessel		Late			
	Phase Duration	1.3	0.50	4.0	1.3	24.0	[illegible]	[illegible]	.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010	[illegible]	[illegible]	.0
	Halogens	0.007	0.050	0.58	0.35	0.031	0.10	0.020	0.25
	Alkali Metals	0.003	0.050	0.50	0.25	0.013	0.10	0.015	0.35
	Te Group	0.006	0.0	0.55	0.050	0.019	0.005	0.005	0.25

The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation

In-Containment Source Term Differences

	Report	Gap Release 2023	Gap Release NUREG-1465	Early In-vessel 2023	Early In-vessel NUREG-1465	Late In-vessel 2023	Late In-vessel NUREG-1465	Ex-vessel 2023	Ex-vessel NUREG-1465
BWR	Phase Duration	0.70	0.50	6.7	1.5	44.6	10.0	3.1	3.0
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The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation
- Progressive releases to containment due to enhanced reactor coolant system modeling

In-Containment Source Term Differences

MELCOR

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
BWR	Report	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
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PWR		Gap Release		Early In-vessel		Late			
	Phase Duration	1.3	0.50	4.0	1.3	24.0	[illegible]	[illegible]	2.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010	[illegible]	[illegible]	.0
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The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Longer in-vessel phase durations due to progressive core degradation
- Progressive releases to containment due to enhanced reactor coolant system modeling
- Larger release magnitudes prior to lower head failure due to early loss of the primary pressure boundary (by safety relief valve seizure and hot leg creep rupture)

In-Containment Source Term Release Rates

BWR		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Report	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	Phase Duration	0.70	0.50	6.7	1.5	44.6	10.0	3.1	3.0
	Noble Gases	0.023	0.10	0.14	0.63	0.0001	0.0	0.003	0.0
	Halogens	0.007	0.10	0.11	0.17	0.004	[illegible]	[illegible]	[illegible]100
	Alkali Metals	0.007	0.10	0.047	0.13	0.0006	[illegible]	[illegible]	[illegible].12
	Te Group	0.005	0.0	0.091	0.033	0.005	[illegible]	[illegible]	[illegible]083
PWR		**Gap Release**		**Early In-vessel**		**Lat[illegible]**		[illegible]	
	Phase Duration	1.3	0.50	4.0	1.3	24.0	[illegible]	[illegible]	[illegible].0
	Noble Gases	0.019	0.10	0.21	0.73	0.0008	[illegible]	[illegible]	[illegible].0
	Halogens	0.003	0.10	0.16	0.27	0.001	0.010	0.009	0.12
	Alkali Metals	0.001	0.10	0.15	0.19	0.0005	0.010	0.008	0.17
	Te Group	0.003	0.0	0.15	0.038	0.0008	0.0005	0.002	0.12

The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Assumes uniform release rate across the entire phase duration

In-Containment Source Term Release Rates

MELCOR

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
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	Halogens	0.003	0.10	0.16	0.27	0.001	0.010	0.009	0.12
	Alkali Metals	0.001	0.10	0.15	0.19	0.0005	0.010	0.008	0.17
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The NRC has determined (SECY-94-302, December 19, 1994) that design basis source terms will not include the ex-vessel and late in-vessel phases.

- Assumes uniform release rate across the entire phase duration
- Release rates (release fraction/hour) are generally smaller

In-Containment Source Term Release Rates

		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
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	Noble Gases	0.023	0.10	0.14	0.63	0.0001	0.0	0.003	0.0
	Halogens	0.007	0.10	0.11	0.17	0.004	[illegible]	[illegible]	100
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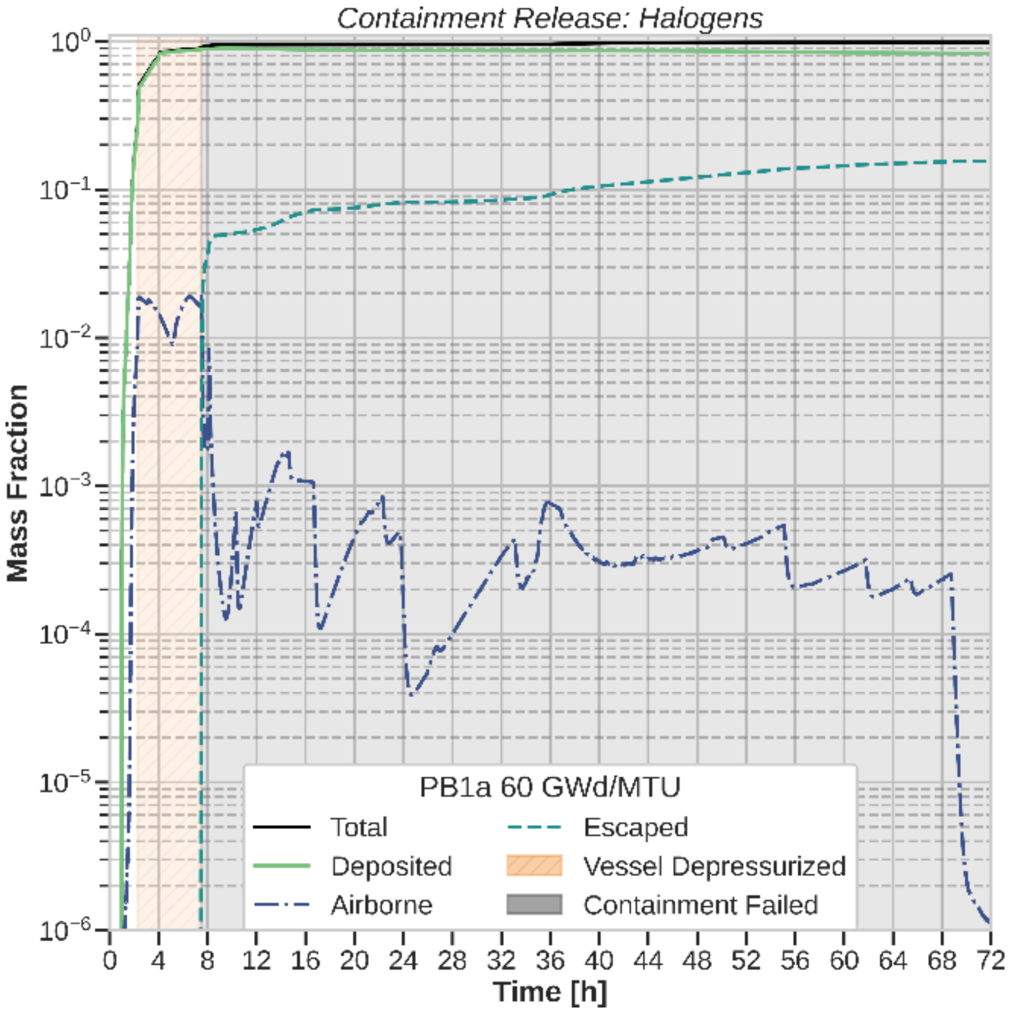
- Assumes uniform release rate across the entire phase duration
- Release rates (release fraction/hour) are generally smaller
- Larger Te group release magnitude prior to lower head failure

Deep Dive

In-containment Source Term

- In-containment source term characterizes **total** radioactive inventory in containment
  - In-containment source term combines deposited, airborne, and escaped radionuclide inventories
- MELCOR simulations can track deposited and airborne masses separately
  - This additional information not used in determining in-containment source term
- Radionuclide removal mechanisms accounted for in downstream calculations with RADTRAD



10 CFR 50.2 – Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release

Alternative Source Term

MELCOR

"Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183

SAND2023-01313 Peer Review Assessment (ERI/NRC 23-201)

- Fulfills Criteria — Alternative Source Term (AST) must be based on major accidents involving a substantial meltdown of the core
- Fulfills Criteria — AST must be represented in terms of the quantities, times, rates, chemical speciation for fission product release into containment
- Fulfills Criteria — AST must not based on a single accident scenario but characterizes a spectrum of credible severe accident events
- Fulfills Criteria — AST must have a defensible technical basis
- Fulfills Criteria — AST must be peer reviewed

Process for Source Term Development

1. BWR and PWR core damage accident scenario identification
2. Develop radionuclide inventory and decay heat using the SCALE code package
3. Perform accident progression and source term analyses using MELCOR
4. Develop statistically representative source term across all accident scenarios and BWR/PWR plants

Evolution from SAND2011-0128

- Overall SAND2023-01313 methodology is consistent with SAND2011-0128
  - Focus on assessing impact of HBU/HALEU fuel on alternative source term
- Key areas of consistency between the studies are
  - Nuclear power plants modeled
  - Accident scenarios simulated
  - Radionuclide chemical classes represented
  - Radiological release phases first identified in NUREG-1465 are defined using SAND2011-0128 criteria
  - Representative release phase source terms and timings are statistical median values

Extending SAND2011-0128 Source Terms

MELCOR

- Plants analyzed – *from SAND2011-0128*
  - BWR: Mark I containment (Peach Bottom) and Mark III containment (Grand Gulf)
  - PWR: Ice Condenser containment (Sequoyah) and large-dry containment (Surry)
- Accident scenarios analyzed – *from SAND2011-0128*
  - BWR: SBLOCA, LBLOCA, STSBO, LTSBO, ATWS
  - PWR: SBLOCA, LBLOCA, STSBO

Phase	Onset Criteria – *from SAND2011-0128*	End Criteria – *from SAND2011-0128*
Gap Release	RPV water level below top of active fuel	Release of 5% of initial, total Xe inventory from fuel
Early In-Vessel	Release of 5% of initial, total Xe inventory from fuel	Lower Head Failure
Ex-Vessel	Lower Head Failure	95% of total ex-vessel Cs releases
Late In-Vessel	Lower Head Failure	95% of total late in-vessel Cs releases

Peer Review Findings

- Ex-vessel and late in-vessel phase criteria have limited technical justification
- NRC determined (SECY-94-302, December 19, 1994) design basis source terms will not include ex-vessel and late in-vessel phases

SAND2023-01313 Accident Selection

- NUREG-1560 – "Individual Plant Examination Program"
- Based on SAND2011-0128 accident selection
  - Consistent with NUREG 1560 IPE results
- Representative accident sequences similar to those selected for NUREG-1465
  - Provides coverage of all major sequences
- Incorporates SBO, LOCA and ATWS scenarios and range of mitigating system operation

Peer Review Findings

- More recent PRA studies may potentially show different core damage contributors
- For the intended applications the scenarios used in the current [SAND2023-01313] appropriate with regards to the progression of severe accidents, radionuclide release and transport.

Peach Bottom Accident Scenarios

MELCOR

Initiating Events

7 SBOs

- 4 immediate loss of DC power
- 3 with prolonged DC power

2 LOCAs

Coolant Injection

RCIC operation

- 3 scenarios credited RCIC

No coolant injection

- 6 scenarios had no credit for any coolant injection system

RPV Status

RPV at low pressure prior to lower head failure

- 8 low pressure scenarios

RPV at high pressure until lower head failure

- 1 high pressure scenario

Containment Status

Early failures

- Drywell liner melt-through
- Torus overpressure
- Drywell head flange leakage

Late Failure

- High-temperature penetration failure

Containment failures occurred at or after lower head failure

Grand Gulf Accident Scenarios

MELCOR

Initiating Events

5 SBOs

1 ATWS

1 LOCA

- Recirculation line break

Coolant Injection

RCIC operation

- 3 scenarios credited RCIC

No coolant injection

- 4 scenarios had no credit for any coolant injection system

RPV Status

RPV at low pressure prior to lower head failure

- 6 low pressure scenarios

RPV at high pressure until lower head failure

- 1 high pressure scenario

Containment Status

Early failures

- Hydrogen deflagration
- High containment pressure (ATWS)

Late Failure

- High containment pressure

Containment failures generally occurred at or after lower head failure

Surry Accident Scenarios

Initiating Events

2 SBOs

3 LOCAs

Coolant Injection

1 scenario crediting coolant injection

4 scenarios with no coolant injection

RPV Status

RPV at low pressure prior to lower head failure

- 3 low pressure scenarios

RPV at high pressure in SBOs

- **All** SBOs exhibit hot-leg creep rupture prior to lower head failure

Containment Status

Early failures

- Hydrogen deflagration

Late Failure

- High containment pressure

Containment failures occurred at or after lower head failure

Sequoyah Accident Scenarios

Initiating Events

2 SBOs

5 LOCAs

Coolant Injection

5 scenarios crediting coolant injection

2 scenarios with no coolant injection

RPV Status

RPV at low pressure prior to lower head failure

- 2 low pressure scenarios

RPV at high pressure in SBOs

- **All** SBOs and RCP seal LOCAs exhibit hot-leg creep rupture prior to lower head failure

Containment Status

Early failures

- Hydrogen deflagration

Late Failure

- High containment pressure

Containment failures occurred at or after lower head failure

BWR Radionuclide Inventories

MELCOR

Class (kg)	60 GWd/MTU - 5 wt% Enrichment	80 GWd/MTU - 5 wt% Enrichment	60 GWd/MTU - 10 wt% Enrichment	80 GWd/MTU - 10 wt% Enrichment
BWR Mark I – Peach Bottom				
Noble Gases	1323.99	1848.13 (+40%)	1280.34 (-3%)	1790.12 (+35%)
Halogens	52.83	73.70 (+40%)	49.41 (-6%)	69.53 (+32%)
Alkali Metals	748.78	980.11 (+31%)	817.97 (+9%)	1082.33 (+45%)
Te Group	142.94	195.01 (+36%)	139.99 (-2%)	190.51 (+33%)
Ba/Sr Group	551.99	763.09 (+38%)	586.41 (+6%)	814.05 (+47%)
Ru Group	1058.01	1598.56 (+51%)	919.02 (-13%)	1374.61 (+30%)
Mo Group	973.05	1305.64 (+34%)	1007.92 (+4%)	1364.59 (+40%)
Lanthanides	2943.70	3702.34 (+26%)	2922.84 (-1%)	3686.46 (+25%)
Ce Group	2469.33	2916.84 (18%)	2559.90 (+4%)	3107.02 (+26%)

*percent differences shown relative to reference core (60 GWd/MTU - 5 wt% enrichment)
** all fuel bundles assumed to reach reported burnup

Peer Review Findings

- **Radionuclide class mass** differences are not equal to **radionuclide class activity** differences for the considered enrichments and burnups
- Unlikely that siting calculations would be significantly impact by burnup

PWR Radionuclide Inventories

Class (kg)	60 GWd/MTU - 5 wt% Enrichment	80 GWd/MTU - 5 wt% Enrichment	60 GWd/MTU - 8 wt% Enrichment	80 GWd/MTU - 8 wt% Enrichment
PWR with Large-Dry Containment – Surry				
Noble Gases	740.20	987.15 (+33%)	717.66 (-3%)	959.00 (+30%)
Halogens	29.31	39.35 (+34%)	27.44 (-6%)	37.06 (+26%)
Alkali Metals	421.27	537.41 (+28%)	455.26 (+8%)	584.21 (+39%)
Te Group	74.62	99.01 (+33%)	73.02 (-2%)	96.81 (+30%)
Ba/Sr Group	305.28	401.76 (+32%)	323.92 (+6%)	428.01 (+40%)
Ru Group	559.35	807.23 (+44%)	487.92 (-13%)	701.18 (+25%)
Mo Group	530.59	689.06 (+30%)	546.71 (+3%)	714.95 (+35%)
Lanthanides	1035.01	1396.16 (+35%)	1048.46 (+1%)	1409.24 (+36%)
Ce Group	1535.14	1780.67 (+16%)	1599.41 (+4%)	1903.19 (+24%)

*percent differences shown relative to reference core (60 GWd/MTU - 5 wt% enrichment)
** all fuel bundles assumed to reach reported burnup

Peer Review Findings

- **Radionuclide class mass** differences are not equal to **radionuclide class activity** differences for the considered enrichments and burnups
- Unlikely that siting calculations would be significantly impact by burnup

Iodine and Cesium Chemical Form

- NUREG-1465
  - 5% Iodine inventory is gaseous (I_2 and other organic iodides)
  - 95% Iodine inventory is CsI
  - Remaining Cs inventory assumed volatile (CsOH)
- SAND2023-01313 – consistent with SOARCA
  - 100% Iodine inventory reacts with Cesium to form CsI
  - 5% of the total Iodine and Cesium inventory present in gap
  - Of Cesium not forming CsI
    - 5% assumed to form CsOH
    - 95% assumed to form Cs_2MoO_4

Peer Review Findings

- Uncertainty in Iodine speciation persists despite experimental studies (FPT3, DF-4, and BECARRE)
- Fukushima Daiichi post-accident analyses confirm assumption that Cs_2MoO_4 is dominant chemical form of Cs
- Recommended consideration of/validation against French CEA HBU VERDON tests

Other Analysis Assumptions

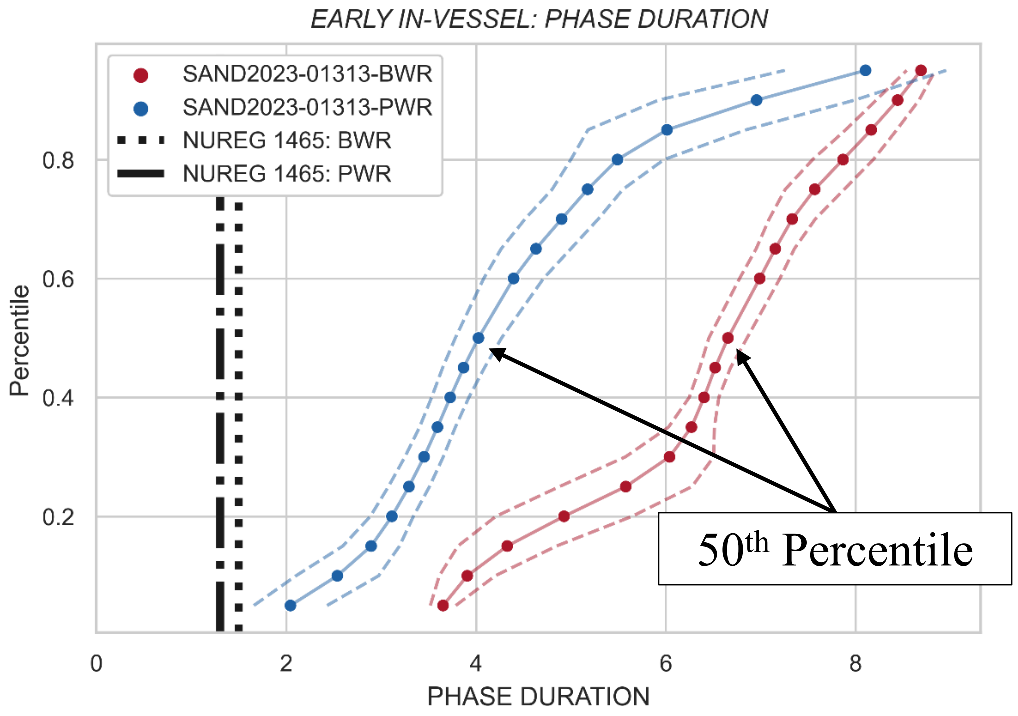
- In-containment source term does not consider impact of
  - Variation in the gap inventory at the start of the accident
  - Fraction of aerosolized iodine in containment
  - Radionuclide removal and retention in containment
- Source term analyses based on current state-of-the-art
  - Latest major code version – MELCOR 2.2
  - Majority of modeling best-practices established under SOARCA
- Some modeling best-practices have evolved since SOARCA
  - Time-at-temperature fuel rod failure model uses default time-at-temperature fuel rod lifetime curve
  - UO_2 and ZrO_2 liquefaction temperatures reduced to 2479 K to account for material interactions
  - Failure temperature of oxidized fuel rods have been reduced to 2479 K

Other Analysis Assumptions

- Relative contribution of accident sequences to total BWR/PWR CDF not changed by cores with extended enrichment HBU
- Dominant uncertainty from range of possible accidents that could be realized (i.e., aleatory uncertainty)
  - Phenomenological (or epistemic) uncertainty not incorporated into BWR/PWR in-containment source terms
  - Impact of phenomenological uncertainties considered through sensitivity calculations
  - Key phenomena identified in a PIRT study are investigated through sensitivity studies
- Containment removal mechanisms not credited
  - Some removal mechanisms, such as containment sprays, are incorporated in downstream RADTRAD calculations
  - Suppression pool scrubbing not credited
- Release fractions (source terms) below 1×10^{-6} considered negligibly small and truncated

Non-Parametric Statistical Analysis

- Non-parametric bootstrap methodology used to determine statistically representative source term across accident scenarios
  - Can be applied to data that follow any distribution
  - Utilizes repeated re-sampling (bootstrapping) of data
  - Estimates empirical cumulative distribution function (ECDF) of a given quantity of interest (QoI)
- Representative source term is the median (50th percentile) estimate from the ECDF
  - Equally weights all simulations

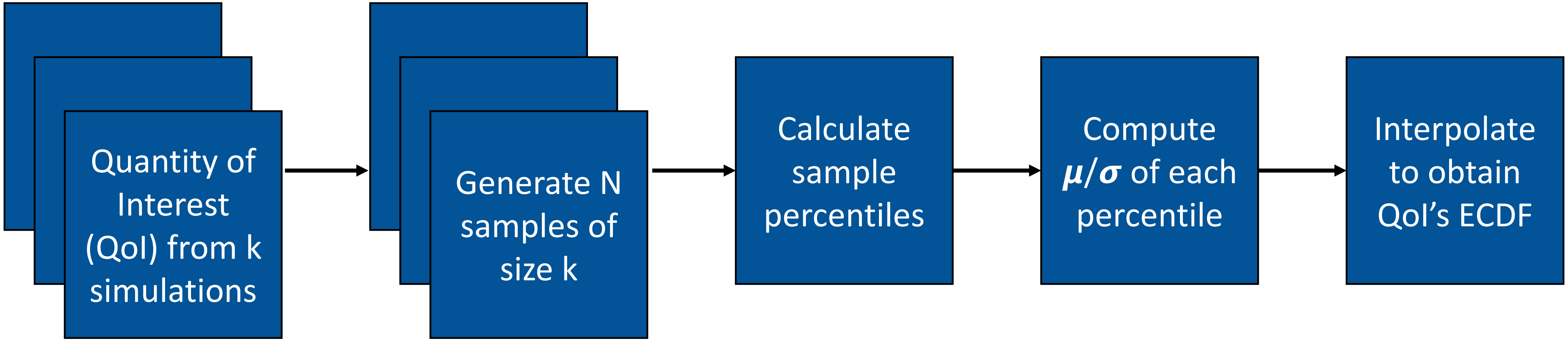


*Dashed colored lines illustrate confidence intervals spanning ± standard deviation ($\boldsymbol{\sigma}$) at each percentile

Peer Review Finding

- Representative source term based on median value appropriate to avoid introducing bias from potential outliers

Bootstrap Procedure



- Incorporates variability due to different plants and accident scenarios in representative source term
  - Bounds on empirical cumulative distribution function (ECDF) characterize sampling uncertainty

Results and Discussion

Restating Key Aspects of the Analysis

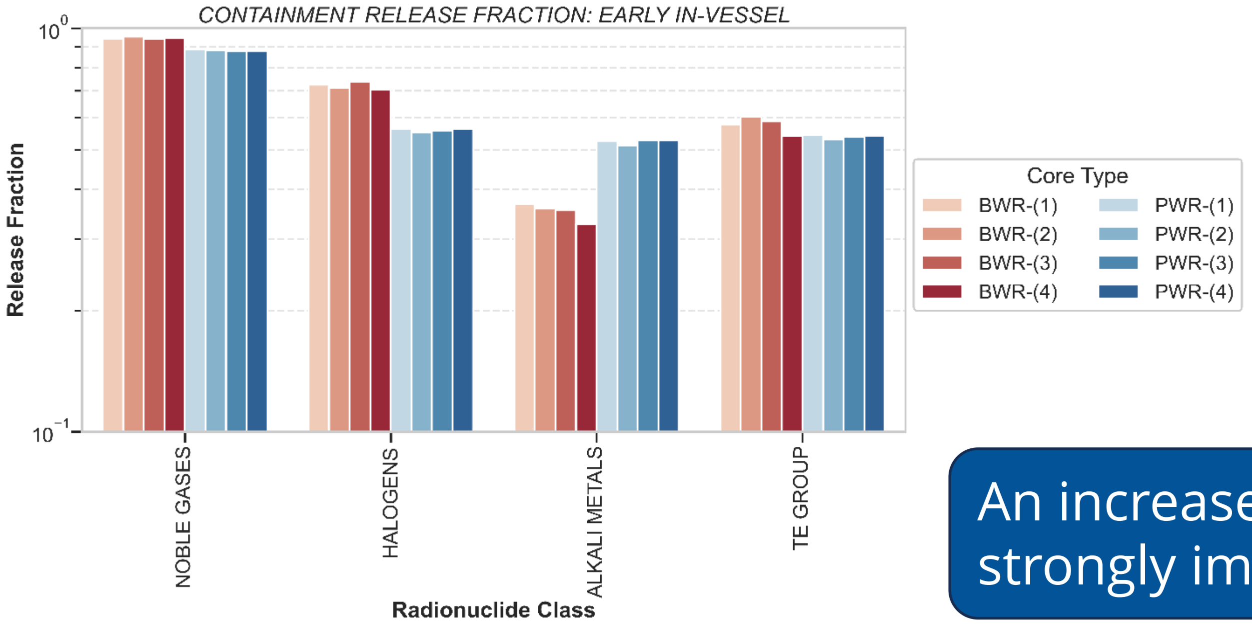
- Objective
  - Extend the NUREG-1465 alternative source term to address LWRs with cores designed to utilize HBU fuel with varied fuel enrichments
- Plants analyzed
  - BWR: Mark I containment (Peach Bottom) and Mark III containment (Grand Gulf)
  - PWR: Ice Condenser containment (Sequoyah) and Large-dry containment (Surry)
- Reactor cores analyzed
  1. Core average burnup of 60GWd/MTU for enrichment of 5 wt%
  2. Core average burnup of 80GWd/MTU for enrichment of 5 wt%
  3. Core average burnup of 60GWd/MTU for enrichment of 8 wt% (peak 10 wt% for BWRs)
  4. Core average burnup of 80GWd/MTU for enrichment of 8 wt% (peak 10 wt% for BWRs)
- Accident scenarios analyzed
  - BWR: SBLOCA, LBLOCA, STSBO, LTSBO, ATWS
  - PWR: SBLOCA, LBLOCA, STSBO

Phase	Onset Criteria	End Criteria
Gap Release	RPV water level below top of active fuel	Release of 5% of initial, total Xe inventory from fuel
Early In-Vessel	Release of 5% of initial, total Xe inventory from fuel	Lower Head Failure
Ex-Vessel	Lower Head Failure	95% of total ex-vessel Cs releases
Late In-Vessel	Lower Head Failure	95% of total late in-vessel Cs releases

Revisiting the Impact of Reactor Core on In-containment Source Term

BWR		Early In-vessel			
	Core Type	(1)	(2)	(3)	(4)
	Phase Duration	6.7	6.3	6.5	6.3
	Noble Gases	0.94	0.96	0.94	0.94
	Halogens	0.71	0.71	0.76	0.71
	Alkali Metals	0.31	0.31	0.31	0.26

PWR		Early In-vessel			
	Core Type	(1)	(2)	(3)	(4)
	Phase Duration	4.0	3.8	4.2	3.8
	Noble Gases	0.93	0.92	0.91	0.92
	Halogens	0.57	0.56	0.57	0.58
	Alkali Metals	0.5	0.5	0.5	0.51



Core Types:

(1) 60 GWd/MTU LEU,

(2) 80 GWd/MTU LEU

(3) 60 GWd/MTU HALEU

(4) 80 GWd/MTU HALEU

An increase in burnup and enrichment does not strongly impact the in-containment source term

BWR In-containment Source Term Evolution

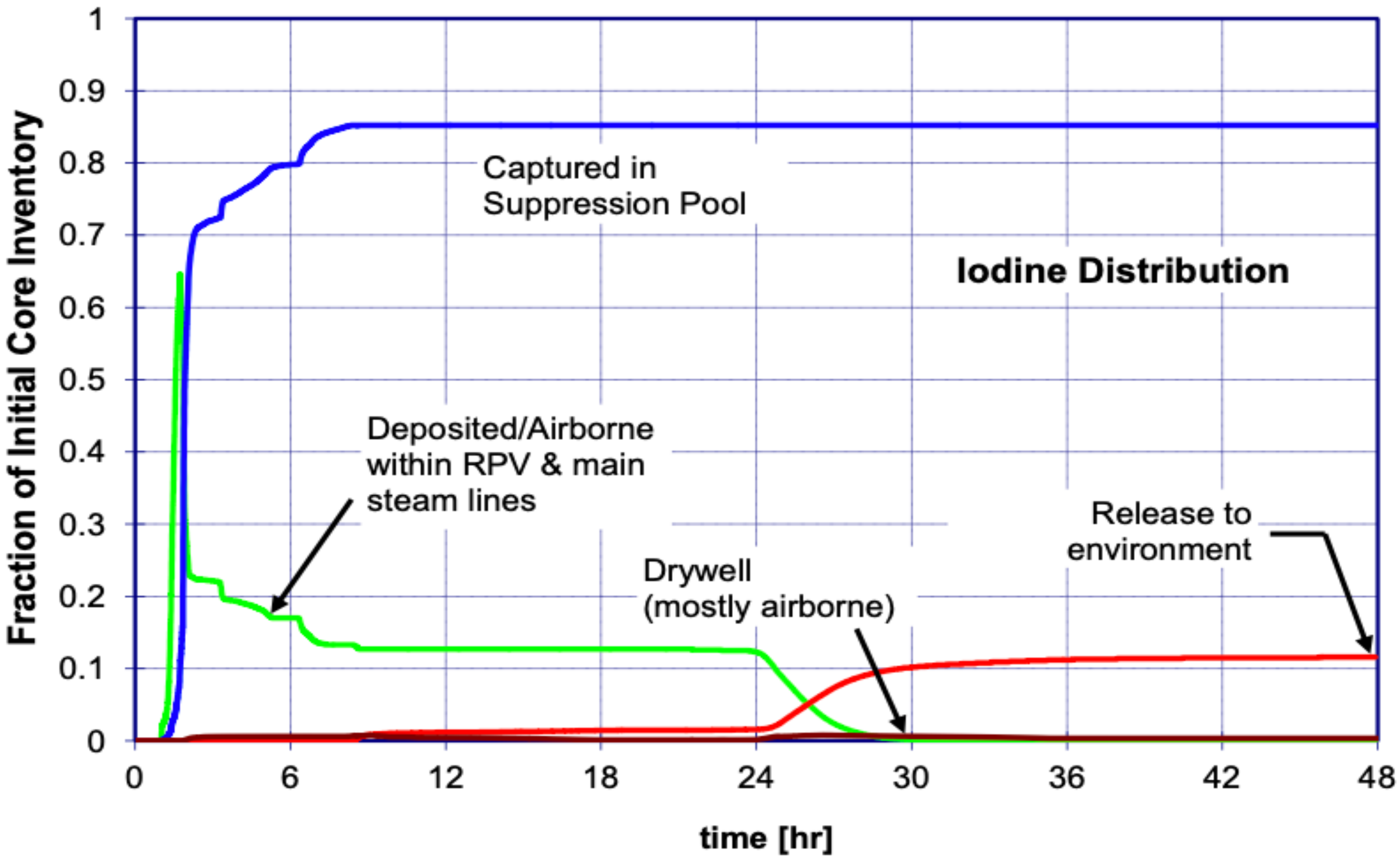
	Gap Release			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Phase Duration (hr)	0.70	0.16	0.50	6.7	8.0	1.5
Noble Gases	0.016	0.008	0.050	0.95	0.96	0.95
Halogens	0.005	0.002	0.050	0.71	0.47	0.25
Alkali Metals	0.005	0.002	0.050	0.32	0.13	0.20
Te Group	0.003	0.002	0.0	0.56	0.39	0.050
Ba/Sr Group	0.0006	0.0	0.0	0.005	0.005	0.020
Ru Group	<1.0e-6	0.0	0.0	0.006	0.003	0.003
Mo Group	1.9E-05	0.0	0.0	0.12	0.020	0.003
Lanthanides	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0005

- SAND2023-01313 and SAND2011-0128 utilized MELCOR
- Accident scenarios and modeling best-practices lead to tendency for increased early in-vessel halogen releases
- Peach Bottom and Grand Gulf modeling best-practices in SAND2023-01313 represent improvements due to SOARCA

BWR In-Containment Source Terms Consistent with SOARCA

- SOARCA found limited in-vessel halogen retention during early-in vessel phase

PB SOARCA halogen releases (STSBO without RCIC blackstart)



*In-containment source terms reported in SAND2023-01313 characterize **total** radioactive inventory in containment

PWR In-containment Source Term Evolution

MELCOR

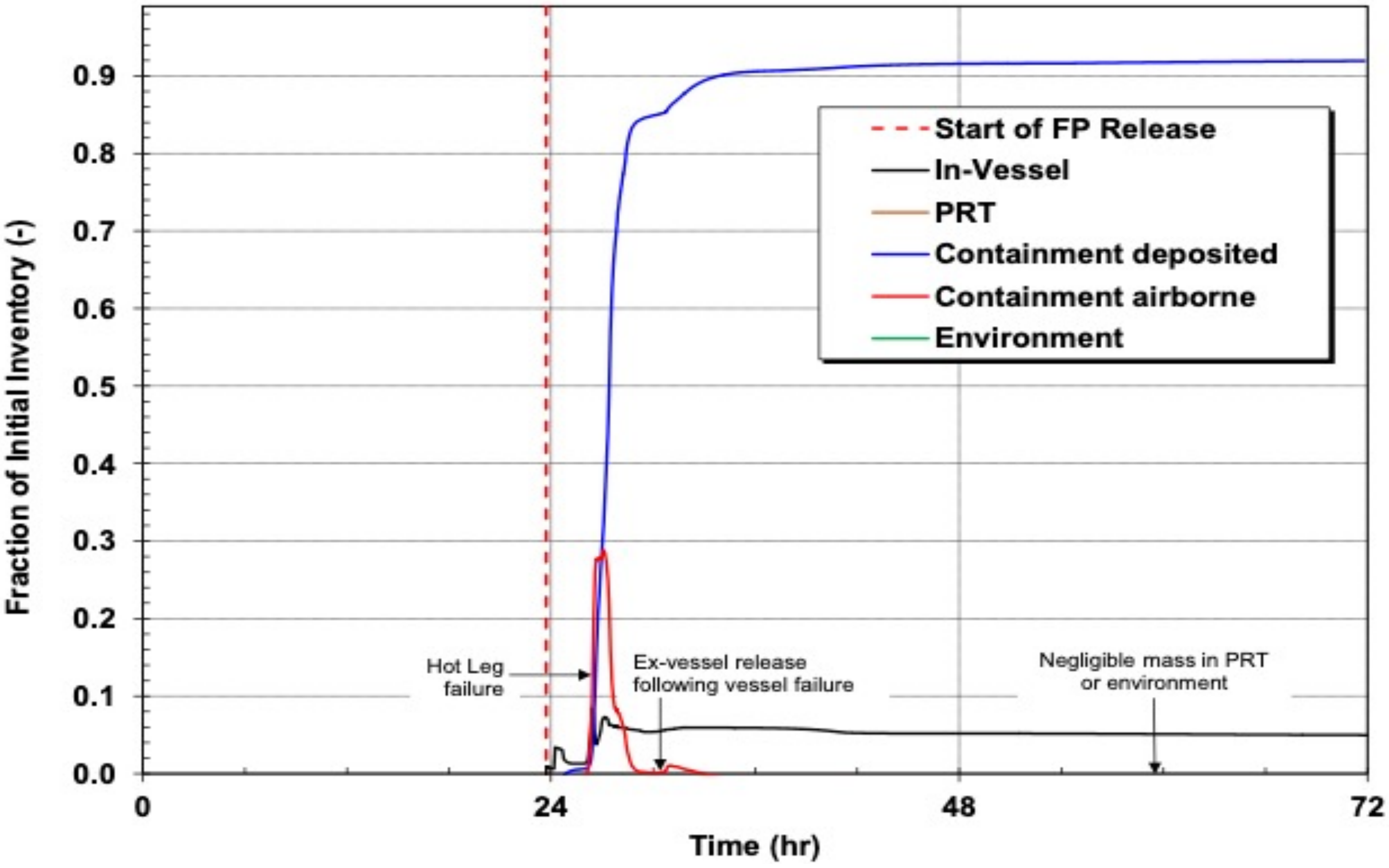
	Gap Release			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Phase Duration	1.3	0.22	0.50	4.0	4.5	1.3
Noble Gases	0.026	0.017	0.050	0.93	0.94	0.95
Halogens	0.007	0.004	0.050	0.58	0.37	0.35
Alkali Metals	0.003	0.003	0.050	0.50	0.23	0.25
Te Group	0.006	0.004	0.0	0.55	0.30	0.050
Ba/Sr Group	0.001	0.0006	0.0	0.002	0.004	0.020
Ru Group	<1.0e-6	0.0	0.0	0.008	0.006	0.003
Mo Group	2.0E-05	0.0	0.0	0.15	0.080	0.003
Lanthanides	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	0.0	0.0	<1.0e-6	<1.0e-6	0.0005

- SAND2023-01313 and SAND2011-0128 utilized MELCOR
- Accident scenarios and modeling best-practices lead to tendency for increased early in-vessel halogen releases
- Surry and Sequoyah modeling best-practices in SAND2023-01313 represent improvements due to SOARCA

PWR In-Containment Source Terms Consistent with SOARCA

- SOARCA found limited halogen in-vessel retention after hot leg creep rupture

SQN SOARCA halogen releases (LTSBO)



*In-containment source terms reported in SAND2023-01313 characterize **total** radioactive inventory in containment

In-containment Release Rate Evolution

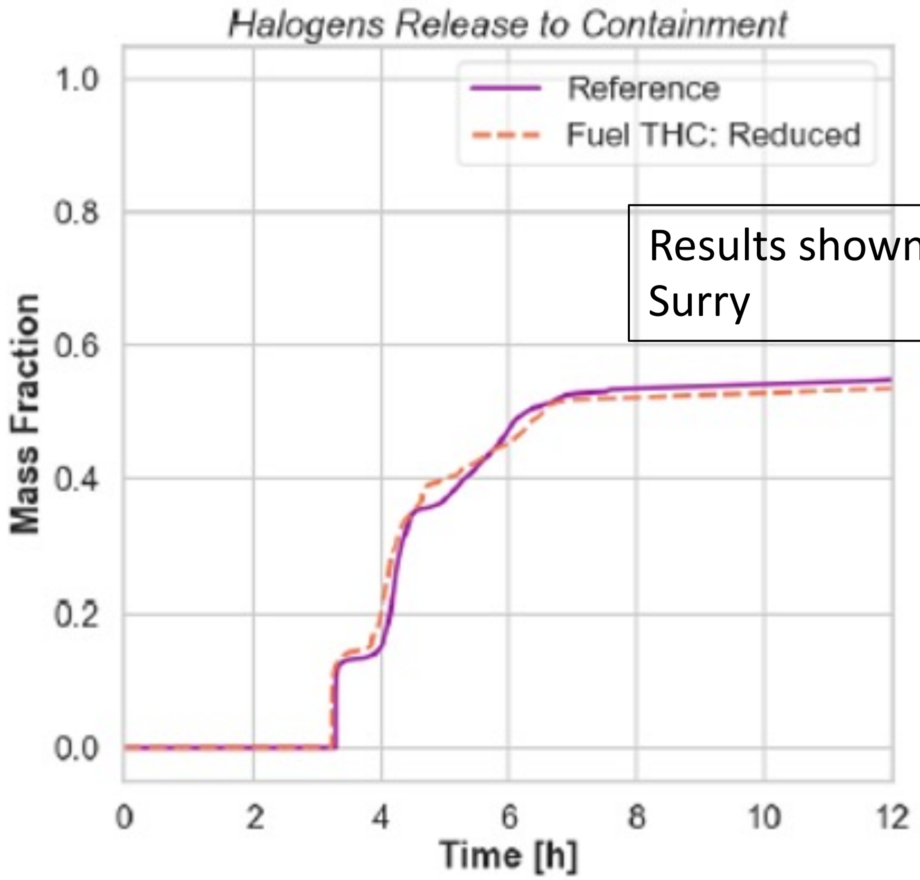
	BWR			PWR		
	Early In-vessel			Early In-vessel		
Study	**2023**	**2011**	**NUREG-1465**	**2023**	**2011**	**NUREG-1465**
Noble Gases	0.14	0.12	0.63	0.21	0.21	0.73
Halogens	0.11	0.059	0.17	0.16	0.082	0.27
Alkali Metals	0.047	0.016	0.13	0.15	0.051	0.19
Te Group	0.091	0.049	0.033	0.15	0.067	0.038
Ba/Sr Group	0.0009	0.0006	0.013	0.0007	0.0009	0.015
Ru Group	0.0009	0.0003	0.002	0.002	0.001	0.002
Mo Group	0.017	0.003	0.002	0.045	0.018	0.002
Lanthanides	<1.0e-6	<1.0e-6	0.0001	<1.0e-6	<1.0e-6	0.0002
Ce Group	<1.0e-6	<1.0e-6	0.0003	<1.0e-6	<1.0e-6	0.0004

Reported as [release fraction/hour]

Fuel Thermal Conductivity Sensitivity

Increased burnup leads to decrease of fuel thermal conductivity

Sensitivity Case	Fuel Thermal Conductivity [W/m-K]
Reference	4.92
Reduced	2.02
Low	0.2



Results shown for Surry

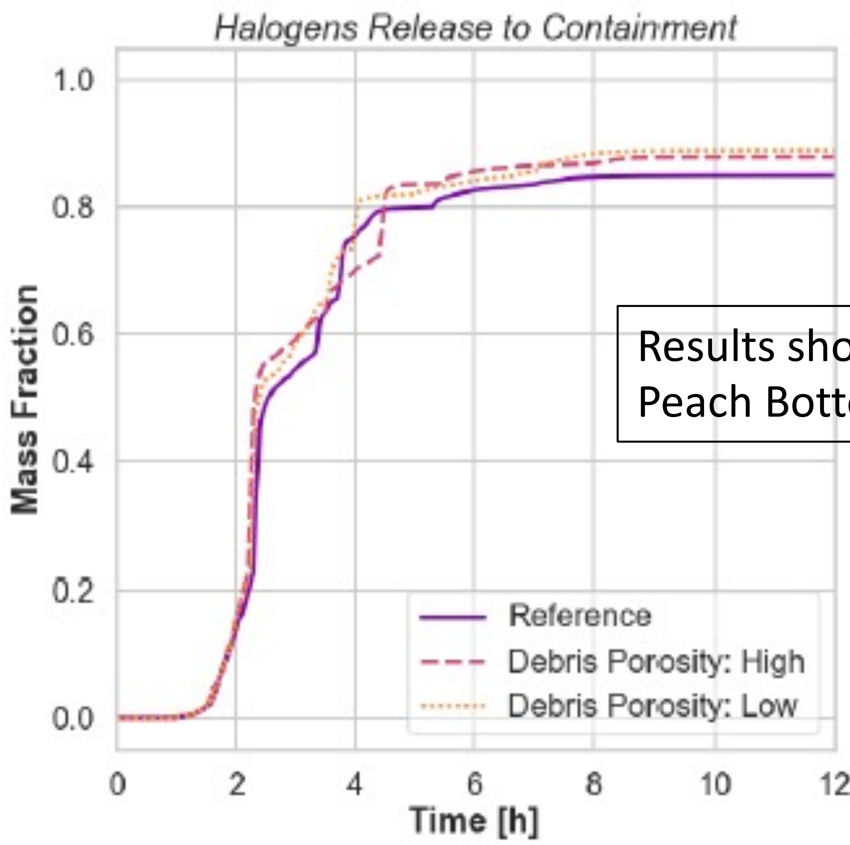
No impact from variation of fuel thermal conductivity

In-vessel Particulate Debris Porosity

Very high burnups have been postulated to promote disintegration of the fuel material

Three sensitivity cases to assess impact on in-containment source term

Sensitivity Case	In-Vessel Particulate Debris Porosity
Reference	0.4
High	0.6
Low	0.2



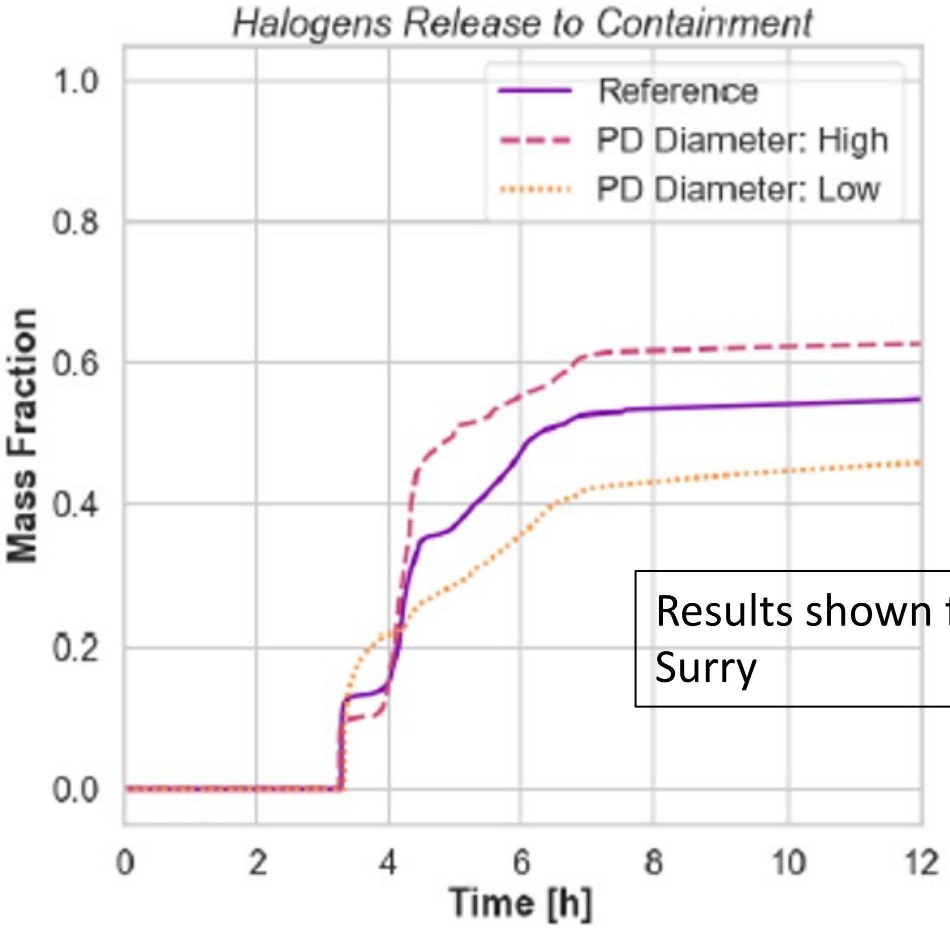
Results shown for Peach Bottom

No impact from variation of in-vessel particulate debris porosity

Diameter of In-vessel Particulate Debris Sensitivity

Higher burnups result in a greater degree of fuel breakup

Sensitivity	In-core Particulate Debris Diameter [cm]	Lower Plenum Particulate Debris Diameter [cm]
Reference	1.0	0.2
High	1.5	0.5
Low	0.5	0.1



Results shown for Surry

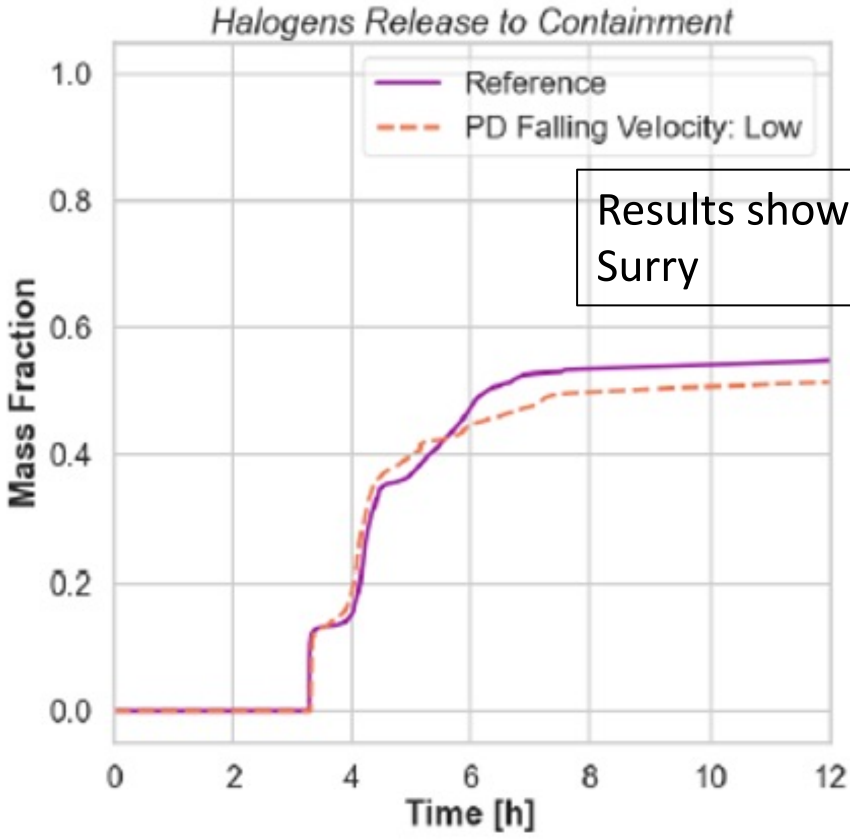
Variation in particulate debris diameter impacts in-containment source term

Impact smaller than changes across accident scenarios

Particulate Debris Falling Velocity Sensitivity

Particulate debris sizes could impact particulate debris fall velocity into lower plenum

Sensitivity	In-Vessel Particulate Debris Fall Velocity [m/s] *Peach Bottom*	In-Vessel Particulate Debris Fall Velocity [m/s] *Surry*
Reference	0.94	0.094
Low	0.094	0.064



Results shown for Surry

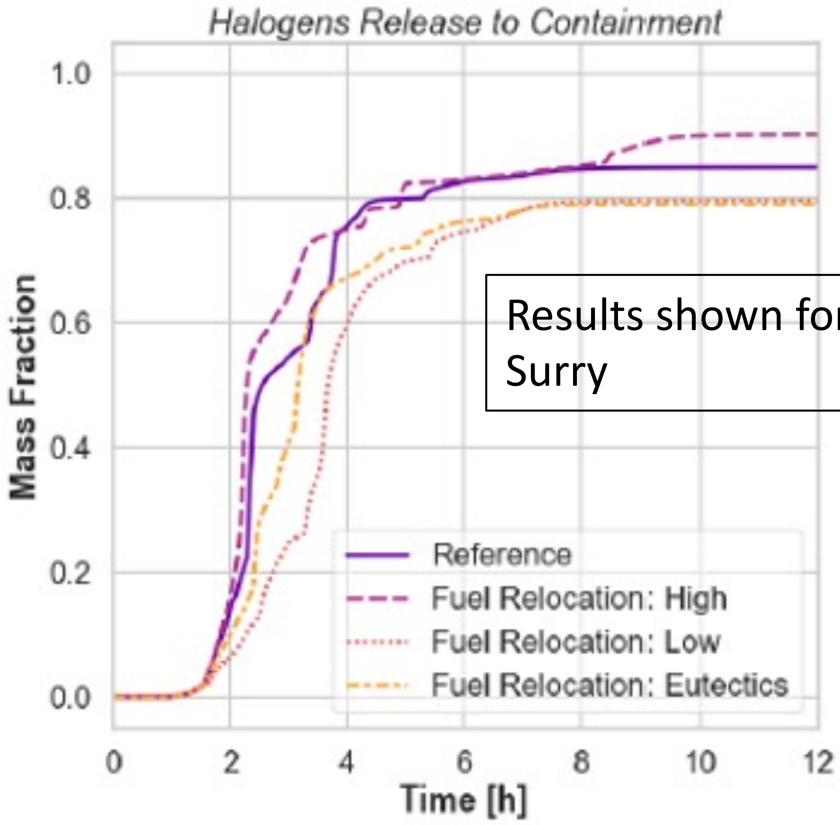
No impact on source term due to variation in particulate debris fall velocity

Fuel Relocation Temperature Sensitivity

Material interactions can cause early failure of fuel assemblies and other core components

- MELCOR uses either the interactive materials model or eutectics model to represent material interactions

Sensitivity	Fuel Relocation Temperature [K]
Reference	2479
High	2728
Low	2230
Eutectics	Eutectics model



Results shown for Surry

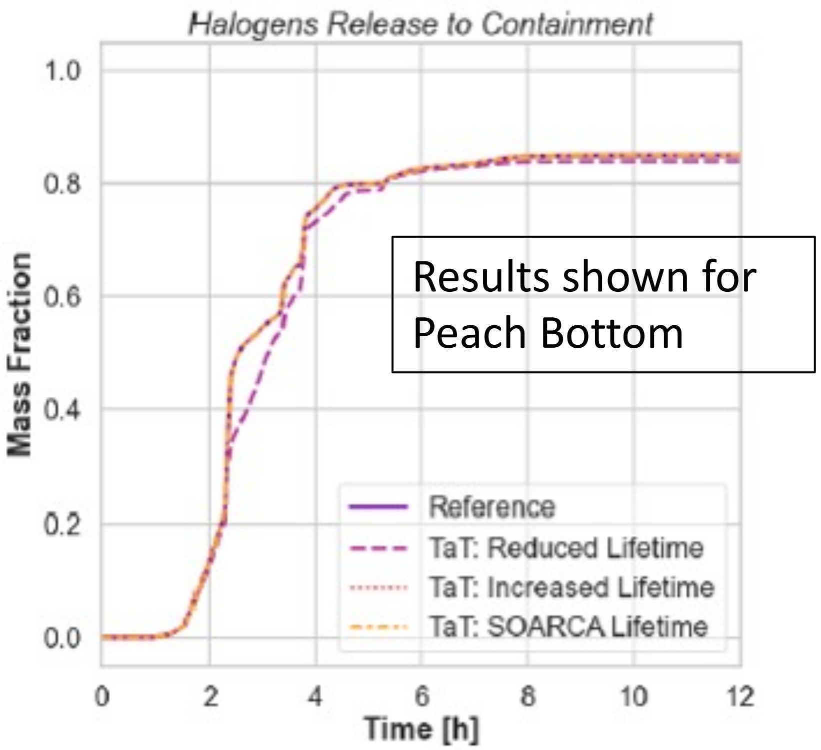
Material interactions that cause early fuel failure and can impact accident progression timings and in-containment source terms based on SOARCA uncertainty studies

Fuel Rod Lifetime Sensitivity

Fuel assemblies at high temperatures exhibit early failures

- Early failures captured in MELCOR simulations using a lifetime function

Sensitivity	Fuel Rod Lifetime Model
Reference	Default time-at-temperature model
Increased Lifetime	Lifetime function that accrues damage from 22.2 hours to 20 minutes at temperatures from 2100K – 2600K
Reduced Lifetime	Lifetime function that accrues damage from 1.67 hours to 3.3 minutes at temperatures from 2100K – 2600K
SOARCA Lifetime	Lifetime function that accrues damage from 10 hours to 5 minutes at temperatures from 2100K – 2600K

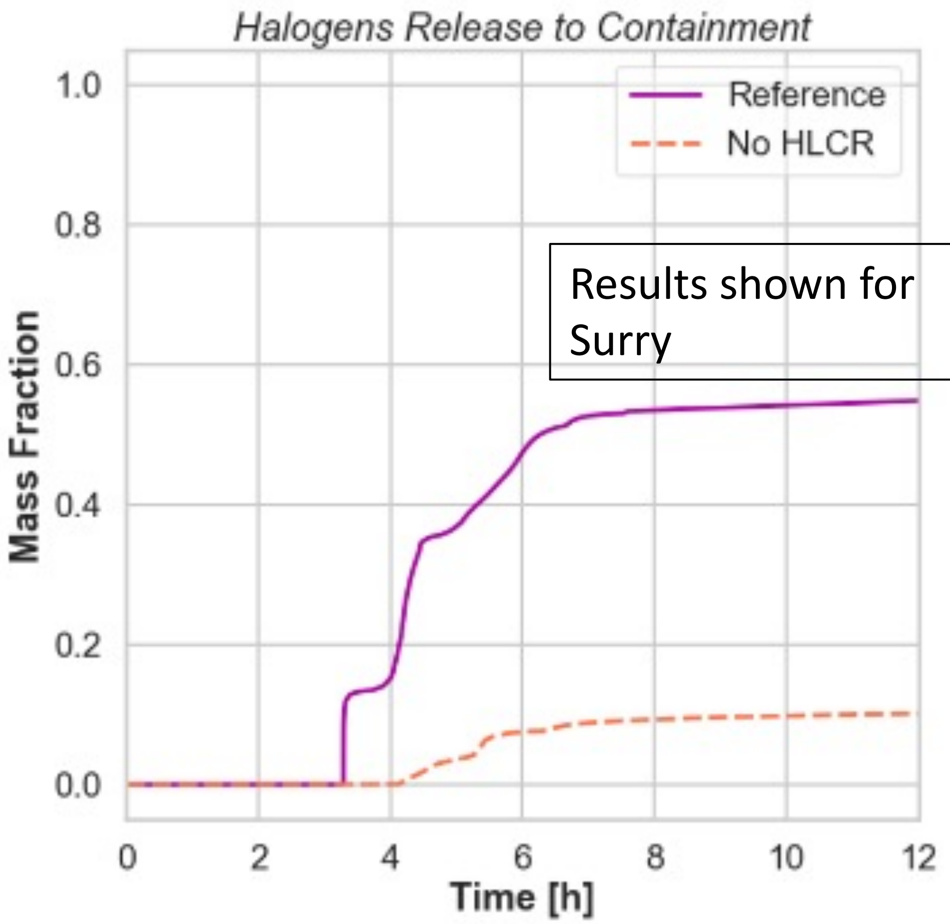


No impact due to variation of the fuel rod lifetime modeling on source term
Oxidized fuel assembly temperature failure model generally dominates

Hot Leg Creep Rupture Sensitivity

Key insight from SOARCA is potential for induced RPV pressure boundary failures

- Severe accident conditions lead to high pressure and temperature conditions at RPV boundary
- Thermally-induced hot leg creep rupture found likely for PWRs
- BWRs exhibited thermally-induced seizure of cycling SRVs

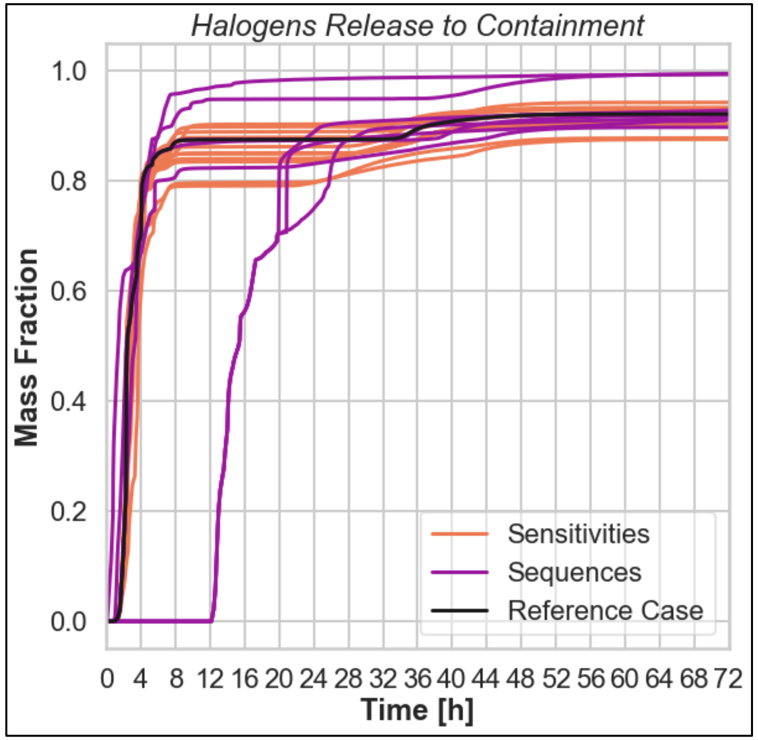


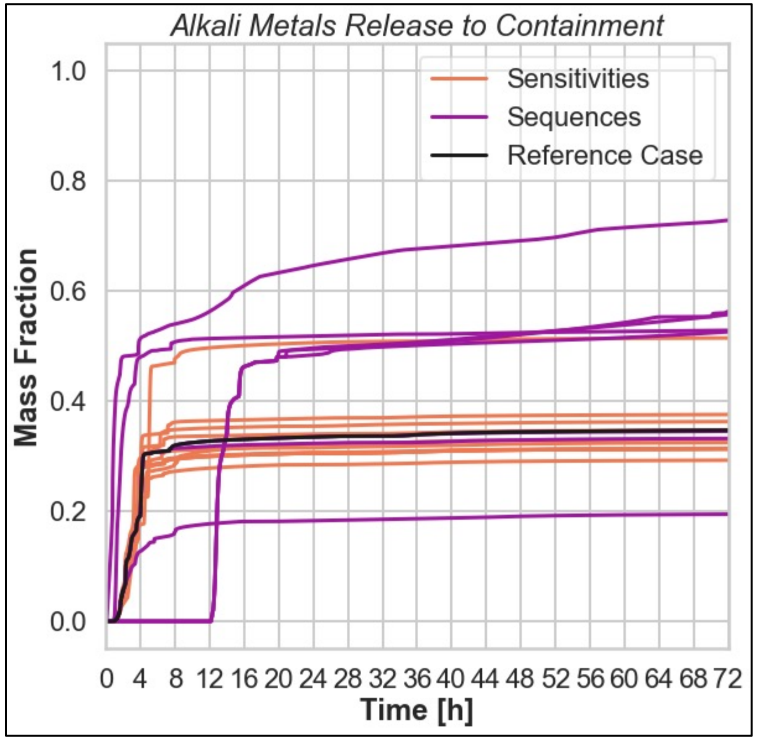
Results shown for Surry

Sensitivity	RPV Induced Pressure Boundary Failure Modeling
Reference	Hot leg creep rupture enabled
No HLCR	Hot leg creep rupture disabled

Significant increase in early in-vessel source term for induced RPV failure for SBOs

In-containment Source Term Variability



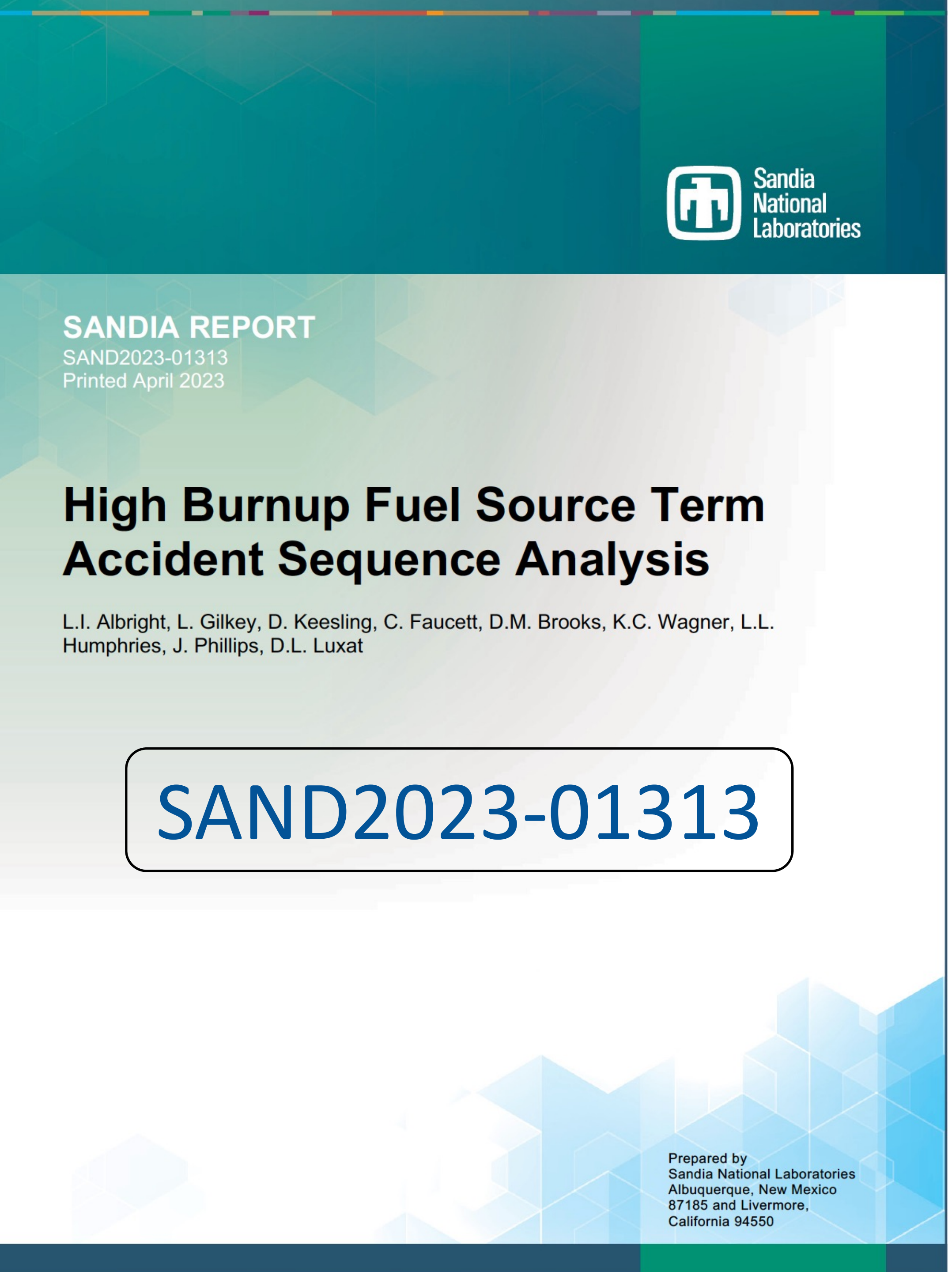


Peer Review Finding

- Potential for combined effects of various sensitivity studies to be larger than separate effects

Nonlinear processes in severe accidents tend to limit amplification of response variability in multi-parameter sensitivity studies such that single scenario variability is less than variation across scenarios

In-containment source term variation dominated by variation across sequences


Sandia National Laboratories

SANDIA REPORT
SAND2023-01313
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High Burnup Fuel Source Term Accident Sequence Analysis

L.I. Albright, L. Gilkey, D. Keesling, C. Faucett, D.M. Brooks, K.C. Wagner, L.L. Humphries, J. Phillips, D.L. Luxat

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico
87185 and Livermore,
California 94550

SAND2023-01313

Summary

- Increased burnup or extended enrichment does not significantly impact source term
  - Most significant variation in source term arises due to differences between accident scenarios
- Status of RPV has significant impact on early in-vessel releases
  - Low pressure scenarios exhibit more significant releases to containment
  - NUREG-1465 prescribed larger number of high pressure scenarios than SAND2023-01313
- Early in-vessel source term greatly reduced if RPV pressure boundary intact

Independent Peer Review

Focus of SAND2023-01313 Peer Review

MELCOR

ERI/NRC 23-201

PEER REVIEW OF THE IN-CONTAINMENT SOURCE TERM STUDY FOR HIGH-BURNUP AND HIGH-ASSAY LOW ENRICHED URANIUM FUELS

ERI/NRC 23-201

Work Performed under the Auspices of the
United States Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, D.C. 20555

ENERGY RESEARCH INC
P. O. Box 2034
Rockville, Maryland 20847

April 2023

- Review technical basis of SAND2023-01313
- Recommend improvements to SAND2023-01313
- Assess suitability of SAND2023-01313 source terms for regulatory applications

Peer Review Organization

Panel Membership

- Dr. Mohsen Khatib-Rahbar – Panel Chair
  - Energy Research, Inc. (ERI)
- Dr. Richard S. Denning
  - Consultant
- Mr. Jeff Gabor
  - Jensen Hughes
- Dr. Didier Jacquemain
  - Organization for Economic Co-operation and Development/Nuclear Energy Agency (OECD/NEA)
- Dr. Luis E. Herranz
  - Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT)
- Dr. Yu Maruyama
  - Japan Atomic Energy Agency (JAEA)

Panel Objectives

- Assess technical adequacy with respect to:
  - Overall analysis approach
  - Specific applications of MELCOR to development of in-containment source terms
- Assess appropriateness of severe accident sequences selected
- Assess applied models and assumptions in terms of
  - Current understanding of severe accidents and source terms
  - Adequacy considering available experimental data, and observations
- Assess that source terms are representative, rather than conservative or bounding
- Assess adequacy of documentation against
  - Completeness of technical bases specification
  - Approach to analysis of uncertainties

Peer Review Process

- Draft High Burnup Fuel Source Term Accident Sequence Analysis (Completed 2021)
- Virtual Meetings (began in 2022)
  1. Briefing on the peer review objectives and the draft report by NRC and SNL
     - Panelist review reports delivered to SNL
     - Preliminary resolution of comments by SNL
     - Preparation of the draft peer review report
  2. Discussion of draft peer review report, comment resolution, and summary of unresolved comments
     - Final resolution of comments by SNL
     - Revision of High Burnup Fuel Source Term Accident Sequence Analysis report
  3. Discussion of revised report, peer review panel findings, and conclusions
     - Final High Burnup Fuel Source Term Accident Sequence Analysis report released (2023)
     - Final peer review report released (2023)

Acceptability of the SAND2023-01313 Source Term

- “[The peer review panel] **endorses the approach** taken in [SAND2023-01313]”
- “[SAND2023-01313] provides a **defendable technical basis** for the proposed source terms”
- “The peer review panel finds that the four nuclear power plants considered in the [SAND2023-01313] **reasonably represent the U.S. nuclear fleet**”
- “The **spectrum of accidents is sufficient** to satisfy the following stated attributes of an acceptable alternative accident source term (RG 1.183):

    The accident source term must be expressed in terms of times and rates of appearance of radioactive fission products released into containment, the types and quantities of the radioactive species released, and the chemical forms of iodine released.

Qualities of the SAND2023-01313 Source Term

- Study is a significant technical improvement using state-of-the-art methods implemented in latest version of MELCOR
- In-containment source terms for HBU/HALEU fuels are representative MELCOR estimates, rather than conservative or bounding estimates
- No bias in the approach identified that could overestimate in-containment source terms
- Sensitivity studies documented in SAND2023-01313 valuable in supporting applications
  - Sensitivities explored limitations in understanding of HBU/HALEU fuel response under severe accident conditions
  - Results demonstrated impact of thermally induced (creep) depressurization of RCS for PWRs on in-containment source terms

Peer Review Report Recommendations

- Gap release phase incorporated into the early in-vessel phase
  - *The panel considers the current approach of separating the gap and early in-vessel release phases, a product of the simplified single channel treatment of the STCP models of circa 1980s that is reflected in the NUREG-1465 source terms, outdated. During severe accidents, the gap and in-vessel releases from the fuel overlap to the extent that it is not possible to truly separate the two as distinct phases. Therefore, it is recommended that the gap release be incorporated into the early in-vessel release phase.*
- More appropriate to represent impact of burnup using core inventories for HBU expressed in terms of radiological activities
  - *The implication of comparison of mass inventories in kilogram [SAND2023-01313] is to incorrectly conclude that at higher fuel burnups, off-site doses would likely be substantially higher for high burnup fuels as the direct result of larger core mass inventories of radionuclides. In fact, when compared on the basis of integrated radiological activity, there would not be any significant differences for the two levels of fuel burnup.*
  - Examples shown in the next presentation: “Follow-on Calculations”

Other Comments And Recommendations

- Panelists requested additional clarification (reflected in final report) that
  - Containment bypass scenarios and air ingression not considered in development of tabular source terms
  - Fission product removal mechanisms in containment not included in tabular source terms
    - Captured in MELCOR simulations, but post-processed out of reported MELCOR source terms
- Peer reviewers acknowledged more recent PRA studies could have different contributors to core damage
  - For the intended applications the scenarios used in the current [SAND2023-01313] appropriate with regards to the progression of severe accidents, radionuclide release and transport
- Panelists noted for most radionuclides no increase in activity with burnup sufficient to impact siting calculations
- Peer reviewers noted the uncertainty in Iodine speciation based on experiments (FPT3, DF-4, and BECARRE)
- Peer review noted that current Fukushima Daiichi post-accident analyses confirm the assumption that Cs_2MoO_4 is dominant chemical form of Cs
- Peer review panel considered the use of median estimates appropriate to avoid bias due to potential outliers

Other Comments And Recommendations

Finally, even though tabular severe accident in-containment source terms provide a simplified tool for regulatory applications and analyses, it is important to recognize their limitations and the panel encourages the direct application of a state-of-the art severe accident code to specific issues when appropriate.

Fission Product Retention in Suppression Pools

MELCOR

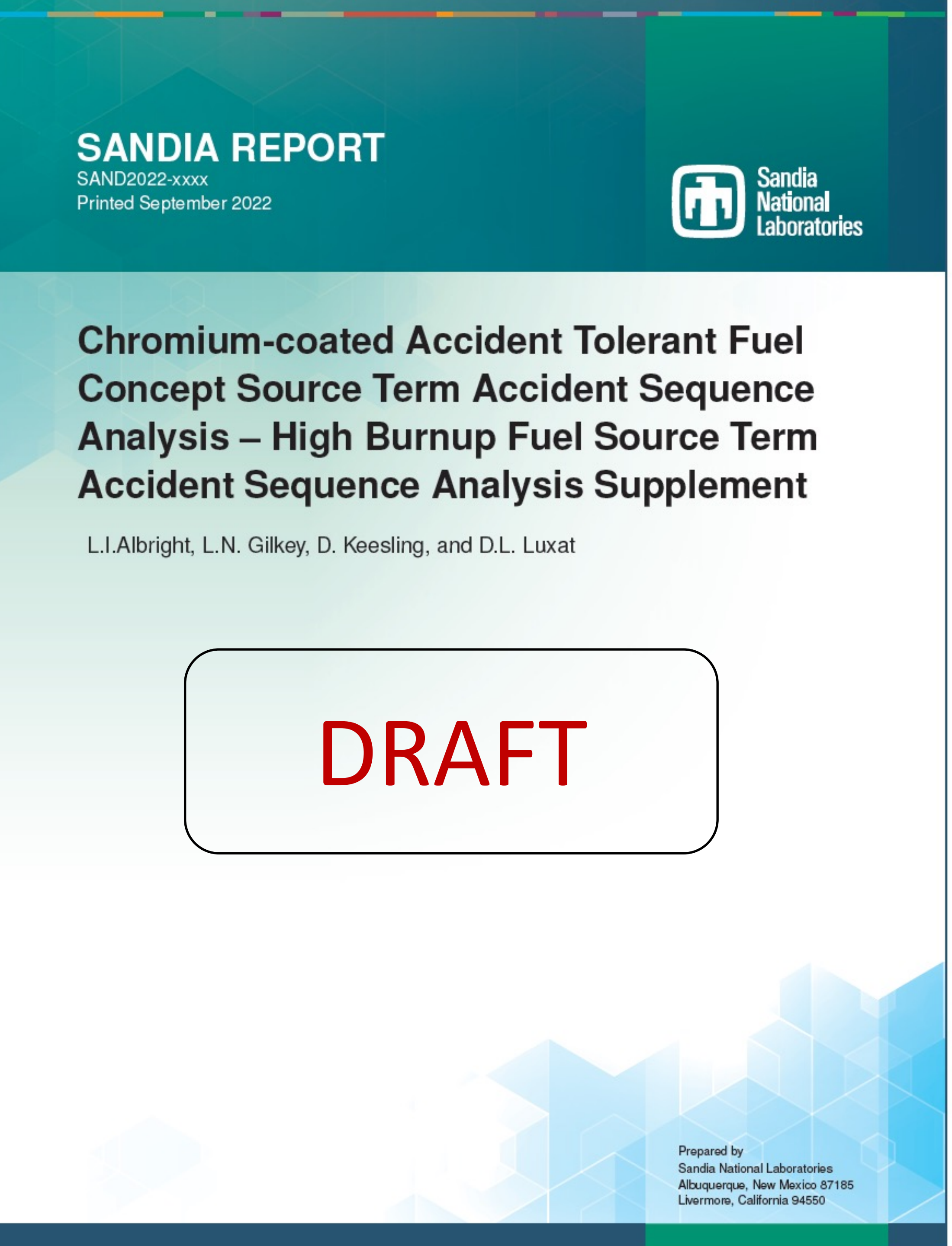
	Gap Release		Early In-vessel	
Release Category	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory
Noble Gases	0.016	0.016	0.95	0.95
Halogens	0.005	1.30E-06	0.71	0.06
Alkali Metals	0.005	1.20E-06	0.32	0.006
Te Group	0.003	<1.0e-6	0.56	0.038
Ba/Sr Group	0.0006	<1.0e-6	0.005	0.0003
Ru Group	<1.0e-6	<1.0e-6	0.006	7.40E-06
Mo Group	1.90E-05	<1.0e-6	0.12	0.0001
Lanthanides	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6
Ce Group	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6

Peer Review Findings

- In-containment source terms should consider the impact of retention in suppression pools, especially for SBO scenarios that discharge directly into the suppression pool
- Estimates of retention in suppression pools provided in SAND2023-01313 could be used in regulatory guidance to establish suppression pool decontamination factors

Significant effect of retention in suppression pool on key radionuclide groups

Upcoming Work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Chromium-coated Accident Tolerant Fuel Concept Source Term Accident Sequence Analysis – High Burnup Fuel Source Term Accident Sequence Analysis Supplement

L.I.Albright, L.N. Gilkey, D. Keesling, and D.L. Luxat

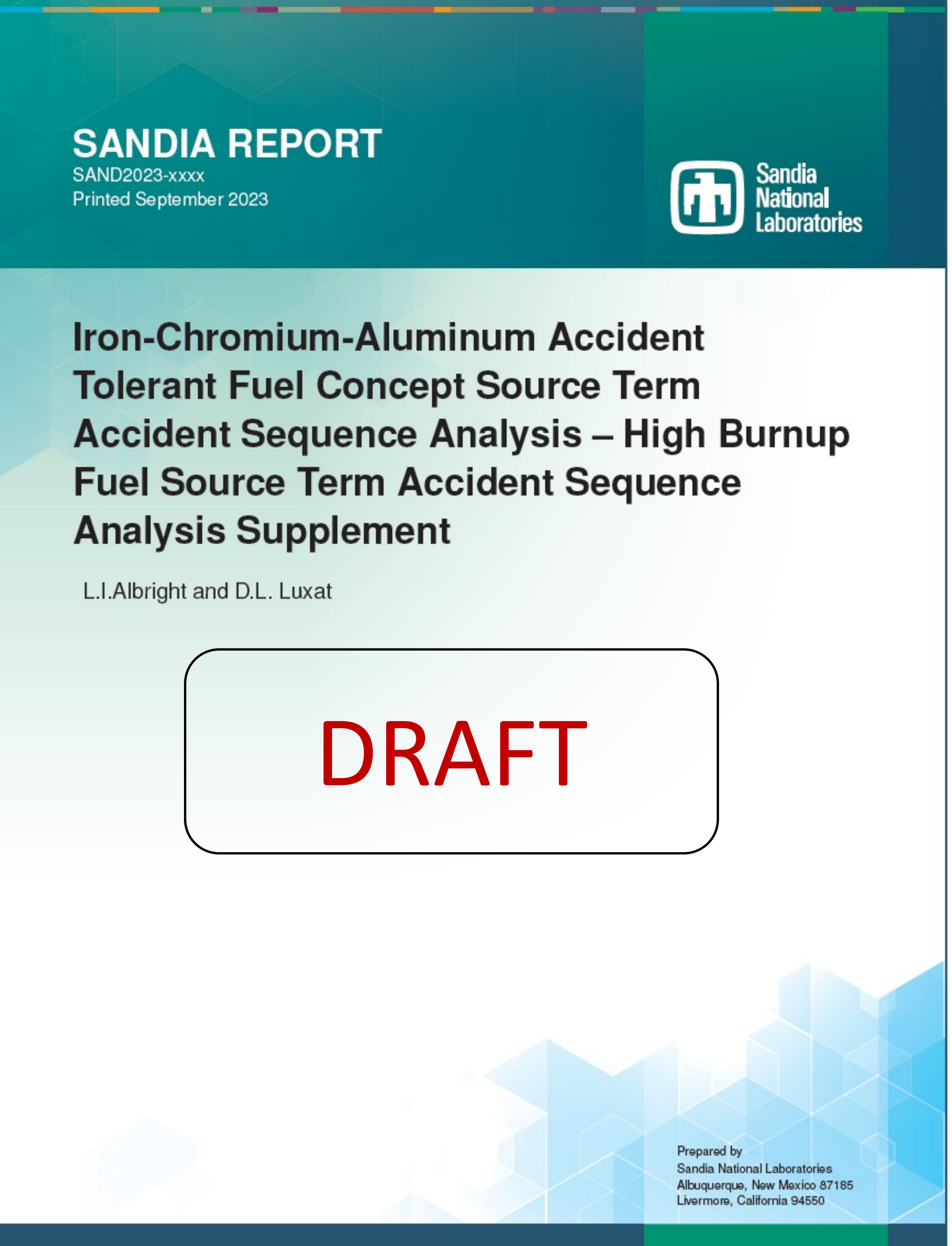
DRAFT

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185
Livermore, California 94550

Cr-Coated ATF Concept

- Cr-coated ATF concept most similar to conventional fuels
  - Thin, protective chromium coating on Zircaloy fuel cladding delays exothermic Zircaloy oxidation onset
- Cr-coated analysis informed by ATF severe accident PIRT (NUREG/CR-7283) findings

MELCOR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Sandia National Laboratories

Iron-Chromium-Aluminum Accident Tolerant Fuel Concept Source Term Accident Sequence Analysis – High Burnup Fuel Source Term Accident Sequence Analysis Supplement

L.I.Albright and D.L. Luxat

DRAFT

Prepared by
Sandia National Laboratories
Albuquerque, New Mexico 87185
Livermore, California 94550

FeCrAl ATF Concept

- FeCrAl ATF concept utilizes a novel fuel cladding material
  - Substitution of Zr-based alloy with an FeCrAl alloy
  - Intended to reduce both oxidation in the core and associated hydrogen production
- FeCrAl analysis informed by ATF severe accident PIRT (NUREG/CR-7283) findings
  - Sensitivity analyses deployed to interrogate FeCrAl cladding knowledge uncertainties

Thank you for your attention!

MELCOR

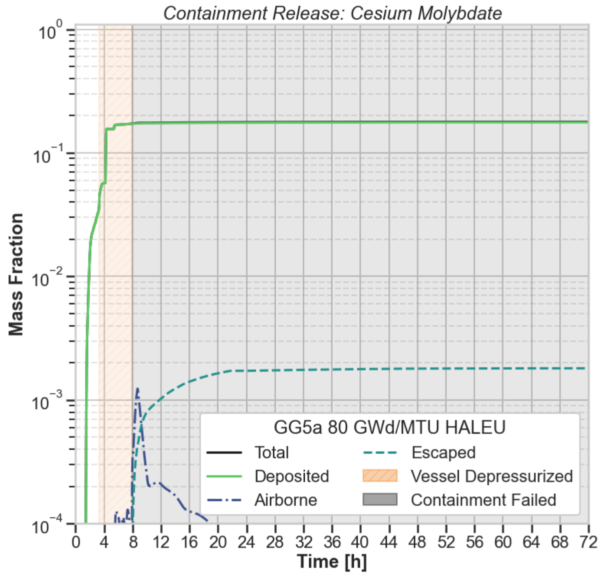
Backup Slides

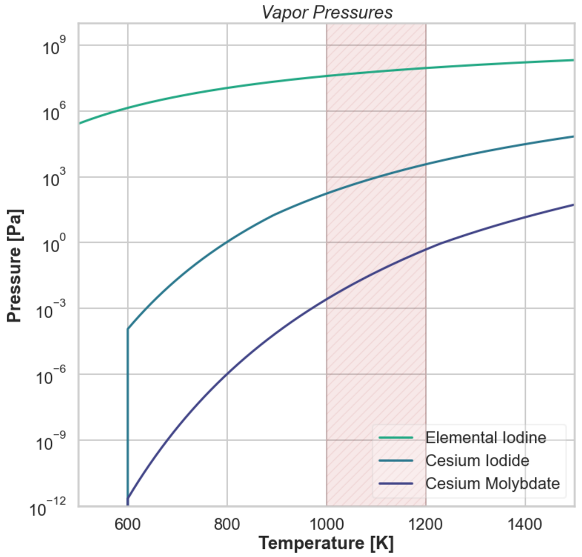
Acronyms

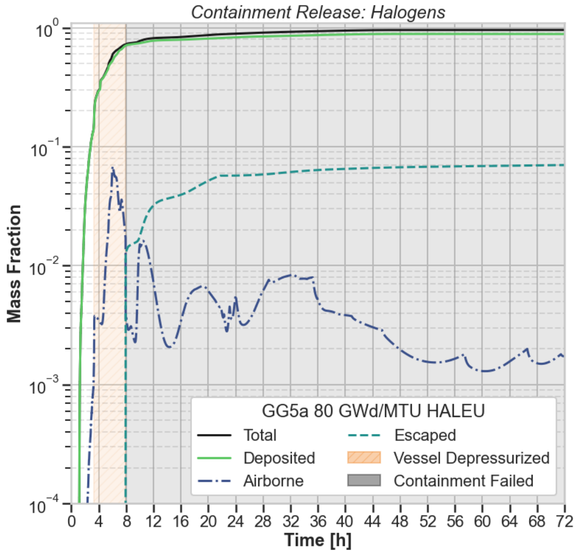
Acronym	Definition	Acronym	Definition
AC	Alternating current	NRC	Nuclear Regulatory Commission
ADS	Automatic depressurization system	ORNL	Oak Ridge National Laboratories
AFW	Auxiliary Feedwater	PB	Peach Bottom
AST	Alternative source term	PIRT	Phenomena Identification and Ranking Table
ATF	Accident tolerant fuel	PORV	Pilot-operated relief valve
ATWS	Anticipated transient without scram	PRA	Probabilistic risk assessment
BWR	Boiling water reactor	PRT	Pressurizer relief tank
CCFL	Counter current flow	PWR	Pressurized water reactor
CDF	Core damage frequency	QoI	Quantity of interest
DC	Direct current	RCIC	Reactor core isolation cooling system
ECCS	Emergency core cooling system	RCP	reactor coolant pump
ECDF	Empirical cumulative distribution function	RCS	Reactor coolant system
GG	Grand Gulf	RHR	Residual heat removal
HALEU	High-assay low-enriched uranium	SBLOCA	Small-break loss of coolant accident
HBU	High burnup	SBO	Station blackout
HLCR	Hot leg creep rupture	SOARCA	State-of-the-Art Reactor Consequence Analyses
HPCI	High pressure coolant injection system	SQN	Sequoyah
HPSI	High-pressure safety injection	SRV	Safety relief valve
LBLOCA	Large-break loss of coolant accident	STCP	Source Term Code Package
LEU	Low-enriched uranium	STSBO	Short-term station blackout
LOCA	Loss of coolant accident	SU	Surry
LPCI	Low-pressure coolant injection	TDAFW	Turbine-driven auxiliary feedwater
LPSI	Low-pressure safety injection	TID	Technical information document
LTSBO	Long-term station blackout	TMI-2	Three Mile Island Unit-2
LWR	Light water reactor		

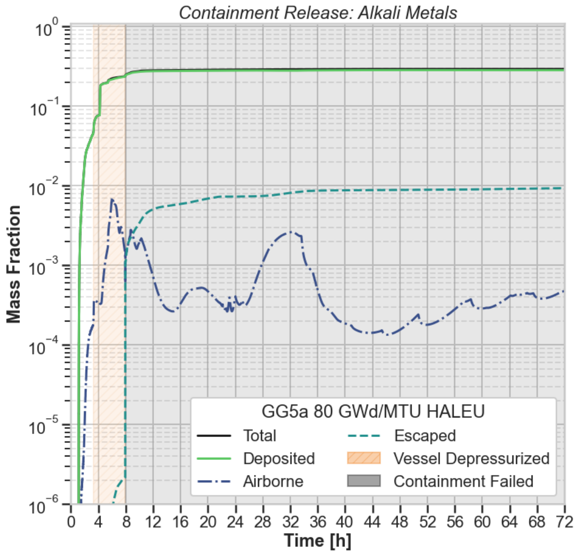
Cs and I Releases

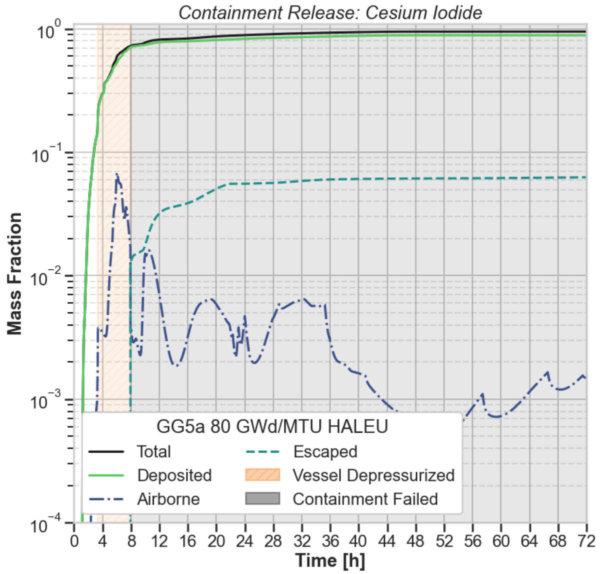
- SAND2011-0128 considered deposition of radionuclides on the lower head, leading to significantly decreased in-vessel phase releases.
  - This consideration delays a significant fraction of radionuclide release to containment until after lower head failure during the ex-vessel phase (employed for Peach Bottom and Sequoyah)
  - This practice is no longer considered appropriate, and was not employed in SAND2023-01313
- CsI (all original I inventory and ~10% original Cs inventory) transports readily from the primary system to containment during core damage due to the relatively large CsI vapor pressures at elevated primary system temperatures
  - Consistent with Peach Bottom SOARCA results











NUREG-1465 Accident Selection

MELCOR

- Dominant sequences were chosen based on impact on source term
  - PWRs are predominantly LOCA accidents
  - BWRs are predominantly SBO/ATWS accidents

PWR Plants	Sequence	Description
Surry	AG	LOCA (hot leg), no containment hear removal systems
	TMLB	LOOP, no PCS and no AFWS
	V	Intefacing system LOCA
	S3B	SBO with RCP seal LOCA
	S2D-δ	SBLOCA, no ECCS and H2 combustion
	S2D-β	SBLOCA w/ 6" hole in containment
Oconee 3	TMLB	SBO, no active ESF systems
	S1DCF	LOCA (3"), no ESF systems
Sequoyah	S3HF1	LOCA RCP, no ECCS, no CSRS w/ reactor cavity flooded
	S3HF2	S3HF1 w/ hot leg induced LOCA
	3HF2	S3HF1 w/ dry reactor cavity
	S3B	LOCA (1/2") w/ SBO
	TBA	SBO induces hot leg LOCA - H2 burn fails containment
	ACD	LOCA (hot leg), no ECCS no CS
	S3B1	SBO delayed 4 RCP seal failures, only steam driven AFW operates
	S3HF	LOCA (RCP seal), no ECCS no CSRS
	S3H	LOCA (RCP seal) no ECCS recirculation

BWR Plants	Sequence	Description
Peach Bottom	TC1	ATWS w/ reactor depressurized
	TC2	ATWS w/ reactor pressurized
	TC3	TC2 with wetwell venting
	TB1	SBO with battery depletion
	TB2	TB1 with containment failure at vessel failure
	S2E1	LOCA (2"), no ECCS and no ADS
	S2E2	S2E1 with basaltic concrete
	V	RHR pipe failure outside containment
	TBUX	SBO with loss of all DC power
LaSalle	TB	SBO with late containment failure
Grand Gulf	TC	ATWS early containment failure fails ECCS
	TB1	SBO with battery depletion
	TB2	TB1 w/ H2 burn fails containment
	TBS	SBO, no ECCS but reactor depressurized
	TBR	TBS with AC recovery after vessel failure

High Burnup Fuel Source Term Accident Analysis
Boiling-Water Reactor Follow-On Calculations

ACRS Radiation Protection And Nuclear Materials Subcommittee Briefing

November 16, 2023

Shawn Campbell and Michael Salay
Fuel & Source Term Code Development Branch
Division of Systems Analysis
Office of Nuclear Regulatory Research

Background and Motivation

- *The High Burnup (HBU) Peer Review panelists commented on the potential impact of the suppression pool on the containment source term.*
- *Table 5-16 of SAND2023-01313 provides the boiling-water reactor (BWR) containment release fractions including and excluding the suppression pool.*
- *Supplemental investigations following the peer review in BWRs:*
  - *Investigate fission product concentration variation between different regions of the reactor system and containment since some scenarios and pathways bypass the suppression pool (e.g., main steam line).*
  - *Modified the two (Peach Bottom, Grand Gulf) full-scale BWR input decks to better capture aerosol behavior in the containment and steam line.*
  - *Performed a set of BWR source term calculations.*

Source Term Methodology

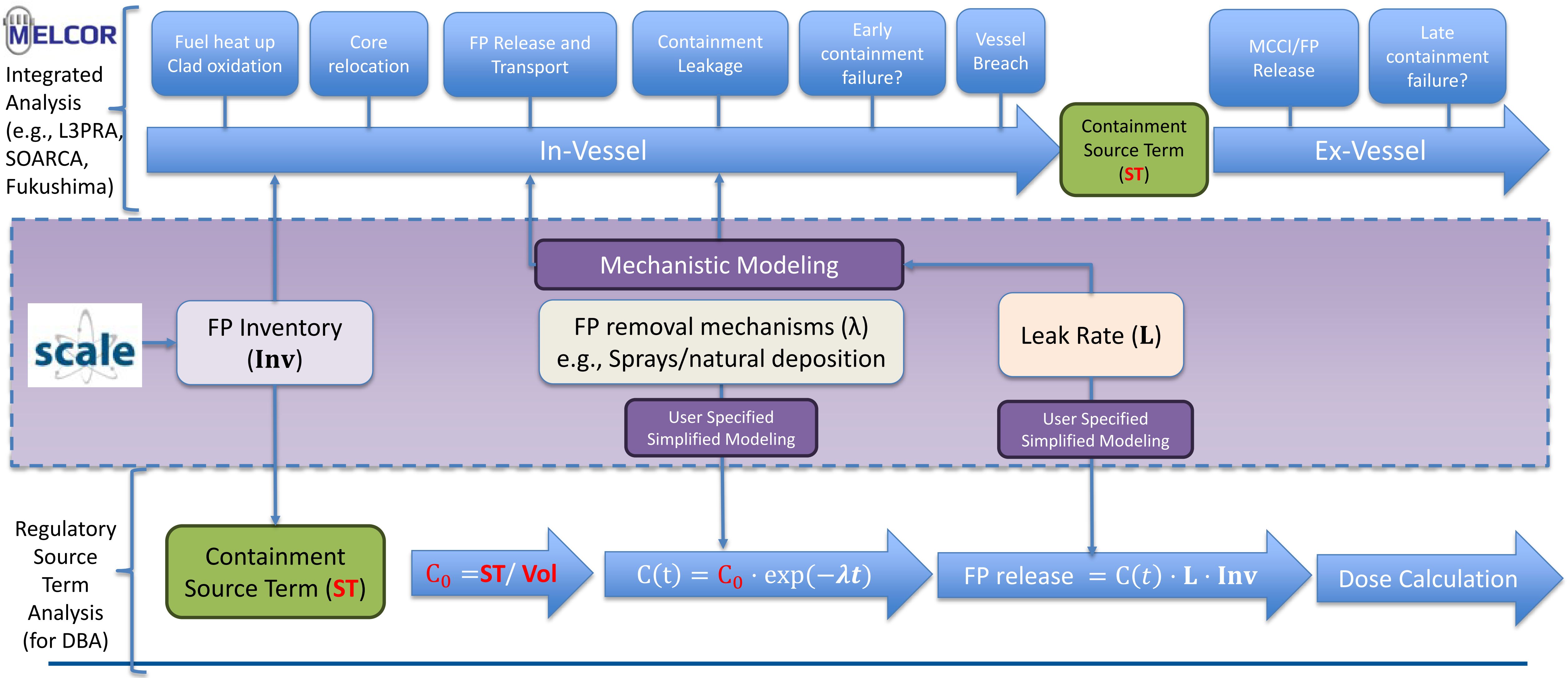
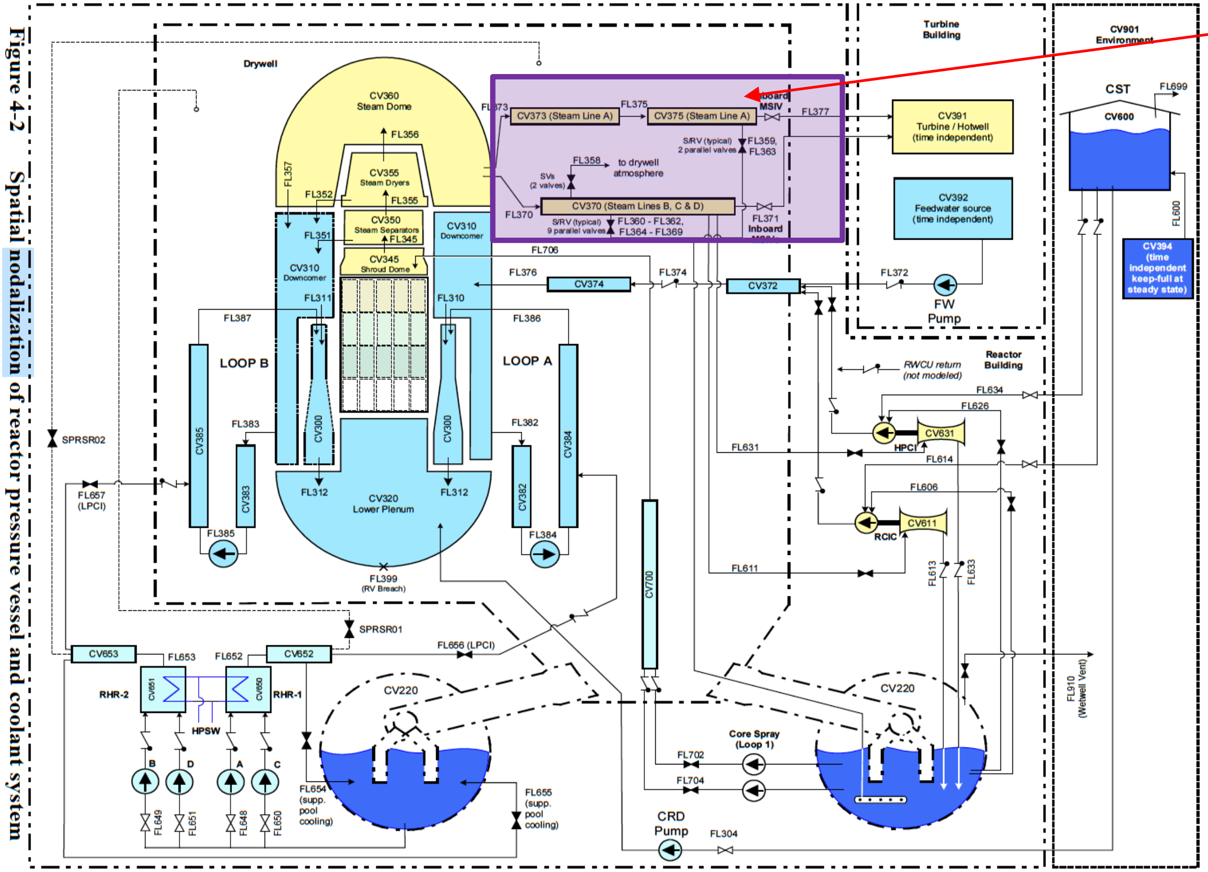
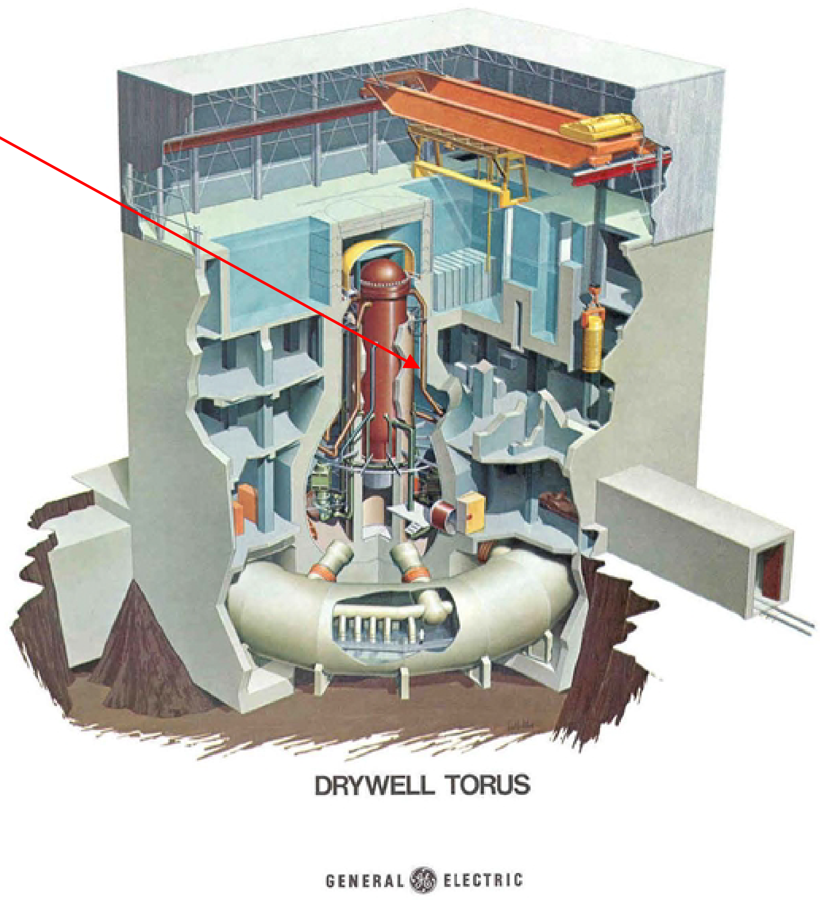


Illustration of BWR Modeling Practices

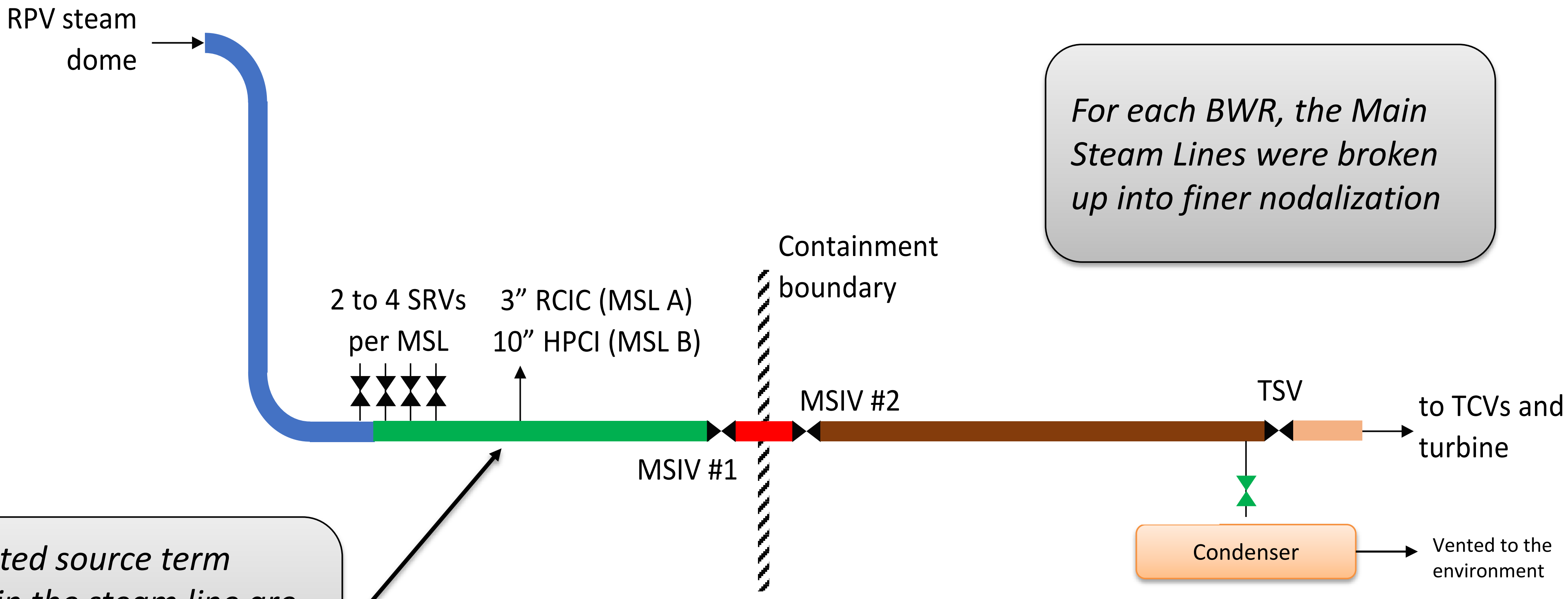
Area with refined modeling



Peach Bottom



New BWR Main Steam Line (MSL) Modeling



For each BWR, the Main Steam Lines were broken up into finer nodalization

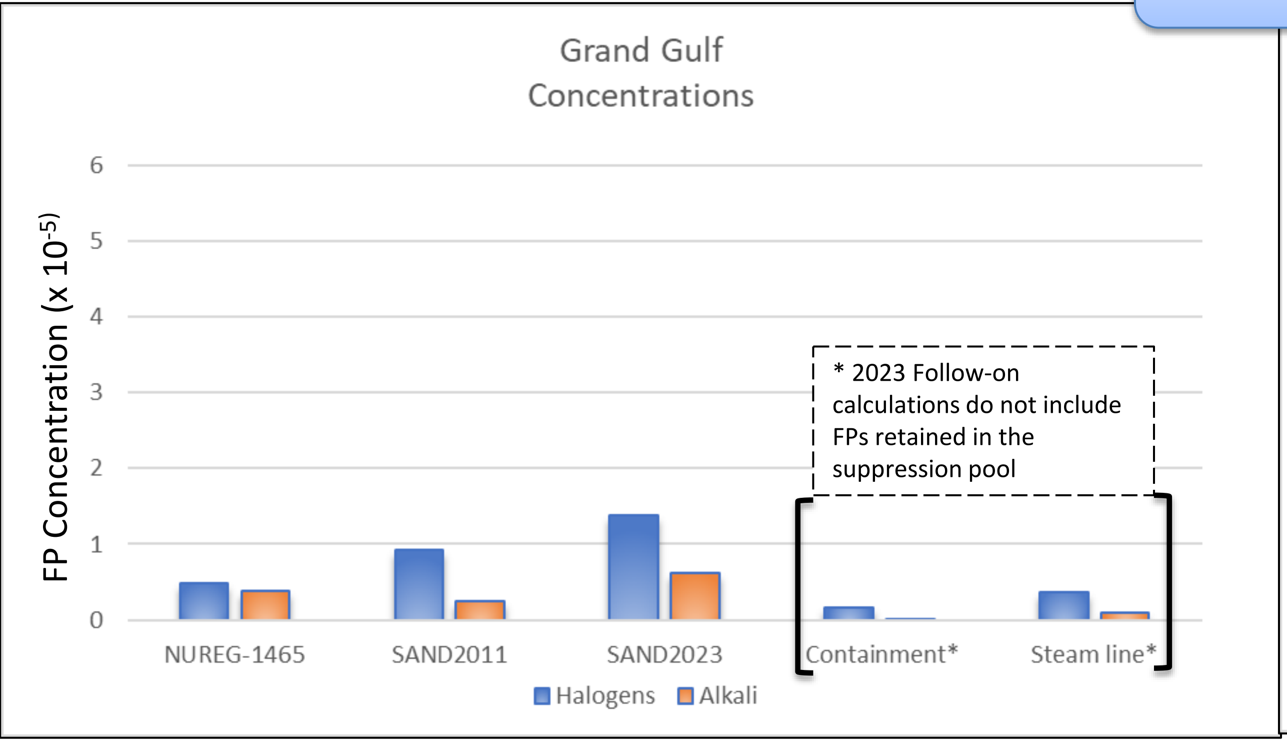
The reported source term fractions in the steam line are averaged airborne fission products in the green portion.

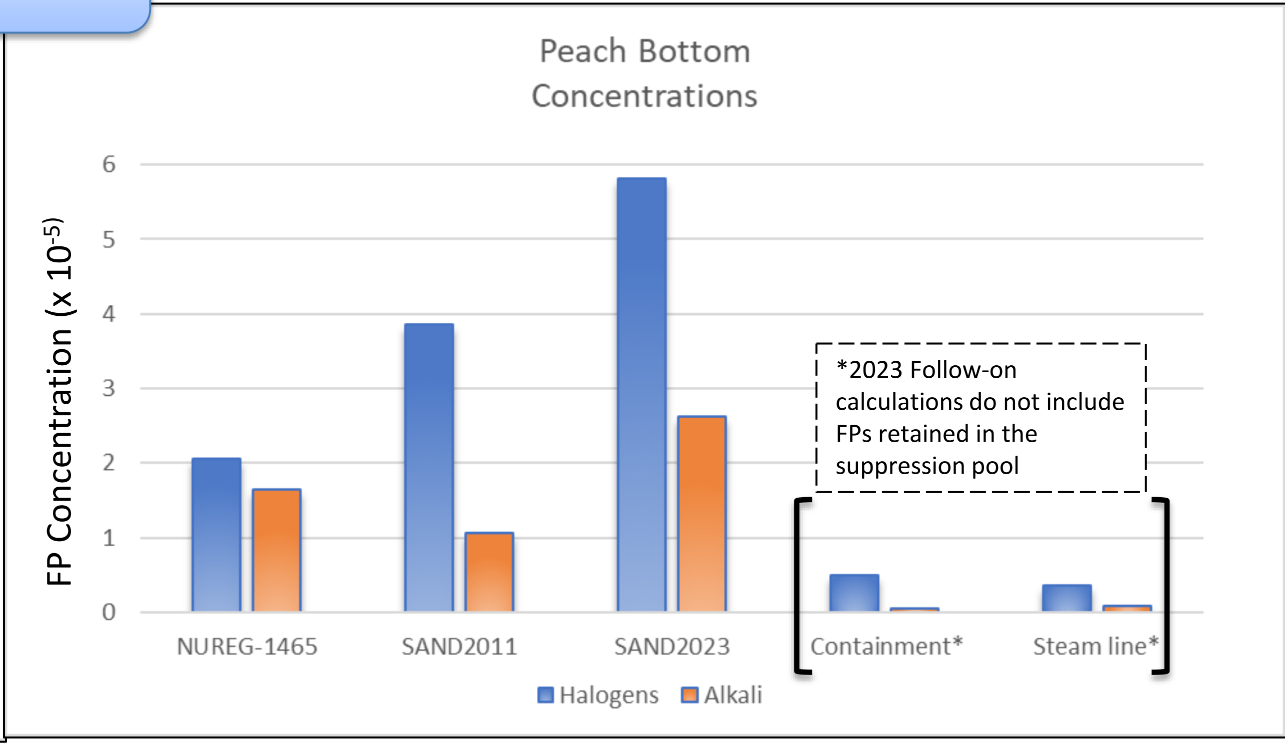
BWR Source Term (ST) Inventory Fractions – Early In-Vessel

Radionuclide Group	RG1.183 (rev0)	RG1.183 (rev1)	SAND2023	Pool (SAND2023 Table 5-16)	Containment (SAND2023 Table 5-16)	Steam Line (*Preliminary* Follow-on Calcs)
Noble Gases	9.50E-01	9.60E-01	9.50E-01	0.00E+00	9.50E-01	1.1E-03
Halogens	2.50E-01	5.40E-01	7.10E-01	6.50E-01	6.00E-02	5.1E-05
Alkali Metals	2.00E-01	1.40E-01	3.20E-01	3.10E-01	6.00E-03	1.3E-05
Te Group	5.00E-02	3.90E-01	5.60E-01	5.20E-01	3.80E-02	2.7E-05
Ba/Sr Group	2.00E-02	5.00E-03	5.00E-03	4.70E-03	3.00E-04	2.4E-07
Ru Group	3.00E-03	2.70E-03	6.00E-03	6.00E-03	7.40E-06	2.4E-07
Mo Group	3.00E-03	3.00E-02	1.20E-01	1.20E-01	1.00E-04	3.0E-06
Lanthanides	2.00E-04	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6	1.0E-11
Ce Group	5.00E-04	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6	8.4E-12

BWR Example Fission Product (FP) Concentrations (C_0)

$C_0 = ST / Vol$



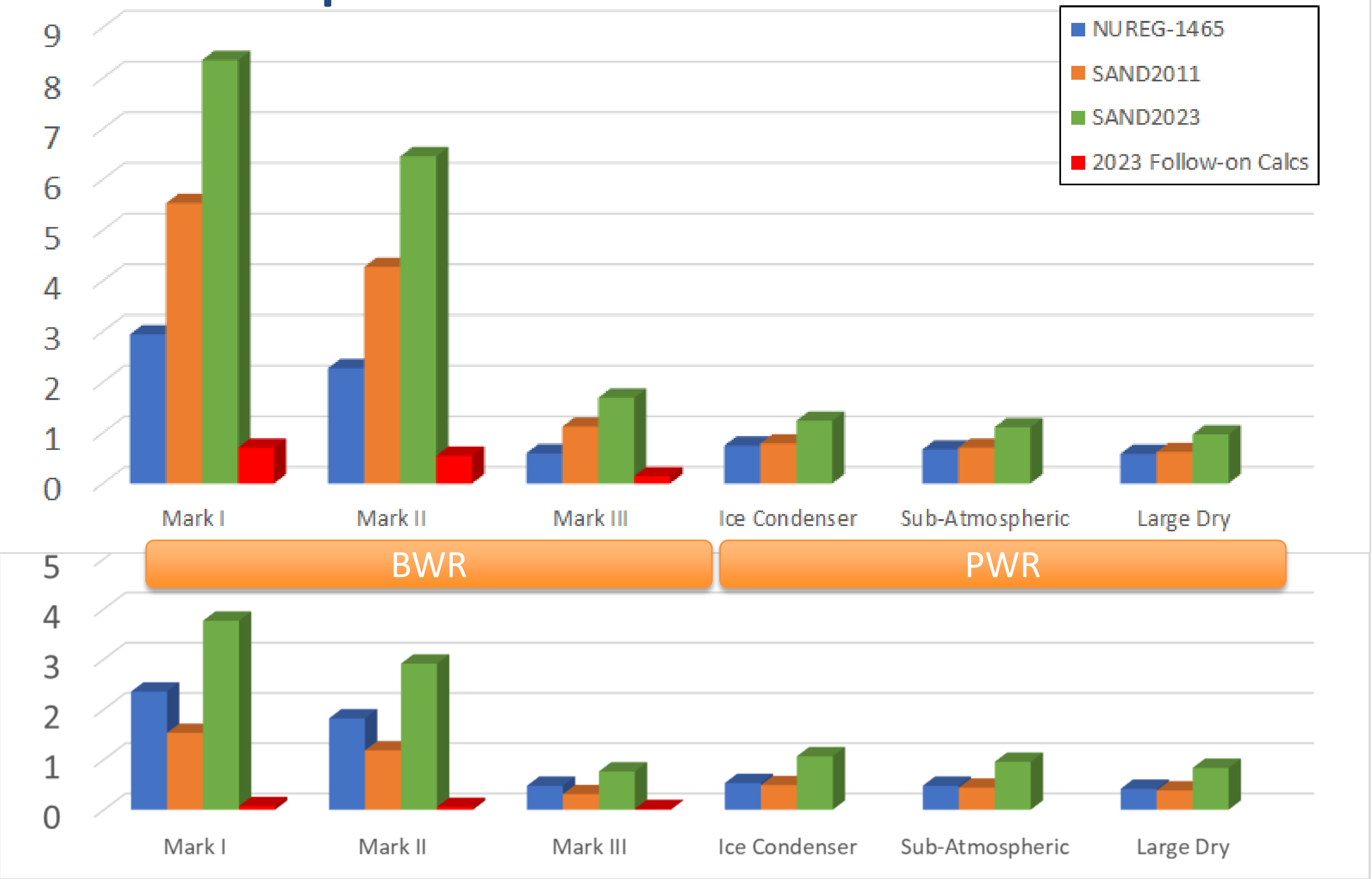


BWR/PWR Example Containment Concentrations

Halogen (Iodine) x 1E-5

C_0 =**ST/ Vol**

Alkali Metals (Cesium) x 1E-5



2023 Follow-on calculations do not include FPs retained in the BWR suppression pool

Typical containment volumes from *Figure 4.1-1 in NUREG/CR-6042, Rev. 2*

Example HBU Inventories

Radionuclide Group	BWR (Bq)	BWR (%) –> HBU	PWR (Bq)	PWR (%) –> HBU
Halogens (I)	3.54E19	<1%	2.53E19	<1%
Alkali Metals (Cs)	4.46E18	+7%	3.09E18	+5%
Chalcogen (Te)	1.16E19	<1%	8.35E18	<1%
	GE14 10x10	GE14 10x10	W 17x17	W 17x17
Core Avg. end of cycle BU (MWd/MTU)	36.2	41.4	43.5	48.3
Avg. Assembly discharge BU (MWd/MTU)	52.6	58.0	60.7	71.6
Initial Enrichment (%)	4.45	5.30	4.65	5.25
Power (MWt)	4016	4016	2893	2893
Cycle Length (months)	24	24	18	24

Conclusions and Next Steps

– Refined modeling provides better estimation of fission product distribution in the steamline.
  - Concentration in the steam line is distinct from that of containment.

– Significant retention of fission products were predicted in the suppression pool.

– Preliminary investigation of fission product inventories show limited effect for high burnup/high-assay low-enriched uranium (HBU/HALEU)fuels.

– Potential application of MELCOR to inform better estimates of fission product removal mechanisms in the simplified tools for regulatory applications and analysis where appropriate.

Backup Slides

Acronyms

Bq	Becquerel	MWt	Megawatt thermal
BWR	boiling-water reactor	PWR	pressurized water reactor
DBA	design-basis accident	RCIC	reactor core isolation cooling
FP	fission product	RG	(NRC) regulatory guide
GE	General Electric	RPV	reactor pressure vessel
HALEU	high-assay low-enriched uranium	SOARCA	State-of-the-Art Reactor Consequence Analyses
HBU	high burnup	SRV	safety relief valve
HPCI	high pressure coolant injection	ST	source term
MSIV	main steam line isolation valve	TCV	turbine control valve
MSL	main steam line	TSV	turbine stop valve
GWd/MTU	gigawatt-days per metric ton of uranium	W	Westinghouse

Table 5-16 Derived BWR release fractions including and excluding the suppression pool inventory for all core variations (60 GWd/MTU, 80 GWd/MTU, LEU and HALEU).

	Gap Release		Early In-vessel		Total (end of 72 hours)	
Release Category	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory	Including Suppression Pool Inventory	Excluding Suppression Pool Inventory
Noble Gases	0.016	0.016	0.95	0.95	1	1
Halogens	0.005	1.30E-06	0.71	0.06	0.87	0.2
Alkali Metals	0.005	1.20E-06	0.32	0.006	0.35	0.039
Te Group	0.003	<1.0e-6	0.56	0.038	0.78	0.26
Ba/Sr Group	0.0006	<1.0e-6	0.005	0.0003	0.048	0.042
Ru Group	<1.0e-6	<1.0e-6	0.006	7.40E-06	0.006	0.0001
Mo Group	1.90E-05	<1.0e-6	0.12	0.0001	0.13	0.002
Lanthanides	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6	3.70E-05	3.60E-05
Ce Group	<1.0e-6	<1.0e-6	<1.0e-6	<1.0e-6	0.003	0.003