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4 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

5 (ACRS)

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7 RADIATION PROTECTION AND NUCLEAR MATERIALS

8 SUBCOMMITTEE

9 + + + + +

10 THURSDAY, NOVEMBER 16, 2023

11 + + + + +

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

14 COMMITTEE MEMBERS:

15 DAVID A. PETTI, Chair

16 RONALD G. BALLINGER, Member

17 VICKI M. BIER, Member

18 CHARLES H. BROWN, JR., Member

19 GREGORY H. HALNON, Member

20 WALTER L. KIRCHNER, Member

21 JOSE A. MARCH-LEUBA, Member

22 ROBERT MARTIN, Member

23 JOY L. REMPE, Member

24 THOMAS ROBERTS, Member

25 MATTHEW W. SUNSERI, Member

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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.

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

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

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

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

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

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

23 When addressing the subcommittee,
24 participants should first identify themselves and
25 speak with sufficient clarity and volume so they may

1 be readily heard. When not speaking, we request that
2 participants mute their computers or microphone by
3 pressing star six.

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

10 And we thought it would be useful to have
11 a briefing on this so members get a more complete
12 picture of the depth of the technical basis upon which
13 1.183 relied.

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

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

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

24 There's a lot of interest as it relates to
25 Reg Guide 1.183 and, you know, a future update to that

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1 regulatory guide. So, I'm glad to see that there are
2 a lot of online participants as well, as those who are
3 in the room.

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

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

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

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

23 But before we get into that, you know, I
24 just wanted to say that, you know, this analysis that
25 the staff and contractors have been working on for

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1 quite some time is really important when it comes to
2 providing technical basis for the use of high burnup
3 fuel and also accident tolerant fuel in the future.

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

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

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

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

25 And so, because of the flexibilities of

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

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

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

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

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

24 So, in 2020, we convened a panel of
25 international experts to create the phenomena

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1 identification ranking table to address the
2 significant phenomenologic issues impacting core
3 degradation and radiological releases for various
4 accident tolerant and high burnup fuels.

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

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

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

16 And both of those are documented in
17 NUREGs.

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

24 And I think we mentioned that this will be
25 used as the, you know, probably part of the technical

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1 basis for the Reg Guide 1.183 Revision 2.

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

4 So, thank you for your attention.

5 MR. CAMPBELL: Yes, thank you very much.

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

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

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

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

17 So, Lucas?

18 MR. ALBRIGHT: Thanks, Shawn.

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

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

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

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

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

4 (Off-microphone comments.)

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

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

9 (Off-microphone comments.)

10 MR. ALBRIGHT: Yes, no problem. If
11 they're not readily available, we can pull them up and
12 --

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

15 MR. ALBRIGHT: All right, thank you all.

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

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

25 This was a multi-year effort to basically

1 extend the NUREG-1465 source terms to higher burnups.

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

7 Next slide, please? Thank you.

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

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

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

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

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

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

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

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

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

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

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

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

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

24 And this study actually looked at accident
25 source terms for, again, light water reactors, but

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1 using high burnup or MOX fuels.

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

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

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

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

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

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

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

25 We note first, Fukushima Daiichi happened

1 a couple months after SAND2011 was published.

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

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

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

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

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

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

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1 that we see.

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

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

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

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

15 What mean is that we have discretized the
16 core.

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

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

25 We have prolonged core damage progression

1 because we have more efficient transfer of heat out of
2 the core as its degrading.

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

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

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

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

18 MEMBER REMPE: I have a couple of
19 questions.

20 MR. ALBRIGHT: Yes?

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

25 For example, maybe we should learn a

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

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

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

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

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

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

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

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

25 We have two sort of concept nodalizations

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1 of a BWR Mark I containment structure on the left,
2 with the reactor vessel in the center there.

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

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

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

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

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

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

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

24 But the idea is that, as time has gone on,
25 particularly since NUREG-1465, these severe accident

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1 data sets from different experiments or different
2 events have given us a clearer idea and a clearer
3 picture of core degradation or core damage progression
4 and the radionuclide releases that occur as a result.

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

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

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

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

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

18 The next data set here is the Fukushima
19 Daiichi accident.

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

24 But one of the key findings that actually
25 came from one of our peer reviewers with their innate

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1 knowledge or intimate knowledge of what's going on
2 over there, was that they're actually seeing
3 confirmation that cesium molybdate is the dominant
4 chemical form of cesium.

5 So, next slide, please?

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

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

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

16 DR. SCHULTZ: Thank you.

17 MR. ALBRIGHT: Thank you.

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

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

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

25 The treatment of iodine in our severe

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

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

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

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

11 The next point is the chemical form of
12 cesium.

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

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

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

25 The next slide -- the next point here,

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1 excuse me, are the molybdenum releases.

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

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

10 Yes?

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

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

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

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

25 MR. ALBRIGHT: I think that's something

1 that Shawn can answer.

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

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

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

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

14 So, yes, for sure.

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

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

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

24 And, again, hopefully, we don't have
25 another accident and we're trying to figure out what's

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1 going on, but if it were, you would understand, well,
2 okay, MAAP hasn't incorporated this data or they did
3 incorporate something as we understand the
4 differences?

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

21 I can't speak to more recent updates
22 necessarily, but some of the indications from
23 Fukushima did tend to highlight the potential for a,
24 shall we call it, a more incoherent degradation that
25 is the downward relocation of debris that we typically

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1 see in MELCOR and a more segmented relocation of
2 debris in to the lower plenum as a result.

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

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

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

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

24 MEMBER REMPE: So, where I'm going with
25 what I'm asking is, because of, again, there's

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1 international comparisons which MELCOR is involved
2 with as well as other countries codes, is that, as you
3 improve the code which has been recommended, and
4 you're saying your going to keep doing it, keep in
5 mind the other codes in industry.

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

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

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

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

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

24 It's generally later in the in-vessel
25 accident progression where you start to either build

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1 up relative to realized debris or relatively, shall we
2 say, multi-cool like debris.

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

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

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

13 MR. LUXAT: Absolutely.

14 MEMBER REMPE: Okay.

15 MR. LUXAT: Absolutely.

16 MEMBER REMPE: Thank you.

17 MEMBER ROBERTS: Just a question of the
18 detail.

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

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

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

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

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

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

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

25 MEMBER ROBERTS: I'm wondering if you

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

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

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

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

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

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

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

23 And this was informed by experiments.

24 And if I remember correctly, the iodine --
25 the percent of the iodine mass that ended up in this

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1 elemental form was single digit percents, so, one,
2 two, three.

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

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

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

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

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

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

25 MEMBER ROBERTS: Okay, thank you.

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

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

7 MR. DICKSON: Yes, that's right.

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

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

14 MEMBER ROBERTS: All right.

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

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

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

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

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

8 Next slide, please?

9 CHAIR PETTI: Do they know why?

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

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

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

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

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

24 Yes, you relocated a lot of metal because
25 you've got a liquefaction which is, you know, taking

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1 the highly metallic.

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

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

9 And also, they stop before it melted.

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

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

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

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

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

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

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

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

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

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

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

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

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

24 Next slide, please? All right, key
25 findings. So, this is going to be a quick overview of

1 the main findings of our analysis. In our report, we
2 sort of have three key findings that we claim at the
3 beginning of this report.

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

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

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

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

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

24 And this is sort of related to some of
25 those findings from the SOARCA analysis, and we'll

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1 touch on that again in a couple of slides.

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

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

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

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

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

23 MEMBER REMPE: So, that's actually a
24 comment I was going to have. And I hate to nitpick on
25 words, but I think that you -- what you said is what

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1 you mean there is release fractions, not magnitude.

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

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

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

11 MR. ALBRIGHT: Thank you.

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

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

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

25 MEMBER ROBERTS: Okay, thank you.

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

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

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

13 MR. LUXAT: Yes, thanks, Shawn.

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

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

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

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

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1 period of interest for our analyses.

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

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

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

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

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

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

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

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

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

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

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

24 Next slide, please? All right, so now, we
25 get into sort of the high level tables or the main

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1 sort of cuts through our source term tables that we've
2 developed here.

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

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

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

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

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

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

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

25 MEMBER ROBERTS: Quick question.

1 Acknowledging what Kim said at the outset, I'm just
2 curious, maybe for logic, are you reconsidering that
3 1994 cutoff?

4 MR. DICKSON: That's a Commission policy.

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

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

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

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

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

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

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

1 MR. DICKSON: Yes, we can think about.

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

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

7 Thank you.

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

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

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

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

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

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

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

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

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

16 Next slide? Oh?

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

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

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

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

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

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

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

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

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

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

25 So, the release rates are fundamentally

1 different, right, in these calculations and in the
2 1465?

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

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

9 CHAIR PETTI: Yes, so, it's a combination
10 of better accident progression and release of what's
11 --

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

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

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

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

22 CHAIR PETTI: Right.

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

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

6 MR. ALBRIGHT: Sure.

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

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

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

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

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

25 When we look at numbers like this, I mean,

1 these are kind of mean. Whether they -- that is the
2 -- it's implied or stated, correct?

3 MR. ALBRIGHT: Yes.

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

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

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

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

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

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

1 sequence.

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

4 MR. LUXAT: Yes.

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

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

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

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

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

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

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

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

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

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

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

24 But within the scope of essentially
25 unmitigated events early in-vessel source terms, it's

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1 really those two characteristics that are principally
2 influencing the nature of the source term.

3 MEMBER MARTIN: Thanks.

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

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

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

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

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

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

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

25 So, this is that advancement in our

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

3 Next slide, please? Thank you.

4 Okay, so -- yes?

5 CHAIR PETTI: Just a question.

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

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

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

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

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

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

25 And that goes back to our frozen chemistry

1 assumption in MELCOR.

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

6 MR. SALAY: Mike Salay.

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

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

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

20 MR. LUXAT: We do.

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

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

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

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

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

5 MR. LUXAT: There is tellurium chem
6 absorption and the coefficient in the validation
7 data.

8 And we do actually have them as a --

9 CHAIR PETTI: Okay.

10 MR. LUXAT: -- show on surfaces.

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

16 But we do have chem absorption models and
17 --

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

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

24 CHAIR PETTI: Yes, cesium iodide, sure.

25 MR. ALBRIGHT: Okay, I think that covers

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1 this slide.

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

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

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

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

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

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

24 The reason we do this is because we have
25 downstream codes in the process that are meant to

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1 handle these mechanisms.

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

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

7 MR. ALBRIGHT: Yes, I apologize.

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

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

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

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

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

25 It needs to have a defensible technical

1 basis.

2 And it must be peer reviewed.

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

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

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

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

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

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

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

23 Next slide, please? So, as I mentioned,
24 we were in this process starting from SAND2011 inputs
25 and trying to maintain consistency so that the two

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1 reports could be sort of compared reasonably.

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

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

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

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

16 Next slide, please? Okay --

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

18 MR. ALBRIGHT: Yes.

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

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

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

25 Was there anything pulling on you to make

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1 changes to those?

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

4 MR. ALBRIGHT: Yes, yes.

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

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

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

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

18 I think -- oh, go ahead.

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

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

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

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

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

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

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

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

14 MR. ALBRIGHT: Thank you, Shawn.

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

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

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

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

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

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

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

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

25 And then, they have their own criteria

1 based on the cesium releases pertaining to those
2 different bases.

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

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

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

13 Oh, yes? Sorry, no, go ahead.

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

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

19 And you talk about not a single accident
20 scenario.

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

25 And do you think, I mean, just saying,

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

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

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

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

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

17 But --

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

23 And so, it literally excluded those.

24 MEMBER REMPE: Yes.

25 MR. CAMPBELL: And so, they tried to get

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

3 MEMBER REMPE: Good.

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

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

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

16 MR. ALBRIGHT: Sure.

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

21 It was an extension exercise at this
22 point. Right?

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

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1 MEMBER REMPE: Yes, maybe change the Reg
2 Guide.

3 MR. ALBRIGHT: Right, there you go.

4 MEMBER REMPE: It's just a comment.

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

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

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

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

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

24 So, starting from the core damage, what
25 happens during the core damage? And you know this

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1 even better than I do, is what's driving this source?

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

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

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

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

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

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

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

25 After all is said and done, so we have

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

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

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

8 Does that -- okay.

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

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

23 And Shawn is going to be talking a little
24 bit later about some of the follow on work that came
25 out of the peer review looking at the transport

1 pathways in more detail to better understand where
2 fission products have migrated into different regions
3 of the containment or the primary system.

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

8 And we'll be discussing that in more
9 detail.

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

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

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

19 MR. ALBRIGHT: Thank you.

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

25 And basically, in 2011, they took that IPE

1 to develop the accident sequences that they looked at
2 so that they were consistent.

3 And then, we have followed those through.

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

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

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

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

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

19 Next slide, please? So --

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

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

24 MR. ALBRIGHT: Say that one more time?

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

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

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

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

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

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

17 MR. ALBRIGHT: Say that one more time?

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

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

23 I don't think you said.

24 Another reason would be, they're not
25 bounding, but because of the way they're used, it

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1 doesn't matter that they're not bounding.

2 MR. ALBRIGHT: Okay.

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

6 Because there's a lot of intended uses.

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

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

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

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

19 MEMBER ROBERTS: Okay, I see.

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

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

25 I think the only statements that come to

1 mind are that the current spectrum of accident
2 sequences provided the coverage of the potential
3 accident sequences that were expected to be covered.

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

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

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

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

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

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

23 The accident signature, the core damage
24 signature, right, it does not change as long as they
25 have representative station blackouts or

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1 representative LOCAs, et cetera.

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

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

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

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

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

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

24 And you are absolutely right, you know, we
25 have done SOARCA uncertainty analysis. Even when you

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1 do SOARCA uncertainty for a single accident, you're
2 seeing a rate and, you know, what parametric do you
3 use.

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

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

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

14 Thank you.

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

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

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

7 MR. CAMPBELL: We're going to have a new
8 AST coming, I'll just say that. We're not redoing
9 2023 at this point, but we're doing really additional
10 follow on work to explore additional sensitivity to
11 that type of thing.

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

14 MR. ALBRIGHT: Thank you.

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

18 We break them out into sort of four
19 categories.

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

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

25 And then, we had two LOCA scenarios.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

25 We, again, have the possibility of high

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

5 So, all scenarios were low pressure
6 scenarios.

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

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

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

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

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

21 And then, we had the same containment
22 failures.

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

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1 wondering about that.

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

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

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

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

14 MEMBER ROBERTS: How about the NUREG-1465?

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

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

23 Because if it's based on that's the point
24 at which you could reverse the transient and make
25 things better and if you had containment failure

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1 before lower head failure, that doesn't seem to make
2 a whole lot of sense.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

15 Five percent of the iodine inventory, so
16 5 percent of cesium iodide is placed into the gap.
17 And then, the remaining 5 percent of cesium hydroxide
18 is made up of -- sorry, of cesium in the gap is made
19 up of cesium hydroxide.

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

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

25 So, big picture, uncertainty in iodine

1 speciation is persists to this day despite several
2 experimental studies.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

9 We have changed the liquefaction and
10 oxidized fuel failure temperatures for UO₂ and ZrO₂ to
11 2479 Kelvin. This is the mean value from the SOARCA
12 analyses -- the SOARCA uncertainty analyses.

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

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

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

24 But we're not really looking at any
25 parametric uncertainties except for the sensitivity

1 calculations where we investigate key phenomenological
2 uncertainties that were identified as part of the HBU,
3 HALEU, ATF, PIRT as being different or as being
4 distinct for high burnup or high burnup HALEU fuels
5 from conventional fuels.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

18 Yes?

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

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

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

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

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

9 My two cents.

10 MR. ALBRIGHT: Thank you.

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

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

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

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

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

25 Next slide, please?

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

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

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

13 MEMBER MARTIN: That's correct, yes.

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

17 MEMBER MARTIN: Maybe more so for BWRs?

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

23 MEMBER MARTIN: Okay.

24 MR. ALBRIGHT: Next slide, please? So,
25 this is the high level overview of this bootstrapping

1 procedure. Again, this was maintained from prior
2 studies and deployed in this study as well.

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

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

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

15 So, we calculate the 5th up to the 95th
16 percentile. And now, we have a distribution of 1,000
17 50th percentiles. The next step is to compute the
18 mean value of that distribution so that it's
19 representative and the standard deviation of each of
20 those percentiles.

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

25 So, I hope that was a clear example for

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

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

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

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

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

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

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

25 MEMBER REMPE: It's up to the subcommittee

1 chairman's discretion.

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

5 MEMBER REMPE: It's at your discretion.

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

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

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

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

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

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

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

25 And we considered accident scenarios,

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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1 of that primary pressure boundary.

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

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

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

14 MEMBER BIER: Quick question.

15 MR. ALBRIGHT: Yes?

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

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

22 So, the -- in the process of how these
23 numbers are used in regulatory practice, basically, we
24 calculate the total inventories and then, the
25 downstream codes will calculate the transport of those

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1 inventories to the environment.

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

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

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

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

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

24 And usually they have conservative models
25 that are chosen.

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

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

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

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

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

19 You're already ready, Shawn.

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

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

24 MEMBER MARTIN: Okay.

25 MR. CAMPBELL: I've got a whole

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

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

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

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

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

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

25 So, we try to explain that as well.

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

3 MR. ALBRIGHT: Thank you.

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

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

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

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

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

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

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

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

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

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

25 So, we should be seeing the same changes

1 across these two reactor types.

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

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

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

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

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

25 Next slide, please? So, this next group

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1 of slides will go over some of our sensitivity cases
2 or sensitivity calculations that were performed as
3 part of this report, very high level. For anyone
4 interested in the further details about some of the
5 different key radionuclide groups, we provide that in
6 the report.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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1 plenum to the lower head.

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

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

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

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

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

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

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

25 In this calculation, we basically looked

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1 at the SOARCA uncertainty analysis distribution of
2 fuel relocation temperatures. And we chose an
3 uncertainty range that spanned the SOARCA evaluation.

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

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

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

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

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

19 However, this uncertainty is also present
20 for conventional fuels.

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

24 And we're actually looking at the same
25 distribution of uncertainty here for these high burnup

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1 fuels in this case.

2 Next slide, please? Oh --

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

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

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

21 So through the temperature history, we
22 accumulate damage until the fuel fails. This is one
23 of the many fuel failure models that we have in
24 MELCOR. What we did in this analysis was we looked at
25 the reference which was the default time at

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1 temperature model, an increased lifetime, reduced
2 lifetime, and then the SOARCA lifetime.

3 And basically what this is, is it's timing
4 where -- a lifetime model where at lower temperatures
5 your fuel is going to stay in for a really long time,
6 22 hours in the case of the increased lifetime. And
7 at really high temperatures, around 2,600, we're going
8 to decrease the amount of time that it can stand. And
9 in our most -- in our smallest lifetime case, we only
10 allow the fuel to stand for three minutes if it
11 reached that high temperature.

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

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

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

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

18 So this is showing our source term
19 variability from an orange, the sequences, and in
20 purple -- sorry, in orange, the sensitivities and in
21 purple, the sequences. This really highlights sort of
22 the idea that we've been talking about all day today,
23 that the sequences are driving our source term
24 variability, not parametric uncertainties in our
25 models. So what you'll see and the easiest way to

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1 explain this is the uncertainty around the purple
2 lines is a larger span than then the uncertainty
3 around the orange lines.

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

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

25 DR. SCHULTZ: Did you happen to do that?

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

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

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

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

19 But generally what we do is that one way
20 to think about it is you're inducing variability
21 across a very complex network of calculations,
22 calculational steps in a code. So think of it, just
23 a myriad of floating point operations that you're
24 walking down this computational tree. And you put an
25 uncertainty up here or a variability up here and it

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1 kind of propagates through this not quite a butterfly
2 effect or a tree falling in Brazil or the Amazon, but
3 in some ways, very similar.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

16 (Simultaneous speaking.)

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

22 We used to worry about a penetration
23 versus global vessel failure and what does it mean.
24 And just again, now it's so important. At that time
25 is when we went from, like, one and half hours to

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1 eight hours and then six point whatever hours. It
2 seems like it's going all over the board. And maybe
3 it's not all at once.

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

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

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

23 Recommend improvements to the draft form
24 of the document prior to its publication. And then
25 any additional recommendations after its publication

1 as well as assess the suitability of the source terms
2 for the regulatory applications that were intended.
3 Next slide, please. The overview of the organization
4 of this panel, there were, let's see, six panel
5 members here from different organizations.

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

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

18 How appropriate were the sequences
19 selected? How the assumptions and applied models line
20 up with our current understanding of severe accidents
21 and source terms, and how adequate are these
22 approaches given current experimental and other
23 existing data sets. Finally, to assess whether our
24 source terms were representative rather than
25 conservative and bounding and then to basically make

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1 a conclusion on the overall technical basis of the
2 approach. So next slide, please.

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

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

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

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1 requirements. Next slide, please.

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

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

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

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1 gap and early in-vessel phases.

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

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

20 We talked about how the peer reviewers
21 acknowledged more recent PRAs that might give us a
22 different distribution of core damage contributors but
23 that the current analysis practice was suitable for
24 its intended purpose. There was the important note
25 that for most radionuclides there is not increase in

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1 activity with burn-up such that it would impact siting
2 calculations. Then we had comments on the uncertainty
3 in iodine speciation that we touched on earlier.

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

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

22 The issue that we will be talking about in
23 the next slide and the next presentation with Shawn is
24 the suppression pool scrubbing or the impact of
25 suppression pool and the radionuclide concentrations

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1 in the steam line. Next slide, please. So this
2 table, I don't want to spend too much time on the
3 details because Shawn is going to go into this in much
4 greater detail that I could in a single slide here.
5 But what we're looking at are for the gap release
6 phase and the early in-vessel phase, the total
7 inventory including the suppression pool, and the
8 total inventory excluding the suppression pool.

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

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

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1 directly into the suppression pool.

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

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

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

24 CHAIR PETTI: What's your timeline for
25 those?

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

7 CHAIR PETTI: We might be interested.

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

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

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

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

25 So overall background here, like I said,

1 the peer review panel has commented on potential
2 impact that a suppression pool could have on what's
3 getting into containment obviously. And that table,
4 it's Table 516 if anyone is interested. It's in the
5 SAND2023 report.

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

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

22 So before I go forward, there's been a lot
23 of confusion about what a source term versus
24 inventories versus all that. So I thought that this
25 could be kind of useful to make sure that we're all on

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1 the same page. We all understand what is a
2 containment source term and it's being used.

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

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

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

25 It's the cumulative amount of fission

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

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

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

20 So that's where you would divide by the
21 volume of your containment, your individual
22 containment, to get your concentration of the source
23 term into that containment structure. So it's a
24 simple ST divided by volume here. That concentration
25 is then used in this simplified approach using removal

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1 mechanisms, whether that may be sprays or a natural
2 deposition or whatever that might be.

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

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

13 (Laughter.)

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

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

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

24 MEMBER REMPE: It is --

25 (Simultaneous speaking.)

1 MEMBER REMPE: But somebody else's would
2 be.

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

9 MEMBER REMPE: It's just a reason to think
10 about.

11 MR. CAMPBELL: Right. Okay. Any other
12 questions?

13 MEMBER ROBERTS: Yeah, just for
14 clarification. When you talked about suppression pool
15 scrubbing, I don't see that on this page. It's not on
16 the removal mechanisms, or --

17 MR. CAMPBELL: It is not part of the
18 removal mechanisms.

19 MEMBER ROBERTS: Okay. So if you don't
20 account for it, then --

21 MR. CAMPBELL: It is not accounted for in
22 this bottom part. And then up on the top, obviously
23 MELCOR is calculating it. But it's part of that whole
24 bookkeeping. It still gets lumped into that green
25 space and brought down.

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1 MEMBER ROBERTS: Right. So it could be
2 part of that mechanism modeling. Mechanism modeling,
3 if you chose to do it that --

4 (Simultaneous speaking.)

5 MR. CAMPBELL: And that's -- yes and no.
6 And that's one of the things I want to kind of tease
7 out here because one of the concerns is, is there
8 anything that's bypassing. And that's one of the
9 aspects that you have to kind of tease out in this
10 whole aspect.

11 So you don't want to just simply say, oh,
12 let's just scrub and take 80 percent, 95 percent of it
13 is just gone. Are there pathways -- release pathways
14 to the people space. Let me bypass this whole thing.
15 And so that's what we're trying to investigate here.

16 MEMBER ROBERTS: Okay, thanks.

17 MR. CAMPBELL: Yeah. All right. Next
18 slide, please. So this is just trying to show where
19 we've tried to refine our models. So over here on the
20 left is a representative -- this is Peach Bottom. So
21 this is our original from 2023.

22 This is the nodalization that we have
23 within MELCOR. So we wanted more refined nodalization
24 of the steam line in particular because this is what
25 we're talking about when I'm saying particularly

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1 through the MSIVs we're trying to understand what's
2 going on up there, what's happening within the steam
3 line, what's the concentration within the steam line.
4 And so we refined our modeling.

5 And I'm afraid I can't zoom in here. If
6 I had my mouse, I would. But in this purple space
7 here, you can see that top bar. It's two volumes.
8 You can see two boxes.

9 That was our steam line A. Thank you very
10 much. That's our steam line A. And so that's the
11 steam line that has the lowest pressure SRV. And so
12 that had a slightly more -- thank you so much.

13 That has a slightly more refined modeling,
14 I guess, with two CVs. But then everything else was
15 lumped together into a single CV, all three of the
16 other lines. So we wanted to break this out a little
17 bit so we could better capture the physics of what's
18 going on in the steam line.

19 So this is where we have our more refined
20 modeling. And the reason we need more refined
21 modeling if you look on the right-hand side over here,
22 you can see it's a rather long -- the steam line is
23 rather long. It's got bends and turns and everything,
24 not to mention the SRV cycling on it, HPCI, RCIC
25 pulling off and so on.

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1 So we wanted to be able to better capture
2 what's going on if we're going to ask questions about
3 what is the fraction of inventory that's making its
4 way into the steam line. So next slide, please.
5 Thank you. So here's our more refined steam line
6 modeling. So for each of the BWRs, each steam line
7 has now been broken up into this nodalization.

8 So the first volume that I have here in
9 blue, this is going from the steam dome down to and
10 including the first SRV. And in our model, that first
11 SRV is the lowest pressure SRV. So that's the one
12 that's going to be cycling.

13 And then the next volume is this green
14 volume. This is everything downstream of that first
15 SRV up to the first MSIV. RCIC and HPCI pull off of
16 this line depending on the steam line obviously.

17 And so we have a much finer nodalization.
18 We also have as a separate volume then in between the
19 MSIVs downstream to the MSIV down to the stop valve
20 and condenser. So all of that has much more refined
21 modeling. And then so now going forward --

22 MEMBER HALNON: Is the only vent to the
23 environment the condenser because you have so many
24 others?

25 MR. CAMPBELL: We do. And we're not going

1 out to that level of detail to try to model everything
2 that can get to the environment. That's really don't
3 by those downstream codes, right? We're not trying to
4 capture -- you might have leakage through the valve
5 stems and so on. We're not trying to capture overall
6 --

7 (Simultaneous speaking.)

8 MEMBER HALNON: That's usually a
9 significant loss, megawatts.

10 MR. CAMPBELL: Sure.

11 MEMBER HALNON: So that we -- we're
12 always chasing. It could be one -- three percent.
13 Getting close though.

14 MR. CAMPBELL: Sure.

15 MEMBER HALNON: Quite a bit.

16 MR. CAMPBELL: Yeah, and as I'm about to
17 say, we kind of stopped our investigation at the first
18 MSIV. And so I'll explain why here in a moment. But
19 most of our investigation is here in this green
20 portion.

21 So in the rest of my presentation, I'm
22 going to be talking about source term fraction kind of
23 in the same way that we talk about our containment
24 source term. I'm going to be talking about source
25 term fraction in the steam line because one release

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1 pathway for BWRs obviously is the MSIVs, right? And
2 this is a release path that has the potential for
3 bypassing the suppression pool.

4 And that's why we thought that it was
5 important to better characterize and understand what's
6 going on in the steam line. So as we go forward, a
7 few things that I need to talk about here about this
8 green portion. First of all, when I report source
9 term in the future, I'm reporting what's in the green
10 portion.

11 We thought this was representative kind of
12 in the same way that the containment source term is
13 what's available for release through a leakage pathway
14 in containment. This is kind of what's available for
15 release through an MSIV. Also distinct from how the
16 source term is reported in containment as a cumulative
17 release into containment, that's harder to do here in
18 this green portion because you don't have fission
19 products entering into this volume and then staying
20 there indefinitely due to cycling of the SRV, through
21 RCIC and HPCI operation, through leakage through the
22 MSIV.

23 This is a dynamic volume. It's not quite
24 as dynamic obviously as that blue portion. But this
25 is a dynamic volume. And so you're going to have a

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1 lot of in and out.

2 So we needed a better way to come up with
3 a value for source term. And what we came up with is
4 a time averaged source term. And I'll try to flush
5 that out a little bit more as we go.

6 But we talked about what we're going to
7 capture here is a time averaged airborne fission
8 product source term. And so emphasis on airborne
9 because we're already taking into account all of those
10 deposition mechanisms. This is another thing that's
11 distinct from how we report in containment where those
12 downstream codes then look at whatever might deposit
13 through sprays or through whatever.

14 We've already taken that into account here
15 in this green portion in order to try to capture all
16 of that physics, the ins and the outs. We're
17 reporting it all as an overall fraction of airborne
18 time averaged within that phase. So I'll pause there.
19 That was a lot.

20 Okay. Well, then I'll move on from there.
21 And if we need to cycle back to this, I can. So this
22 is what we're reporting here. We've seen most of
23 these values already many times.

24 You've got your Reg Guide 1.183, Rev. 0,
25 Rev. 1, and then 2023 values that my colleagues have

1 been talking a lot about. And then these next two, I
2 need to point out. These are just the Table 5-16
3 values.

4 These are not doing a bunch of additional
5 calculations. This is a bookkeeping exercise here of
6 what is or is not in the suppression pool. The only
7 thing that's truly unique on this slide versus what we
8 did in the 2023 report is in the last column, and this
9 is this fraction that is sitting in that green portion
10 of the steam line.

11 Now I want to point out here this is
12 fraction. So you may look at that and say it's
13 incredibly small, e to the -5 . But you have to
14 remember this is a different volume.

15 Concentration is what's really important
16 here, not fraction. And so what we need to really do
17 is look at what's the overall concentration comparison
18 between what's in containment versus what may be
19 sitting in the steam line. And that's why the next
20 slide is going to be really important where now we
21 have to look at an example and divide by the volume,
22 then do a concentration comparison between what's in
23 containment versus what's in the steam line.

24 And that's why the next slide is going to
25 be really important where now we have to look at an

1 example and divide by the volume, then do a
2 concentration comparison between what's in containment
3 versus what's in the steam line. So that's what I
4 want to do in the next slide if you don't mind going
5 to the next slide. So here we are.

6 We got Grand Gulf on the left, Peach
7 Bottom on the right. All I've done here is taken
8 those values from the previous slide and divided by
9 the containment volume. So this is fraction of core
10 inventory per meter cubed. That's the units here.

11 And so you can see 1465 and 2011 and 2023
12 all listed here. And then off here in my brackets,
13 this is what's made it into containment minus what's
14 in the suppression pool. That's what's in that first
15 column of the brackets and then what's in the steam
16 line.

17 So all this is, is the SAND2023 values,
18 the first column. All it is, is the SAND2023 taking
19 away the suppression. That's all I've done here. But
20 then on the steam line side, I just need to point out
21 a few things.

22 Once again, you may look at this and think
23 that steam line containment are comparable. But be
24 careful. Remember these aren't apples and apples
25 source terms, right? The one on the right has already

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1 taken into account any deposition mechanisms within
2 that phase. So this is the concentration throughout
3 that phase, if that makes sense.

4 MEMBER BIER: So I'm trying to understand
5 the whole picture. So the third set of bars is the
6 2023 results --

7 MR. CAMPBELL: Yes.

8 MEMBER BIER: -- that these guys just
9 talked about.

10 MR. CAMPBELL: Correct.

11 MEMBER BIER: And you parcel it out
12 further and get those bars to the right?

13 MR. CAMPBELL: That's right.

14 MEMBER BIER: Do you have a figure that
15 does that to the 2011 numbers, or --

16 MR. CAMPBELL: This has 2011 numbers on
17 there as well.

18 MEMBER BIER: No, but that parcel out
19 process on the right.

20 MR. CAMPBELL: Oh, no, no. I'm actually
21 separating what's in the suppression pool versus --
22 no, we don't have those values. I would assume it's
23 going to be kind of analogous here.

24 MEMBER BIER: Got it. That's what I
25 wanted.

1 MR. CAMPBELL: It's a fair assumption. I
2 don't have that in front of me.

3 MEMBER BIER: Okay.

4 MR. CAMPBELL: A couple things I'll point
5 out. Size of containment matters obviously. Grand
6 Gulf is a much larger containment. So your
7 concentration is diluted here.

8 Also, I'll point out that in the steam
9 line, you notice it really doesn't change too much
10 between the two plants. And that's really a
11 consequence of your steam lines aren't all that
12 different between the plants. It's a relative -- the
13 steam line is relatively similar between these two
14 plants. All right. Yes.

15 So then here the only purpose of this
16 slide was trying to do a comparison of BWR to PWR
17 then. So I'm not talking about the steam line in this
18 slide. So the purpose here is just to say how does
19 this compare to what's going on with Ps.

20 And so this is representative volumes for
21 a couple Mark I, Mark II, Mark III and then for PWRs
22 and ice condensers, sub-atmospheric and a large dry.
23 So I've just divided by some volumes here. And you
24 can see that time progression essentially from 1465 to
25 2011 to 2023, and then if you don't account for what's

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1 in the suppression pool.

2 And you'll see that at least for the
3 halogens, then it's more similar to what you're seeing
4 in a PWR. All right. So the purpose of this one also
5 was -- and this was a recent addition. The purpose of
6 this one really was to find some kind of
7 representative example fission product activities for
8 a few case studies that we've done for high burnup and
9 high enrichment PWRs and BWRs.

10 So we're trying to calculate end of cycle
11 activities for that inventory, a piece of this, right?
12 Remember we're going back to that first slide.
13 Inventory matters in activities.

14 So that's what we're trying to get at here
15 versus the kilograms that we've been talking about in
16 the 2023 report. We're trying to better understand
17 what would it be in terms of activity for some of
18 these reactors going up to higher burn-ups. So here
19 we have for BWRs and PWRs, we have a reference case on
20 the first column.

21 So we've got a core average end of cycle
22 burnup for the reference case for a B of 36.2. We
23 have an average assembly discharge burnup of 52.6 with
24 an enrichment of 4.45. And then we're going to a more
25 higher burnup here.

1 And we're estimating about 58 average
2 assembly discharge burnup. And this is going from an
3 enrichment of 4.45 to 5.3. So we're trying to come to
4 something that's, like, a representative loading
5 pattern for a BWR that might want to go to a higher
6 burnup and tease out what kind of values they could
7 get.

8 What you see here, I find it kind of
9 interesting is that the iodine concentration trends
10 with the power. And since the power hasn't increased
11 here, your iodine concentration hasn't increased. Now
12 the alkalis, however, that increases with your burnup.

13 So you do see a proportional increase of
14 your cesium going to these higher burnups. And this
15 is the same for Bs and for Ps. And then same thing
16 for the tellurium.

17 There's really a weak dependency on burnup
18 enrichment for tellurium as well. Same thing with the
19 Ps. We went to a higher cycling for the Ps and had a
20 higher enrichment and higher burnup there as well.

21 So you're seeing that we're not getting --
22 maybe it's a little deceptive when you're looking at
23 some of the 2023 values and you're seeing a 40 percent
24 increase in cesium and iodine mass. That's not what
25 we're talking about as far as activity. And we're not

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1 trying to say that this is the activity that you're
2 going to see from a licensee.

3 We're just saying this is possibly a
4 little bit more representative of something going to
5 a higher burnup. Okay. Next slide, please. So
6 conclusions, we did some refined modeling, and it
7 provided some better estimates of fission product
8 distribution in the steam line and then in
9 containment. And we found that concentration in the
10 steam line is distinct from that of containment when
11 you're not looking at the suppression pool.

12 MEMBER HALNON: On the steam line, I guess
13 I'm still befuddled a little bit.

14 MR. CAMPBELL: Please.

15 MEMBER HALNON: And you're probably just
16 going to say that's somebody else's job. But we've
17 been looking at license amendments that are trying to
18 take credit for the downstream sections of piping or
19 scrubbing --

20 MR. CAMPBELL: Yeah.

21 MEMBER HALNON: -- and other items. How
22 is that factoring in? Or is that somebody else's
23 issue?

24 MR. CAMPBELL: The answer is it is a
25 regulatory issue of how it's going to be applied.

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1 That being said, one of the things we were trying
2 tease out here is can you apply -- could you apply the
3 same source term that's in containment sans
4 suppression pool? Would you apply that same source
5 term then to MSIV leakage? And I think that's what
6 I'm trying to tease out here is that wouldn't be quite
7 right. You really need to have a -- there is a
8 distinct and distinctly higher concentration in the
9 steam line than there is in containment minus
10 suppression.

11 MEMBER HALNON: Even if you add in all
12 that extra piping just because --

13 MR. CAMPBELL: And that's the thing, if
14 you look at all the downstream effects. But that's
15 typically teased out by the individual applicant or
16 licensees, right, because they're using their
17 downstream codes to look at all of that.

18 MEMBER HALNON: This is the verifying,
19 validating, whatever you want to call it, confirmed
20 starting point. If they can justify some other type
21 of scrubbing suppression pool or downstream, then they
22 might be able to bring it back to a point where their
23 design works.

24 MR. CAMPBELL: Right. And that's what
25 we're trying to tease out here.

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1 MEMBER HALNON: Okay.

2 CHAIR PETTI: So longer term, bigger
3 picture is this stuff will inform and update to 1.183
4 on MSIV?

5 MR. CAMPBELL: I would have to defer to my
6 NRR colleagues on that. Our purpose here first of all
7 was just to be responsive to the peer review, right?
8 That was the purpose here was to say the peer review
9 said to go look at this. Let's be responsive to that
10 because they're saying that this suppression pool is
11 going to have an impact.

12 But we wanted to make sure that there has
13 been talk of suppression pool, suppression pool. We
14 wanted to make sure that we teased out can you just
15 apply the suppression pool or just do a DF completely
16 off of that same containment source term without
17 looking up the steam line separately. And hopefully
18 we've teased out that, no, you do need to look at that
19 steam line as a separate entity.

20 MR. DICKSON: Hey Shawn?

21 MR. CAMPBELL: Yeah.

22 MR. DICKSON: Elijah Dickson with the
23 staff. I'd like to connect some of the other, like,
24 regulatory initiatives that are going on right now.
25 So just a few weeks ago, I spoke in regards to the

1 increased enrichment. And we're looking at the
2 control and design criteria in that case.

3 And part of those alternatives is looking
4 at these mechanistic transport models in support of
5 development of Regulatory Guidance 1.183, Rev. 2. So
6 we are considering all of this based off of the
7 current work, based off comments received from ACRS,
8 individuals in industry. So it's all part of that
9 work right now. And we're still in the process of
10 collecting public comments. Again, I just wanted to
11 kind of connect the dots in regards to other
12 regulatory initiatives that are being done right now.

13 MR. CAMPBELL: Great.

14 MR. DICKSON: Yeah, no problem.

15 MR. KORTGE: Can I ask a question?

16 MR. CAMPBELL: Yeah, please.

17 CHAIR PETTI: Who is --

18 MR. KORTGE: What is the mode of force for
19 the suppression pool?

20 CHAIR PETTI: Sorry, who's speaking?

21 MR. KORTGE: This is David Kortge.

22 CHAIR PETTI: From? Organization?

23 MR. KORTGE: Constellation, Safety
24 Analysis.

25 CHAIR PETTI: Yeah, no, you can't ask.

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1 Only during public comments.

2 MR. KORTGE: Apologies.

3 MEMBER BROWN: As an uninitiated, may I
4 ask a question?

5 CHAIR PETTI: Maybe.

6 MEMBER BROWN: In excruciating detail.
7 I'm trying to figure out what the bottom line is.
8 It's wonderful analysis. It's not answering my
9 question.

10 MR. CAMPBELL: Please.

11 MEMBER BROWN: If I read this and did you
12 analysis, all the green pipes are longer than the
13 other pipes which means to me you've got higher source
14 terms you have to deal with under a severe accident.
15 Is that going to affect now the EPZs or the zones that
16 we have to deal with? Is that -- I mean, they're
17 bigger, a lot bigger.

18 MR. CAMPBELL: Which one is? Can we go up
19 one slide, please?

20 MEMBER BROWN: I'm looking at the one with
21 the bars, the little green, red, and gold, previous
22 slide.

23 MR. CAMPBELL: Previous, let's go up one.
24 Up another one. This one right here?

25 MEMBER BROWN: Yeah, that's the first

1 question. Does that affect --

2 MR. CAMPBELL: So I'll say that the orange
3 right there is Reg Guide 1.183, Rev 1 basis. Is that
4 correct? So the 2011 values there --

5 MEMBER BROWN: Yeah, with the little --

6 MR. CAMPBELL: -- in the orange?

7 MEMBER BROWN: Yeah.

8 MR. CAMPBELL: So then that is used by the
9 downstream codes as the regulatory basis.

10 MEMBER BROWN: So now it's going to get
11 bigger.

12 MR. CAMPBELL: If 2023 was the basis for
13 a reg guide, then --

14 MEMBER BROWN: But you would argue -- I'm
15 trying to be contrary a little bit here. But you
16 would argue that you've proved that the basis --
17 you've had peer review of the basis, an impressive
18 list of people with qualifications to do that review
19 which would seem to indicate that there's a
20 significant increase in what the source term you have
21 to deal with for computing EPZs which would apply to
22 present day plants. Now that's the takeaway that
23 somebody that's not -- I mean, don't ask me to do the
24 calculations. Okay? My mind was exploding while you
25 were doing that. The second question I would have --

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1 the answer is yes, if you adopt it.

2 MR. CAMPBELL: That's where I would have
3 to defer to my friend at NRR.

4 MEMBER BROWN: I understand you're
5 deferring. That's fine. The second is this is high
6 burnup. People want to go to high burnup. Changes
7 your refueling, all that good stuff.

8 But then once you've taken the fuel out,
9 you now have a spent fuel. This would imply that you
10 have a bigger -- something you have to deal with in
11 the spent fuel pools as well, as well as in storage
12 casks once they've cooled down and you've got them in
13 storage casks. Does that compute?

14 You've got more source term you start
15 with. You've got more residual. It's not the
16 instantaneous stuff. But I'm trying to come to a
17 conclusion. How does this affect other things? Like,
18 now do I have to have different casks? I haven't
19 heard any of that in any of the previous high burnup
20 fuel discussions.

21 MR. CAMPBELL: Go back to the point of
22 Lucas's source term. But it's not the high burnup but
23 actually the change in how you're doing sequences and
24 going from high pressure to low pressure that causes
25 that increase primarily, not the increase burnup.

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1 MR. CAMPBELL: In other words, we don't
2 anticipate there being a significant change going to
3 high burnup in the overall activity.

4 (Simultaneous speaking.)

5 MEMBER BIER: But there is a significant
6 change due to the analysis.

7 MR. CAMPBELL: In our source term. In our
8 source term.

9 (Simultaneous speaking.)

10 MEMBER BROWN: Existing low burnup then
11 would have the same results. And so that you're still
12 impacted in terms of the EPZs and/or (audio
13 interference). So if it's not a result of the burnup,
14 you just figured out that you didn't have the right
15 answers before for EPZ and for --

16 MR. CAMPBELL: Elijah is jumping at the
17 bit.

18 CHAIR PETTI: This is a research
19 presentation, then there is a licensing part.

20 MEMBER BROWN: I understand, okay.

21 CHAIR PETTI: This doesn't affect any of
22 the existing plans.

23 MR. CAMPBELL: That's right.

24 MEMBER BROWN: Well, you've got results
25 that has to be evaluated at some point that says, the

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1 analysis that was used is more comprehensive, has a
2 higher result, and it's independent. That's what he
3 just said, whether you use high burnup fuel or the
4 regular fuel we've got now which implies to me that
5 somebody is going to have to sit down with you all,
6 thrash out another -- you have to go address this in
7 the existing plans in terms of an analysis of their
8 occupations at speakeasies and our spent fuel pools
9 and/or past storage and/or transportation of those
10 casks.

11 More shielding is -- whatever. I'm just
12 a poor electrical guy. I do understand having numbers
13 come out significantly different regardless of how you
14 got there.

15 MR. LUXAT: So let me just quickly make a
16 comment that from the peer review, again, the method,
17 they appreciated the advance in the methods. But one
18 of the important comments from the peer review panel
19 was an RD contamination measures related to the
20 suppression pool that were not being --

21 (Simultaneous speaking.)

22 MR. LUXAT: -- for BWRs that were not
23 directly being addressed. And what Shawn has been
24 talking about is if we were to look at the effect of
25 those decontamination or those removable mechanisms in

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1 the suppression pool, A, it would have an impact on
2 what the containment source term is. But it would
3 also importantly have an effect on what the
4 concentration of fission products are nearby other
5 release pathways like the MSIV.

6 And that's what this is teasing out is
7 that the key comment from the peer review was there's
8 an important removal mechanism, a passive removal or
9 a inherent removal mechanism for BWRs. And what we've
10 been doing is we've been looking to expand the
11 technical basis and understand better the transport of
12 fission products out of the reactor system and how it
13 potentially -- how they could potentially be removed
14 by the suppression pool, what that effect would be on
15 the containment source term. But also importantly,
16 what the effect of that removal could be on other
17 release pathways that are not, if you will,
18 interfacing directly with the suppression pool.

19 MEMBER BROWN: There's other mechanisms
20 that remove some of this. That's fine under --

21 (Simultaneous speaking.)

22 MR. LUXAT: That was -- yeah, but that was

23 --

24 MEMBER BROWN: -- standpoint. But what
25 about the non-severe accident standpoint? I have to

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1 deal with removal. And you're still telling me that
2 I've got more stuff that I have to deal with that's
3 got source term in it when you pull it out regardless
4 if you have a severe accident or not.

5 That's the other takeaway I -- that's the
6 uninitiated, bottom line, takeaway for somebody like
7 me. And yet somewhere -- forget the severe accident.
8 I understand a PWR is a PWR.

9 They have other ways of reducing the
10 activity so it doesn't get spread all over the place.
11 But I still have to deal with regular fuel, regular
12 high burn-up, put it into the spent fuel, then put it
13 in a cask when it gets heated. And now I've got
14 larger source terms that I have to deal with in those
15 circumstances.

16 MR. DICKSON: This is Elijah Dickson.
17 With an increased enrichment, the rule making efforts,
18 they are looking at transportation and several other
19 rules as well. In regards to EPZ sizing, there are
20 several regulatory source terms that we use for
21 different purposes.

22 So this is an in containment source term
23 used to size mitigative systems to reduce radioactive
24 release to the environment and inside a facility,
25 certain distance away from a population center. For

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1 EPZ sizing, they do use this in containment source
2 term as one of the analysis that went into the Ten
3 Mile EPZ rule, right? They also used source terms
4 that were derived from the PRAs, WASH-1400 as well.

5 And then also looked at different figures
6 of merit at that time to justify that Ten Mile EPZ.
7 So between the work that's been done in SOARCA with
8 the severe accident work and the more realistic
9 consequence analyses and this in containment source
10 term used as a design tool, right, to size safety-
11 related mitigative equipment. There's no a whole lot
12 of difference that we're seeing between now and what
13 was done back in the late '60s when we did the EPZ
14 sizing.

15 MEMBER BROWN: If you take into these
16 other considerations and other things. I'm trying to
17 transition away from the severe accident. We haven't
18 had, quote, severe accidents.

19 MR. DICKSON: Three Mile Island was.

20 MEMBER BROWN: That was not as severe as
21 a real severe accident.

22 MR. DICKSON: Right.

23 MEMBER BROWN: That was like spitting in
24 the ocean.

25 MR. DICKSON: Right.

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1 MEMBER BROWN: And a lot of dumb stuff.
2 I mean, my point being is what this tells me in your
3 results regardless you've got analyses, methodologies
4 in terms of routine operations. You still built up a
5 different spectrum of stuff that you had --

6 MR. DICKSON: Understood.

7 MEMBER BROWN: -- to deal with on a
8 routine refueling storage, then cask storage, and then
9 transportation issues aside from this severe accident.
10 That's all. I'm just trying to make sure that thought
11 process is in place as well or should be in place.

12 MR. DICKSON: It is. It is. We're
13 thinking a lot about --

14 MEMBER BROWN: There's --

15 MR. DICKSON: -- all of this.

16 MEMBER BROWN: -- routine operations, no
17 accidents. We haven't been getting the right answers.
18 If everybody accepts this as proper analyses --

19 MR. DICKSON: Right.

20 MEMBER BROWN: -- to get the right
21 results. Thank you. I think I've exhausted my brain
22 power.

23 MEMBER BIER: I have one more question
24 that's kind of a follow-up. Charlie, you were saying
25 that, okay, there are other mitigations that are not

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1 reflected in the results here. Would those be
2 reflected in the results of a Level 3 PRA? Or are
3 Level 3s overestimated because they're not modeling
4 all of the (audio interference)?

5 MR. DICKSON: So I'm a practitioner of
6 this source term. I utilize this source term. These
7 are the experts that develop the source term, right?

8 Again, this is Elijah Dickson with the
9 staff. The way Reg Guide 1.183 is set up, Rev. 1 has
10 these tables of fractions of the reactor core source
11 term. In the appendices, Appendix A specifically is
12 the MHA LOCA appendix that tells you how to transport
13 this source term through all the different systems,
14 how to credit different type of removal mechanisms.
15 And that's where Shawn has been discussing how we can
16 make improvements in these particular areas, these
17 mechanistic transport models, in the appendices
18 specifically understood.

19 MR. CAMPBELL: Can we INSPECTOR BOTH: back
20 to that final slide? No, that was it. That was it.

21 MEMBER BROWN: Sorry to disrupt.

22 MR. CAMPBELL: No, thank you, no. So I
23 think the second bullet here is pretty obvious.
24 There's a lot of retention in the suppression pool.

25 Third bullet, preliminary investigation of

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1 the fission product inventory showed limited effect
2 for high burnup and high enriched uranium. So this
3 was that whole inventory aspect that you have marginal
4 increase in your actual activity. So inventory
5 matters.

6 And then finally, this is something that
7 we're investigating right now, we're looking into.
8 There's potential to apply MELCOR to better inform
9 some of those removal mechanisms that I was talking
10 about in that first slide, those lambdas and to inform
11 those for the simplified tools. And I just wanted to
12 point out here we're in the process of drafting a
13 document that summarizes this work that we've been
14 doing on investigating concentrations within the steam
15 line. So this is preliminary in that we're looking
16 for early feedback from ACRS on our process and so on
17 as we're drafting this report.

18 MEMBER BROWN: Well, don't take my
19 questions as being negative or critical. It's nice to
20 see we're not sitting on our past accomplishments, but
21 yet we're trying to make sure we're doing it right for
22 both today and the future. So don't take my comments
23 as being majority that are critical because it's good
24 to see somebody doing good research that comes up with
25 some results we can deal with, at least reduce it to

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1 the understanding of the common man.

2 MR. CAMPBELL: We take all feedback as
3 constructive.

4 MEMBER REMPE: Okay. What are you going
5 to do about the existing source term and the
6 differences between? I mean, you've got this new
7 model of higher release fractions because of the way
8 that the sequences modeled. And can't do back fits
9 probably. And so what will you do with this --

10 MR. DICKSON: We're still in the review
11 process. We're just now kicking off increased
12 enrichment. And with that is the work in developing
13 Reg Guide, Rev. 2.

14 MEMBER REMPE: It's a different problem
15 because it's --

16 MR. DICKSON: It is.

17 MEMBER REMPE: -- a penalty for the high
18 burnup, high enriched fuels because of our increase in
19 knowledge. And how does one deal with that?

20 MR. DICKSON: This is the beginning of
21 that process looking at this report and doing these
22 additional analyses. At this point, I can only share
23 so much and how we want to look at this and what we
24 want to be doing in Rev. 2, Reg Guide 1.183.

25 MEMBER REMPE: This is not going to come

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1 up in the next month or two.

2 MR. DICKSON: Timeline, it's matching up
3 with increased enrichment. So this is under the
4 umbrella of increased enrichment. So I think we said
5 beginning or end of the school year, calendar year
6 2024, beginning of 2025 we'll be seeing something.

7 MEMBER ROBERTS: I'm not sure you answered
8 Vicki's questions. If you can go back to Slide 8.
9 No, the one after that, the one with the colors. That
10 one. No. Thank you.

11 Yeah, I think what I've got out of these
12 last couple hours is there's really no difference in
13 ultimate consequence between the blue, the orange, and
14 the green because they're driven by changes in
15 modeling. And the fission products are going to come
16 out from the core. At some point, the progression,
17 just those somewhat arbitrary division of lower vessel
18 head rupture.

19 That defines when you stop making that
20 plot. But what Vicki asked I think is interesting.
21 If you have a Level 3 PRA with these different models,
22 would the end result be significantly different?
23 Because the timing, right, is not all that different.

24 It's the timing of the events that are
25 happening, not so much the timing of the degradation

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1 of the fission product release. So if you did a Level
2 3 PRA with these different models, would you have a
3 different answer? Or would it just all come out in
4 the wash? I mean, there's nobody here who can answer
5 that question.

6 DR. ESMAILI: And this is Hossein Esmaili.
7 Can I just say so we're not going to be using this or
8 any Level 3 PRA. Remember what Shawn showed in the
9 first or second slide that a Level 3 PRA, that's the
10 thing that we are actually doing right now is that we
11 are going to be doing a very mechanistic accident
12 progression source term calculation, going all the way
13 to lower head failure, MCCI, et cetera.

14 So whatever is going to come out of that,
15 whether it's a containment failure, what is the
16 release to the containment. So we are not looking at
17 the release to the containment. And don't forget,
18 this release to the containment, these bars that you
19 see, it's everything.

20 It's airborne, deposit that in the
21 suppression pool, et cetera. In a consequence
22 analysis, we are not going to be looking at what's in
23 the deposit, et cetera. So for a Level 3 PRA type of
24 analysis, we are just going to go that path, the one
25 that Shawn showed at the beginning.

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1 We're going to go all the way, do a very
2 integrated analysis and look at what's going to come
3 out of the containment, whenever it's leaking or
4 containment failure, et cetera. This one is just for
5 the purpose of doing that bottom one which means that
6 we are doing a simplified approach right? I mean,
7 that simplified approach, somebody is going to take
8 care of the position, et cetera, those lambdas in the
9 real -- up one in the integrated analysis, that's all
10 part of the calculation.

11 MELCOR will calculate the position and the
12 structure, the position on the pool, in the pool
13 surface itself. So in that respect, you know, we are
14 just being very mechanistic. What we come down to is
15 that we specify what that lambda is from this
16 calculation, right?

17 MEMBER ROBERTS: Okay. Thanks, Hossein.
18 I think what I got out of that is the blue that's
19 there would be NUREG-1150. So whatever models produce
20 the blue also produce results at NUREG-1150. The
21 green is whatever is coming out of your current Level
22 3 PRA project.

23 And so you would see if there's an effect
24 between the blue and the green if you can find it with
25 all the other different things that have changed over

1 the last 30 years. But yeah, thanks for the answer.
2 I think that answered my question. So there's no --
3 in the severe accident world or Level 3 PRA world,
4 there's really no distinction to be drawn by this
5 blue-yellow or orange-green.

6 (Simultaneous speaking.)

7 DR. ESMAILI: Yes, in Level 3 PRA, we
8 would be looking at the accident sequences and just
9 combining them, whatever the plant damage say, what is
10 the release category, et cetera. This thing is only
11 when we come up with that green box that Shawn showed.
12 And then we do that path of a simplified model.

13 MEMBER ROBERTS: I didn't get an answer to
14 Charlie's question then. Is the blue versus orange
15 versus green have a real meaning in the reactor space?

16 MR. CAMPBELL: Yeah, that's sort of what
17 I want to say.

18 MEMBER BROWN: I walked away with that.
19 I transitioned back to the regular stuff. There's
20 still the differences you have to deal with.

21 MR. CAMPBELL: Yeah, so some of these are
22 artifacts because it's these regulatory applications.
23 Like, Level 3 PRA, you'd actually -- you want to
24 consider taking out the effects of the suppression
25 pool. You have to do that for the downstream

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1 licensing calcs only. So that's --

2 CHAIR PETTI: I just think it shows how
3 difficult it is to take what's done and let's call it
4 the state of the art. And to try to boil something
5 down that you can put into a licensing approach that's
6 simple because you don't want something complex
7 because there's so much artificiality. And you can
8 get misled as much as you can get an answer that's
9 good. So good luck, Elijah.

10 MEMBER REMPE: Okay. So back in the
11 NUREG-1465 days, they thought an hour and a half was
12 enough for a source term, right? If you did an hour
13 and a half with your 2023 source term, you'd probably
14 get a lower amount, not a higher amount, right, or
15 something comparable? If you want to be consistent,
16 why not just say an hour and a half and cut it off
17 because vessel failure is irrelevant.

18 DR. ESMAILI: So this is Hossein Esmaili.
19 So Joy, I'm just going to defer to NRR, right, because
20 we are -- no, I'm just trying to be very careful
21 because what we are doing is that we are showing
22 everything. We are showing you everything.

23 This is what happens in the containment.
24 This is what happens if you just take out the
25 suppression pool. You can see in terms of the

1 concentration. And concentration is what's really
2 driving this. You go from that green, that big green.
3 You go to that red which is --

4 (Simultaneous speaking.)

5 DR. ESMAILI: And we are going to have
6 meetings. We're going to have public workshops in the
7 next few months starting in January going to April.
8 And whatever regulatory decision they're going to
9 make, it's going to come at the end of that.

10 MEMBER REMPE: So the question really
11 wasn't for you or for --

12 (Simultaneous speaking.)

13 MEMBER HALNON: Okay.

14 MEMBER REMPE: It's more for Elijah.
15 Elijah, that's what was done back in the days of 1465.
16 And that was enough in order to be a consistent
17 predictable regulator. It seems to me that that was
18 enough back then to say, okay.

19 DR. ESMAILI: But talking about the vessel
20 and lower head failure, we have learned over the years
21 that there was the calculations we used to do, like,
22 20, 30 years ago, as soon as the debris would melt and
23 come down. It would reach with the lower head,
24 whether it's a drain plug, whether it's an instrument
25 tube, and fail it and go out. So that was that.

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1 Nowadays --

2 MEMBER REMPE: We used to worry about high
3 pressure injection too and a lot of things we don't
4 worry about anymore. But anyway, it's just a
5 suggestion that might be --

6 DR. ESMAILI: Yeah, yeah. No, that's a
7 very good suggestion. All I'm suggesting is that some
8 of the -- this durations, and this came out of SOARCA,
9 is because we're doing a better modeling of heat
10 transfer to the water, you know, that it has to boil.
11 There's a massive amount of structures in the lower
12 plenum of a BWR against the duration of the (audio
13 interference). So all of those things factor into the
14 fact that now it's in vessel phase. Actually, this
15 was a SOARCA insight.

16 It's going to take a long time, right?
17 And I hear you. I'm not going to make any judgments.
18 Leave it up to NRR, what they want to do. But we have
19 the data. We have the analysis. We can go data
20 mining. We have other choices. It's just that
21 decision has not been --

22 MEMBER REMPE: I get it. I just was
23 trying to figure out how one would get out of this
24 mess. And so it's not a question.

25 DR. ESMAILI: Yeah, the other thing is

1 that you can see that we talk about the suppression
2 pool. It's really not -- we should not be comparing
3 the source term. We should really comparing the
4 concentrations because when you compare, like, in a
5 BWR, we have the suppression pool.

6 So you automatically see that the
7 concentration goes down with the different Mark I or
8 Mark III design. But in the PWR case, you don't have
9 a suppression pool. But you have a huge containment
10 volume, right?

11 So the concentration you can see, it's
12 just like comparable even to a -- the Mark III. So
13 that was the purpose of showing this. Sorry, thank
14 you.

15 CHAIR PETTI: I guess we should probably
16 go out for public comment. Any members of the public
17 that would wish to make a comment, please unmute
18 yourself, name, organization, and comment.

19 Hearing nobody online, we do have someone
20 in the room. Please go ahead.

21 MR. CSONTOS: So I just wanted to say
22 thank you very much for the presentations. The first
23 time -- oh, Al Csontos, NEI, Director of Fuels. Thank
24 you very much for the presentation and discussion. I
25 think it's very helpful. We had a lot of comments and

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1 questions and looking forward to having those
2 workshops that you talked about.

3 I think that will be really important to
4 count as dialogue on how the impact of the new
5 modeling that's here goes out to the actual
6 implementation steps. Charlie, to your question, a
7 comment you made. EPRI did do a scoping study looking
8 at the impact of high enrichment, higher burnup on the
9 back end. Okay. And --

10 MEMBER BROWN: Without severe accident.

11 MR. CSONTOS: Yes, this is more -- you had
12 mentioned dry cask spent fuel, things like that. That
13 report number is 3002027535. I'll pass it on to Larry
14 and folks to provide you.

15 But the bottom line there was that there's
16 very little impact to the back end. There's a small
17 -- longer time that you might have to put it in the
18 pool, but it's manageable. And actually dose rates
19 for the workers actually goes down because you're
20 loading less canisters. So that's the -- but you can
21 read the study.

22 MEMBER BROWN: The good news is you've
23 looked at it.

24 MR. CSONTOS: Yes, we've looked at it.
25 But thank you very much and more on this later. And

1 also during the rulemaking reg basis comments that the
2 industries would like.

3 CHAIR PETTI: Thank you. With that, I
4 don't see any more comments. So I think we -- oh, I
5 want to -- yes.

6 MEMBER HALNON: Yeah, you guys did a great
7 job. I mean, you kept energy through this whole
8 thing, and that kept us going. For those of us like
9 my young friend here who are not modelers, it was
10 understandable. Appreciate it.

11 MEMBER BROWN: I couldn't have asked the
12 question without absorbing a little bit of what you
13 said for the last two and a half hours, three hours,
14 whatever it is now.

15 MEMBER REMPE: It's good to hear things.
16 I mean, one, we were curious about this because of the
17 Reg Guide 1.183 but also the new calculations and
18 understanding this will help others when they're
19 trying to deal with Reg Guide 1.183, Rev. 2.

20 CHAIR PETTI: Very important when we hear
21 the next go around of (audio interference). Technical
22 basis here was just very informative. Okay. Thank
23 you everyone, and we are adjourned.

24 (Whereupon, the above-entitled matter went
25 off the record at 4:44 p.m.)

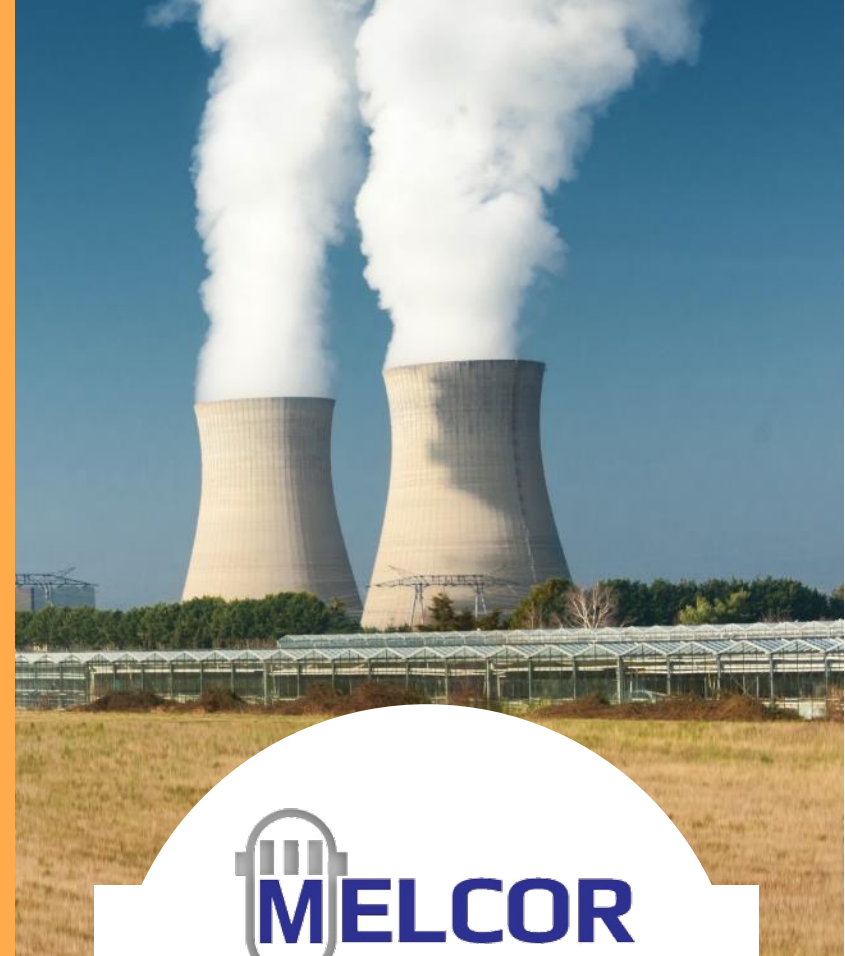
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High Burnup Fuel Accident Source Terms

ACRS Briefing Nov 16, 2023

Presented by Lucas I. Albright and David L. Luxat

Contents



- Motivation and Background
- Key Messages
- Deep Dive
- Summary
- Independent Peer Review
- Upcoming Work



Motivation and Background

SANDIA REPORT

SAND2023-01313
Printed April 2023

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
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California 94550

High Burnup Fuel Source Term Analysis Motivation

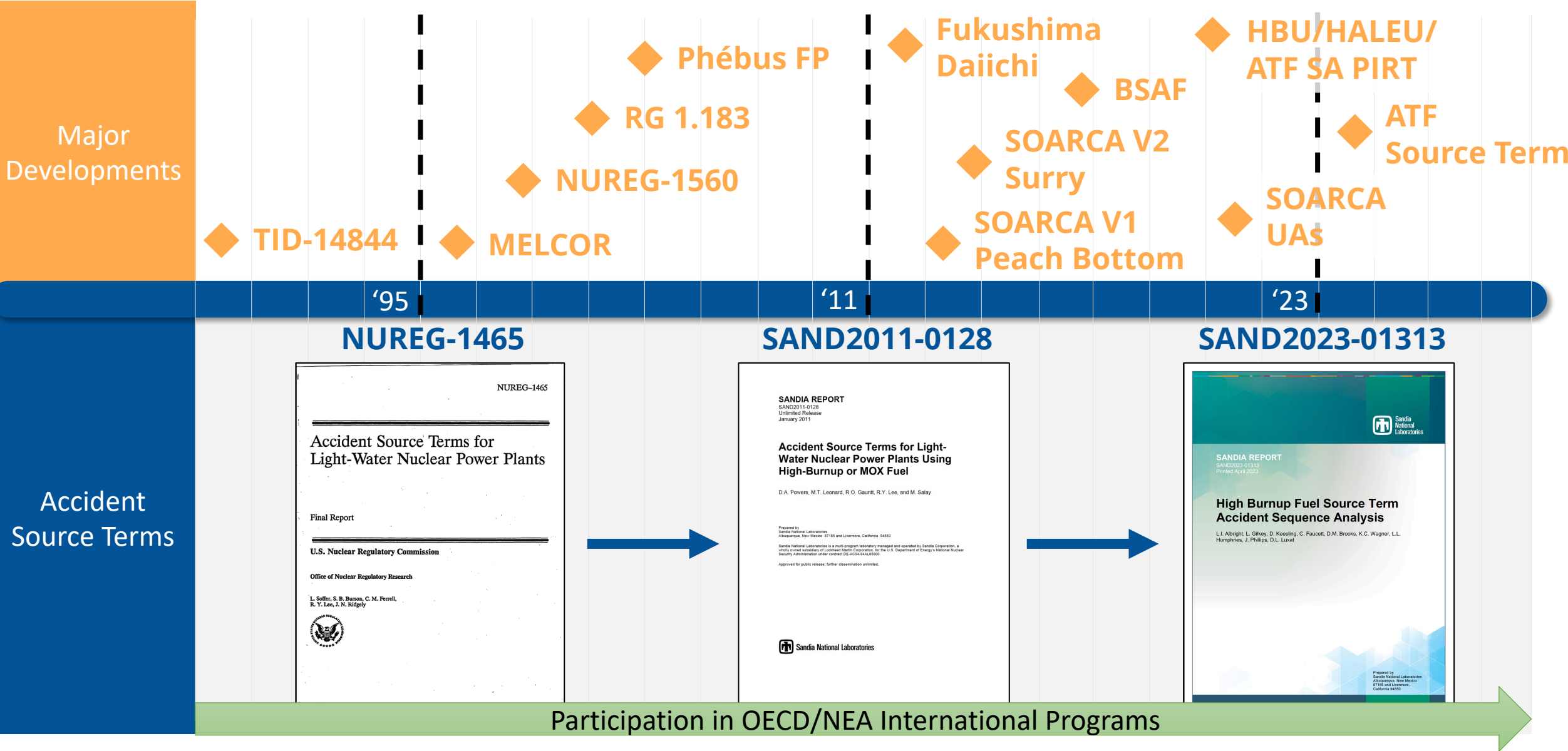
- Develop alternative source term applicable to LWR cores with HBU/HALEU fuel
 - Different burnup levels and enrichments considered
- Extends NUREG-1465 and SAND2011-0128 alternative source terms

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



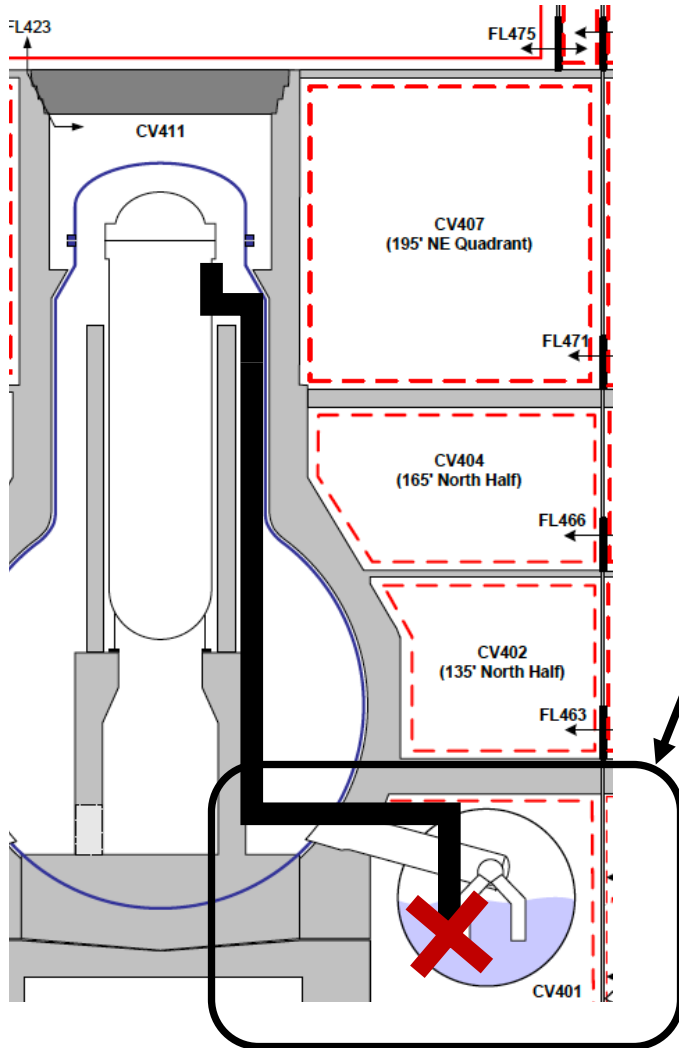
Severe Accident Modeling Advancements



- **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

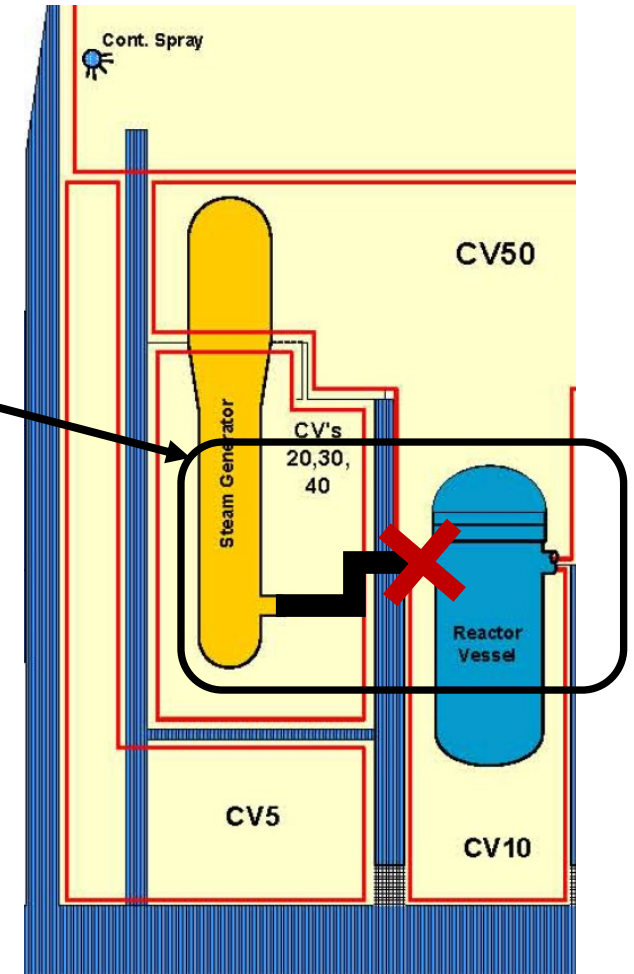
BWR: Thermally induced SRV seizure



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

*Diagrams are for illustration purposes only

PWR: Hot Leg Creep Rupture



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

Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases

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

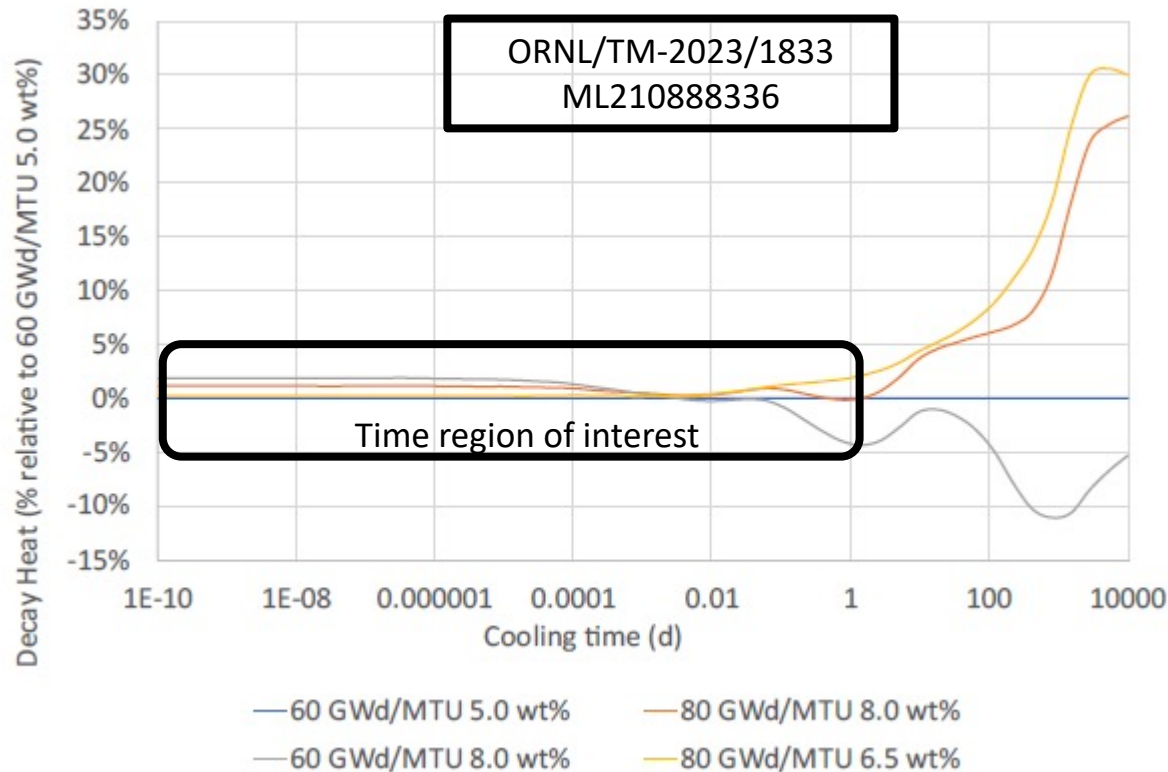
Key Finding 2: Larger early releases to containment result from early primary pressure boundary failure

- 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)

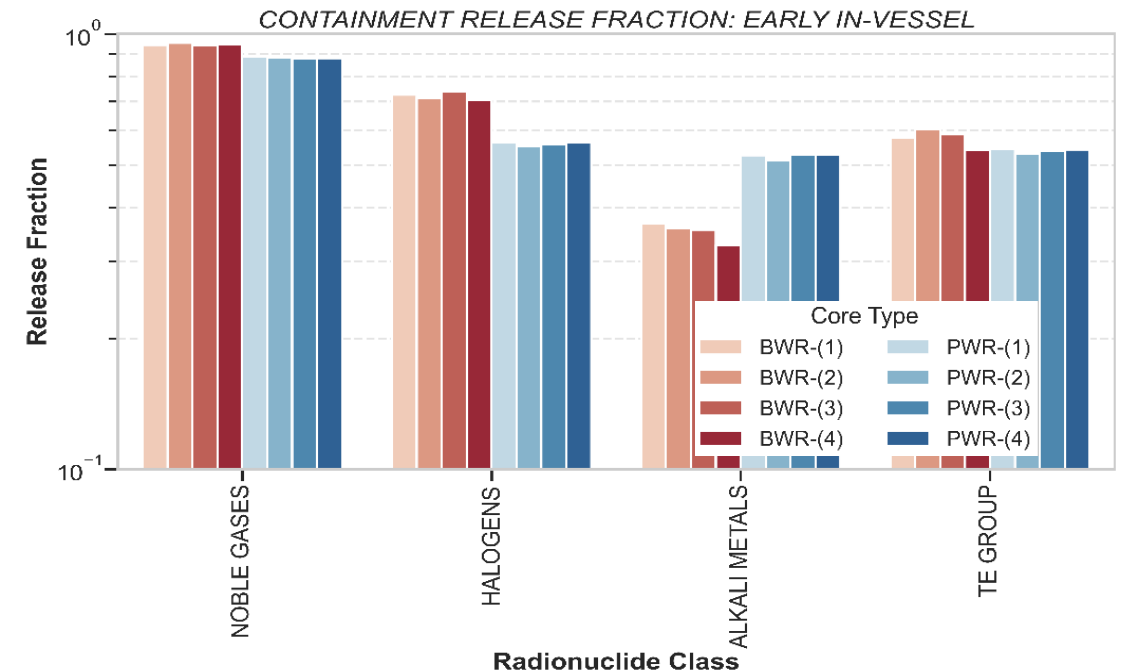
High Burnup and Extended Enrichment Impact on Source Term



Burnup and enrichment do not significantly change decay heat after reactor shutdown

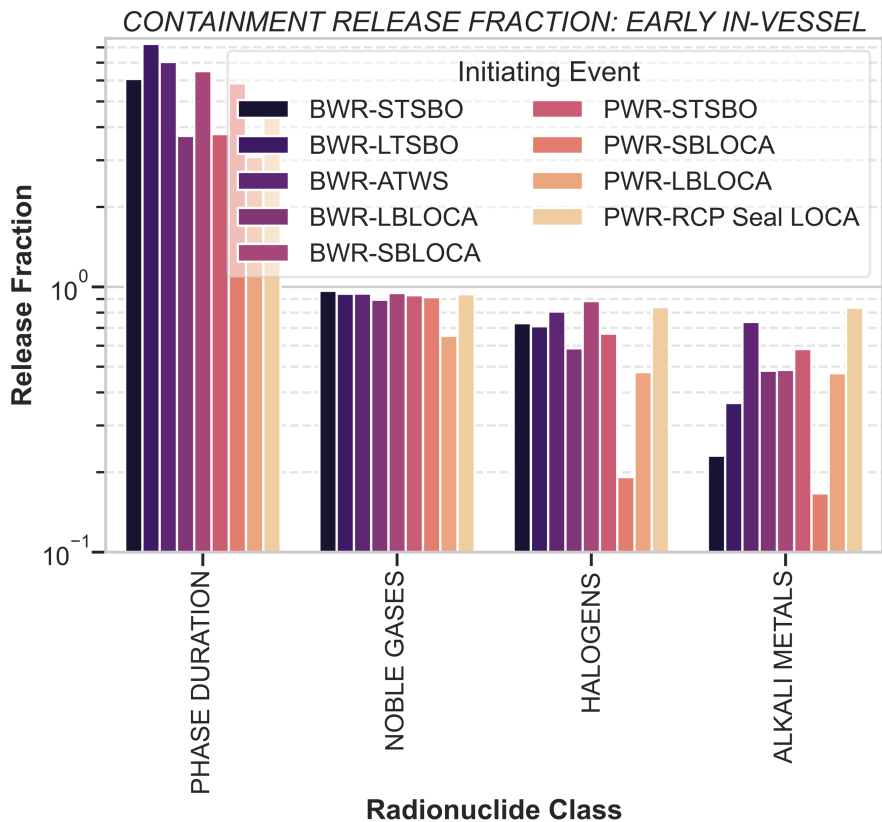
Core Types:

- (1) 60 GWd/MTU LEU (2) 80 GWd/MTU LEU
(3) 60 GWd/MTU HALEU (4) 80 GWd/MTU HALEU



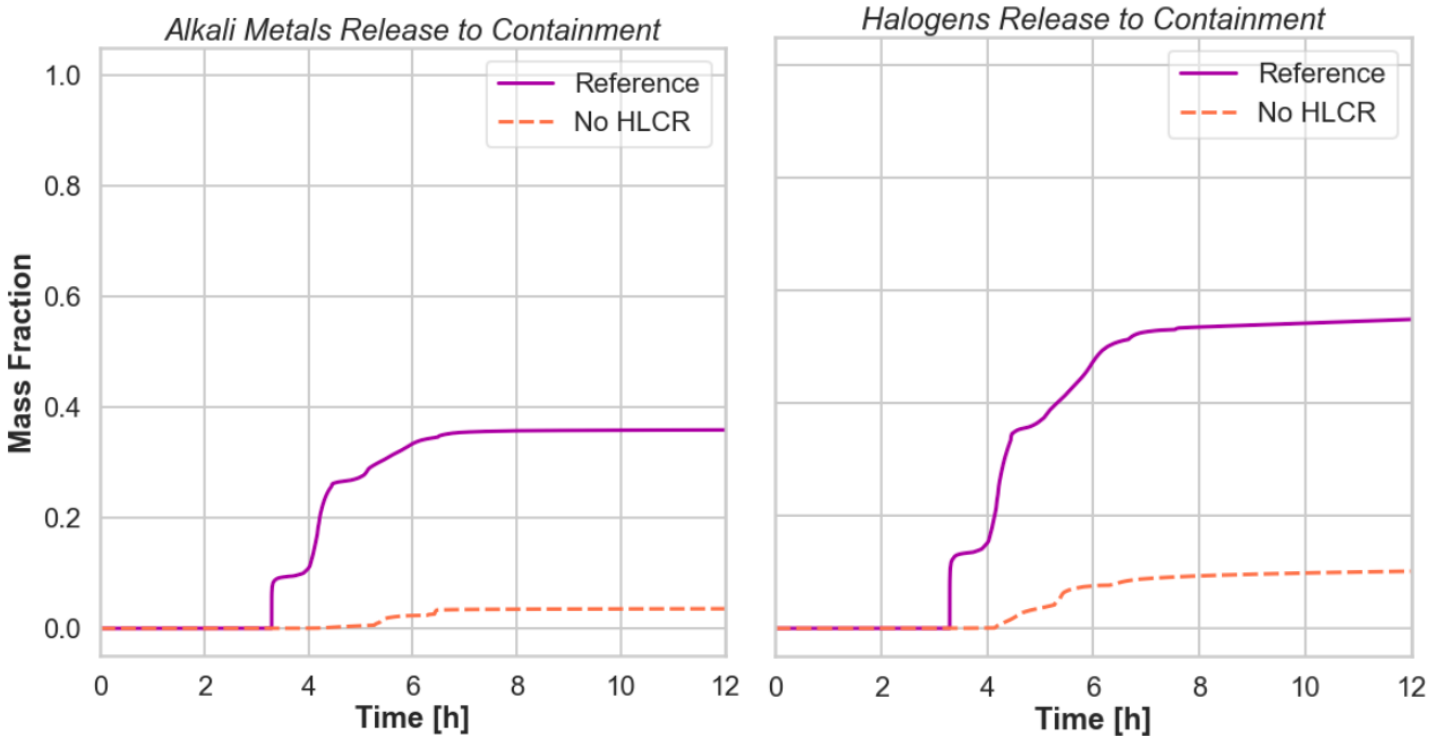
Increased burnup and enrichment does not strongly impact in-containment source term

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



	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
		2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
BWR	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			0.30
	Alkali Metals	0.005	0.050	0.32	0.20	0.021			0.35
	Te Group	0.003	0.0	0.56	0.050	0.19			0.25
PWR	Phase Duration	1.3	0.50	4.0	1.3	24.0			2.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010			0.0
	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



		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
		2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
BWR	Report								
	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			0.30
	Alkali Metals	0.005	0.050	0.32	0.20	0.021			0.35
	Te Group	0.003	0.0	0.56	0.050	0.19			0.25
PWR		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Report								
	Phase Duration	1.3	0.50	4.0	1.3	24.0			2.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010			0.0
	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
- Progressive releases to containment due to enhanced reactor coolant system modeling

In-Containment Source Term Differences



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	0.016	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			0.30
	Alkali Metals	0.005	0.050	0.32	0.20	0.021			0.35
	Te Group	0.003	0.0	0.56	0.050	0.19			0.25
PWR		Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	1.3	0.50	4.0	1.3	24.0			2.0
	Noble Gases	0.026	0.050	0.93	0.95	0.010			0.0
	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
- 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	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
		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	0.001	0.003	0.100
	Alkali Metals	0.007	0.10	0.047	0.13	0.0006	0.001	0.003	0.12
	Te Group	0.005	0.0	0.091	0.033	0.005	0.001	0.003	0.083
PWR	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
		1.3	0.50	4.0	1.3	24.0	10.0	3.1	3.0
	Noble Gases	0.019	0.10	0.21	0.73	0.0008	0.0	0.003	0.0
	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



BWR	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	0.70	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	0.001	0.006	0.100
	Alkali Metals	0.007	0.10	0.047	0.13	0.0006	0.001	0.001	0.12
	Te Group	0.005	0.0	0.091	0.033	0.005	0.001	0.001	0.083
PWR	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	1.3	1.3	0.50	4.0	1.3	24.0	10.0	3.1	3.0
	Noble Gases	0.019	0.10	0.21	0.73	0.0008	0.0	0.003	0.0
	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
- Release rates (release fraction/hour) are generally smaller

In-Containment Source Term Release Rates



BWR	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	0.70	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	0.001	0.003	0.100
	Alkali Metals	0.007	0.10	0.047	0.13	0.0006	0.001	0.003	0.12
	Te Group	0.005	0.0	0.091	0.033	0.005	0.001	0.003	0.083
PWR	Report	Gap Release		Early In-vessel		Late In-vessel		Ex-vessel	
	Phase Duration	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465	2023	NUREG-1465
	1.3	1.3	0.50	4.0	1.3	24.0	10.0	3.1	3.0
	Noble Gases	0.019	0.10	0.21	0.73	0.0008	0.0	0.003	0.0
	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
- Release rates (release fraction/hour) are generally smaller
- Larger Te group release magnitude prior to lower head failure



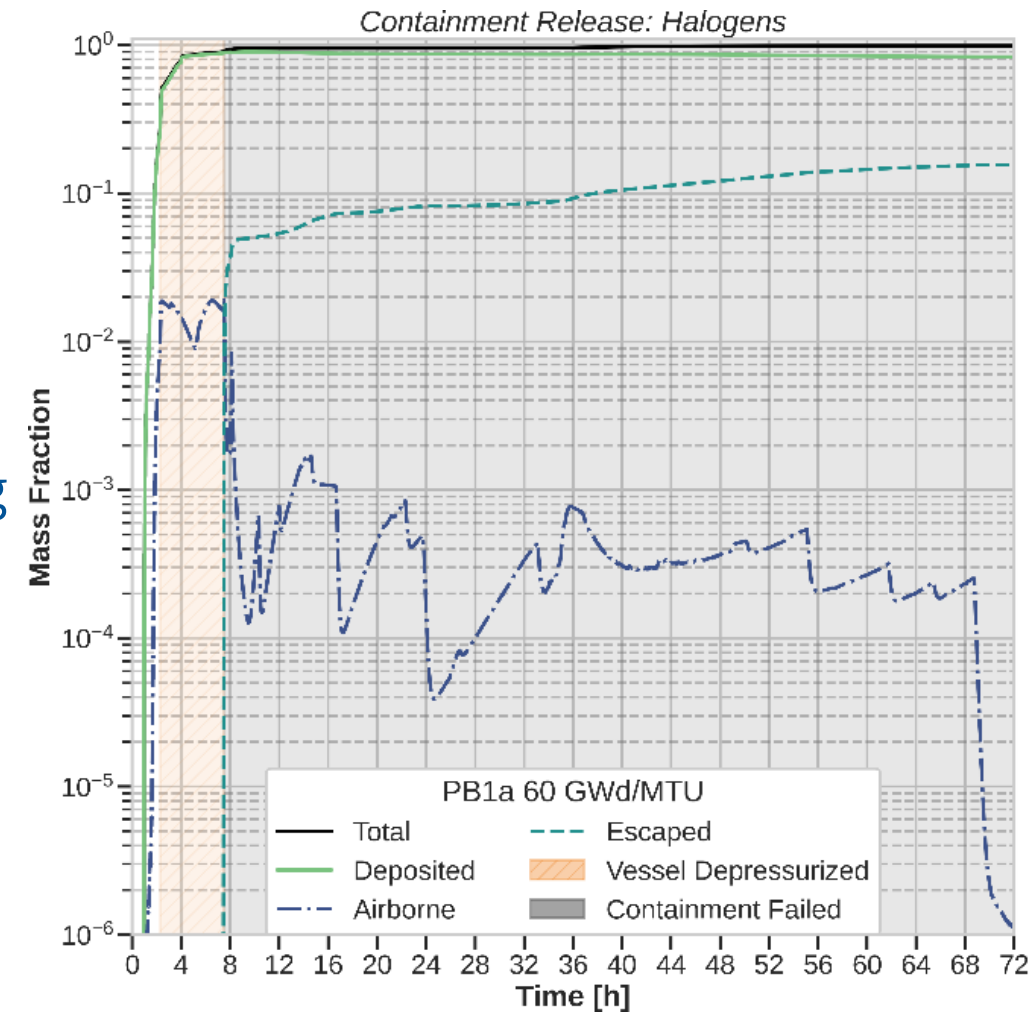
Deep Dive

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



"Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183

SAND2023-01313 Peer Review
Assessment (ER/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 be 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



BWR and PWR core damage accident scenario identification



Develop radionuclide inventory and decay heat using the SCALE code package



Perform accident progression and source term analyses using MELCOR



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



- 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 – <i>from SAND2011-0128</i>	End Criteria – <i>from SAND2011-0128</i>
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

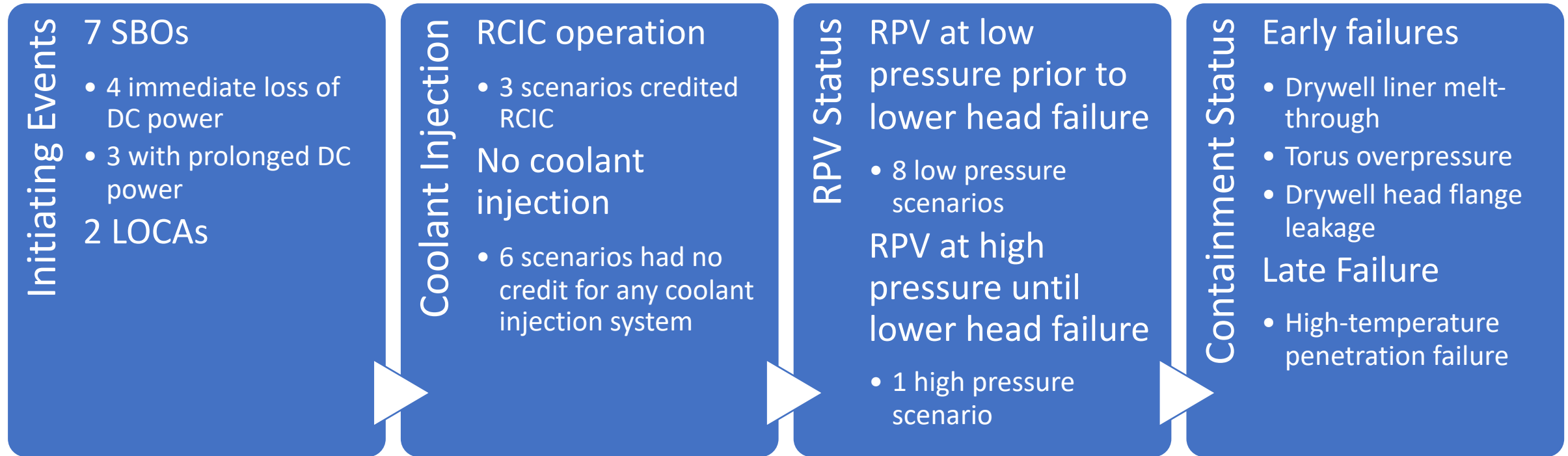


- NUREG-1560 – “Individual Plant Examination Program”
- Based on SAND2011-0128 accident selection
 - Consistent with NUREG 1560 IPE results
- Representative accident sequences similar to those selected for NUREG-1465
 - Provides coverage of all major sequences
- Incorporates SBO, LOCA and ATWS scenarios and range of mitigating system operation

Peer Review Findings

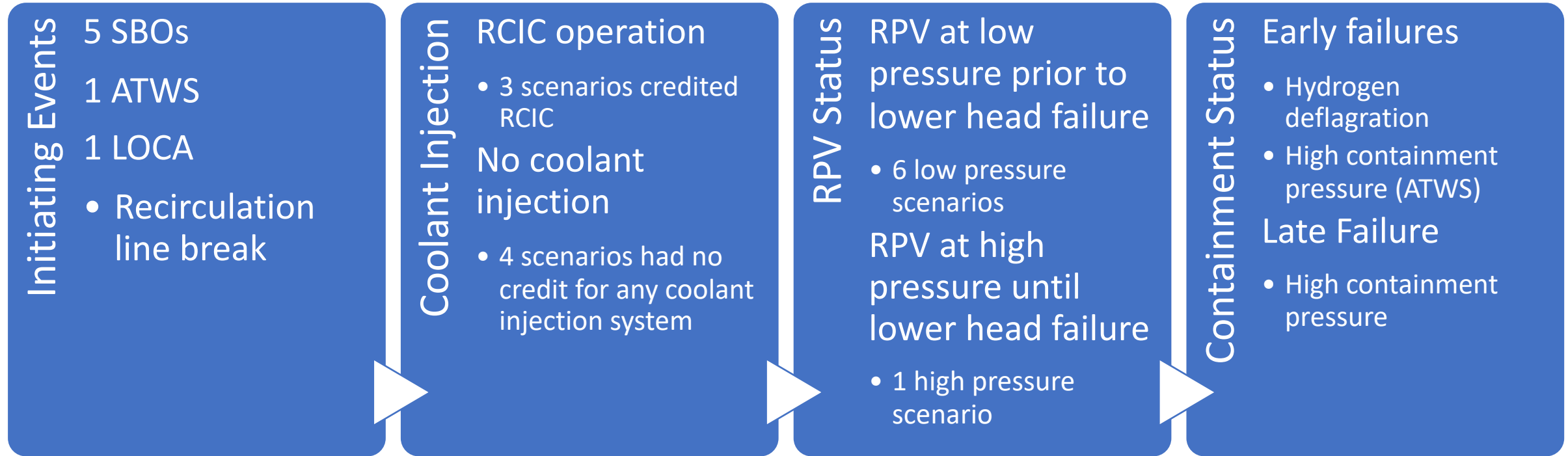
- More recent PRA studies may potentially show different core damage contributors
- For the intended applications the scenarios used in the current [SAND2023-01313] appropriate with regards to the progression of severe accidents, radionuclide release and transport.

Peach Bottom Accident Scenarios



Containment failures occurred at or after lower head failure

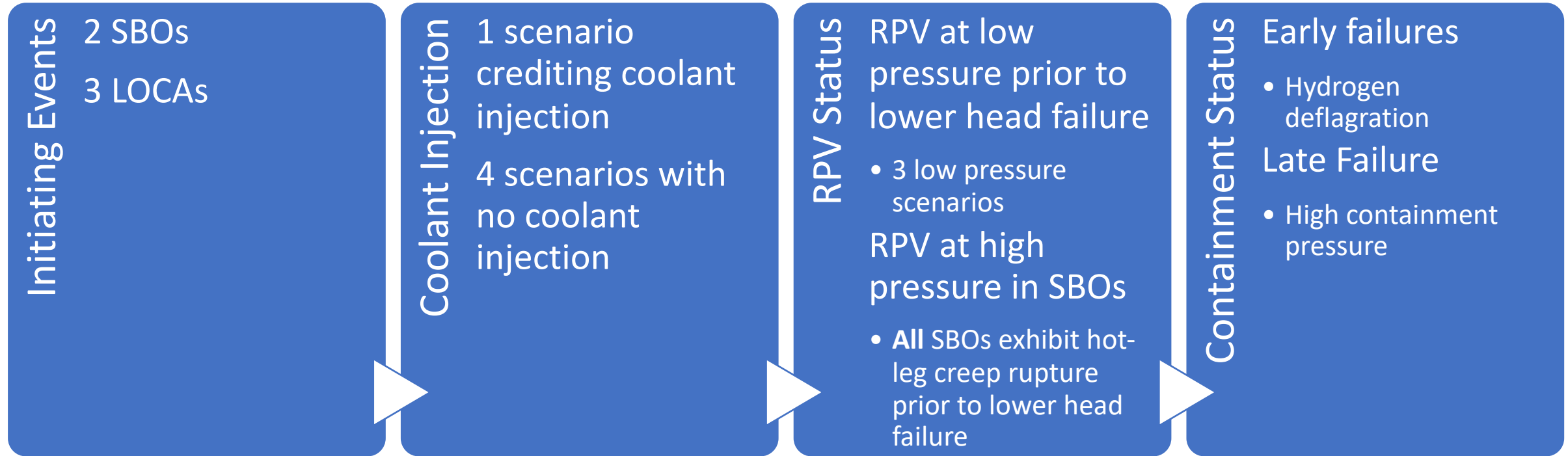
Grand Gulf Accident Scenarios



Containment failures generally occurred at or after lower head failure



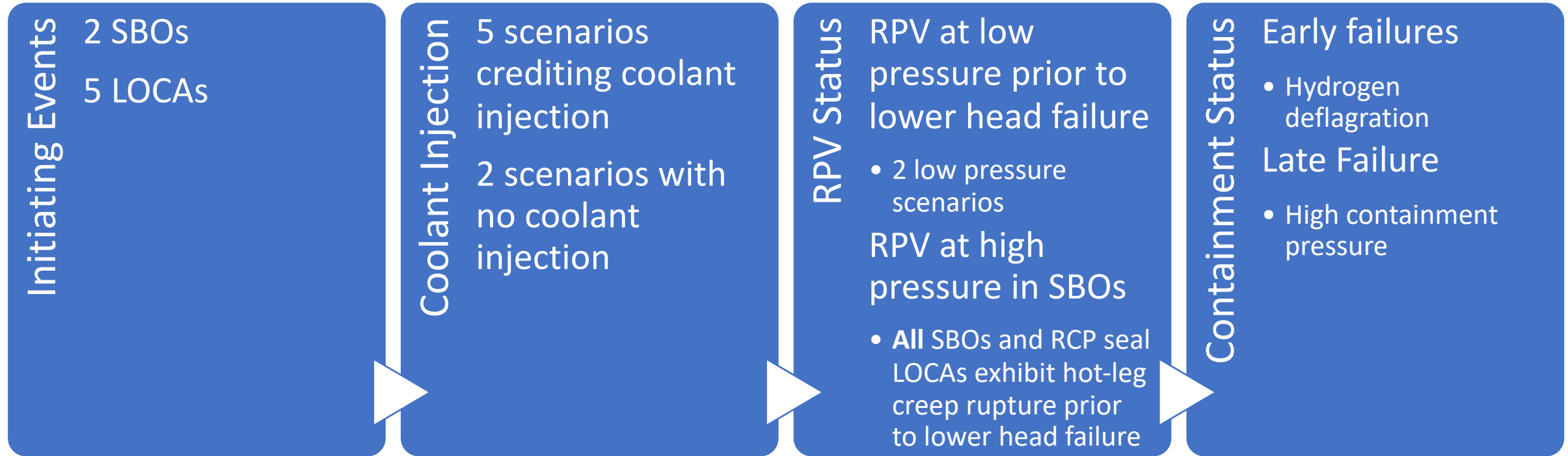
Surry Accident Scenarios



Containment failures occurred
at or after lower head failure



Sequoyah Accident Scenarios



Containment failures occurred
at or after lower head failure



BWR Radionuclide Inventories



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

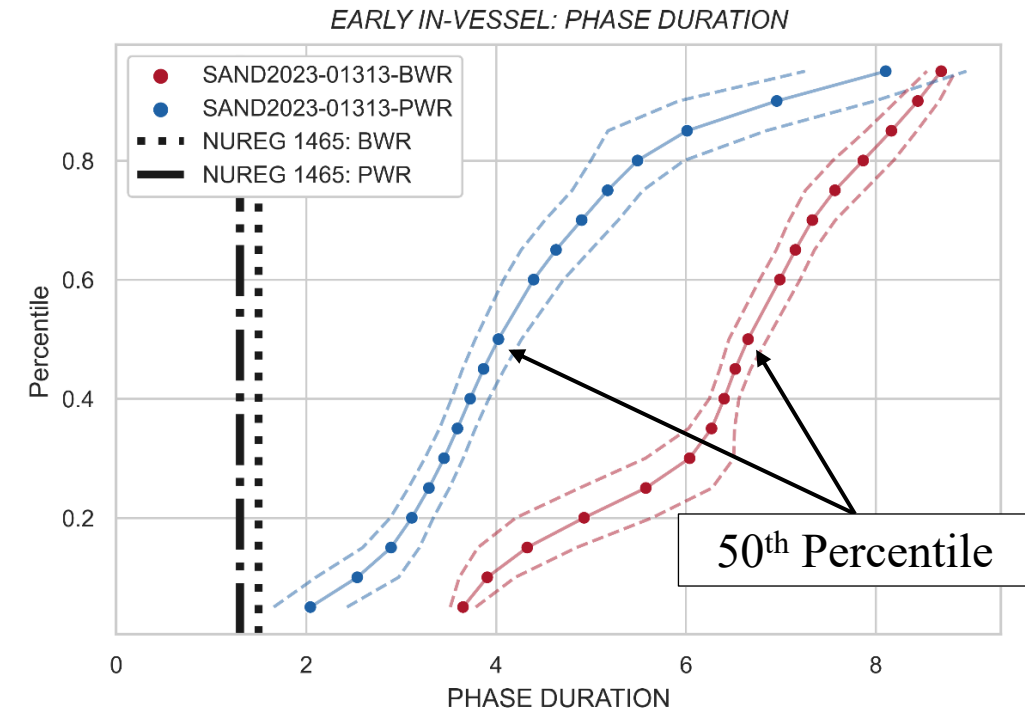


- Relative contribution of accident sequences to total BWR/PWR CDF not changed by cores with extended enrichment HBU
- Dominant uncertainty from range of possible accidents that could be realized (i.e., aleatory uncertainty)
 - Phenomenological (or epistemic) uncertainty not incorporated into BWR/PWR 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

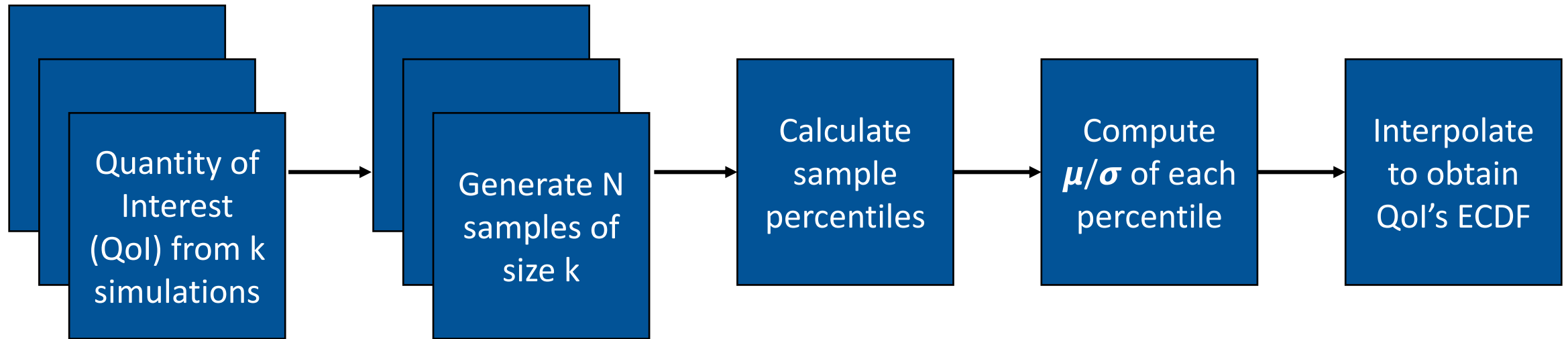


*Dashed colored lines illustrate confidence intervals spanning $\pm$ standard deviation (σ) at each percentile

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



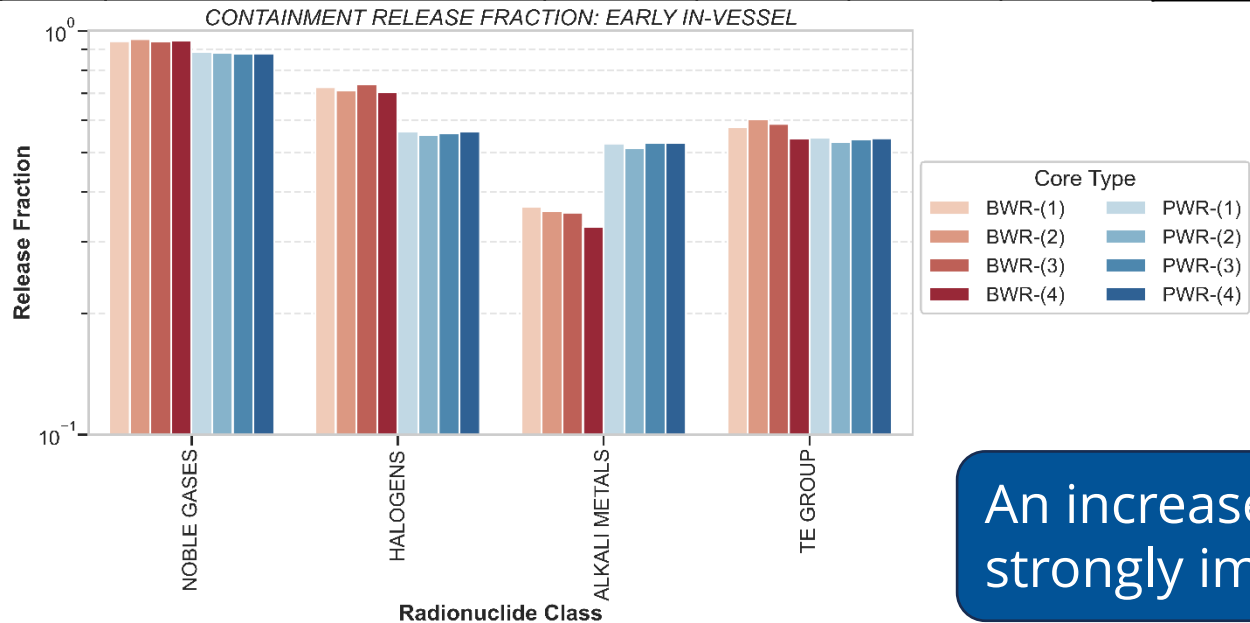
- Objective
 - Extend the NUREG-1465 alternative source term to address LWRs with cores designed to utilize HBU fuel with varied fuel enrichments
- Plants analyzed
 - BWR: Mark I containment (Peach Bottom) and Mark III containment (Grand Gulf)
 - PWR: Ice Condenser containment (Sequoyah) and Large-dry 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				PWR		Early In-vessel			
	Core Type	(1)	(2)	(3)	(4)		Core Type	(1)	(2)	(3)	(4)
	Phase Duration	6.7	6.3	6.5	6.3		Phase Duration	4.0	3.8	4.2	3.8
	Noble Gases	0.94	0.96	0.94	0.94		Noble Gases	0.93	0.92	0.91	0.92
	Halogens	0.71	0.71	0.76	0.71		Halogens	0.57	0.56	0.57	0.58
	Alkali Metals	0.31	0.31	0.31	0.26		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



Study	Gap Release			Early In-vessel		
	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

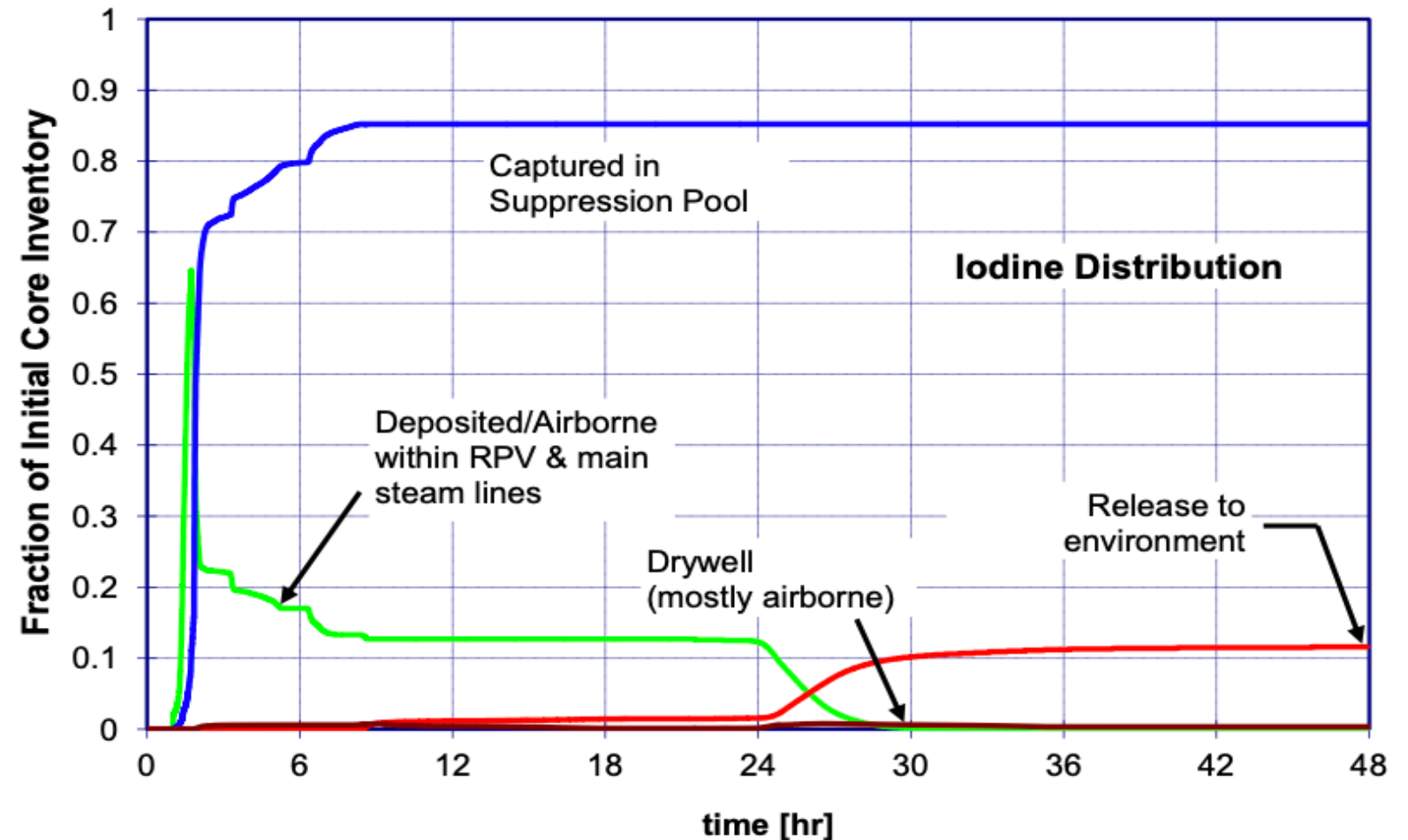
BWR In-Containment Source Terms Consistent with SOARCA



- 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



Study	Gap Release			Early In-vessel		
	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

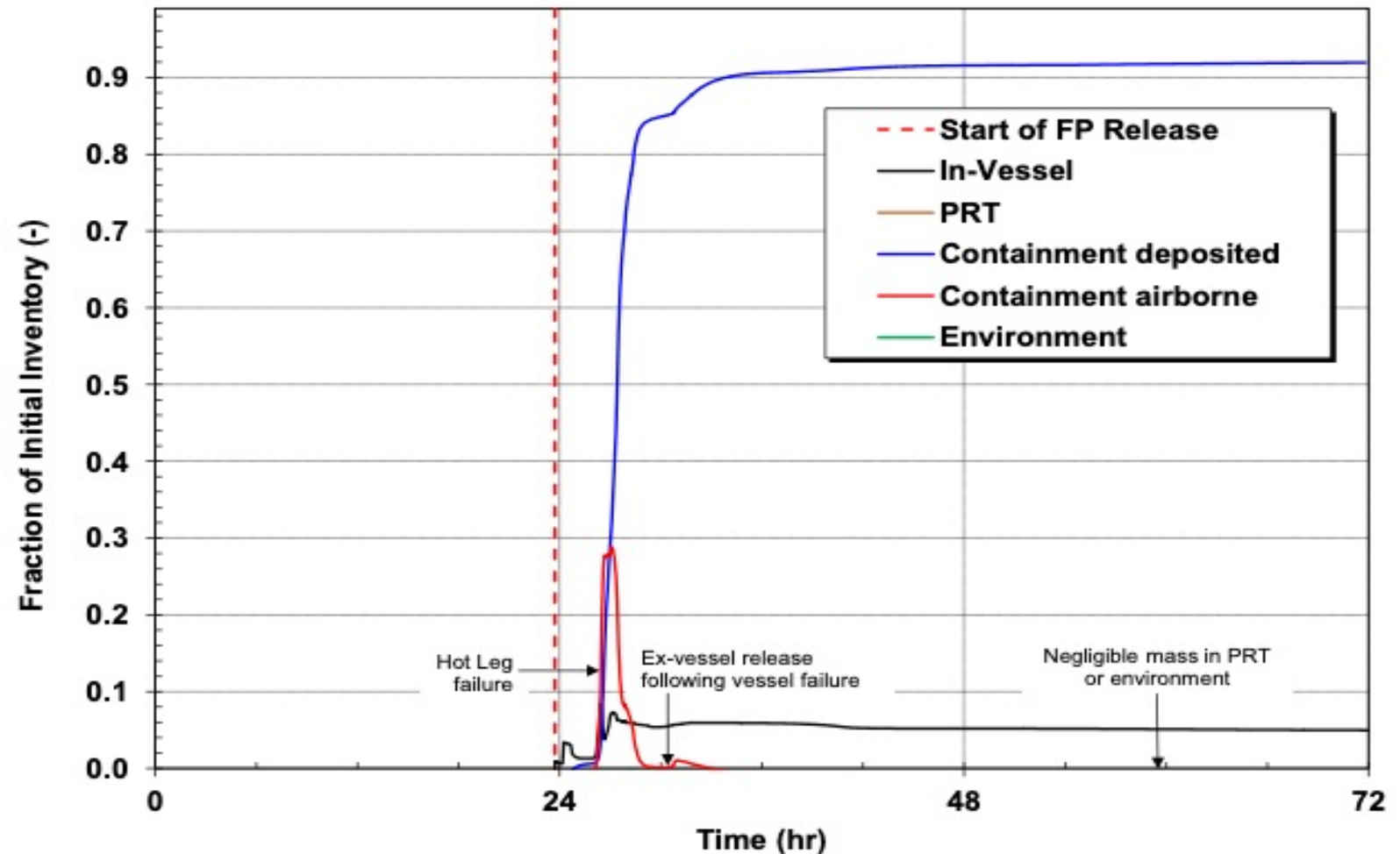
PWR In-Containment Source Terms Consistent with SOARCA



- 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



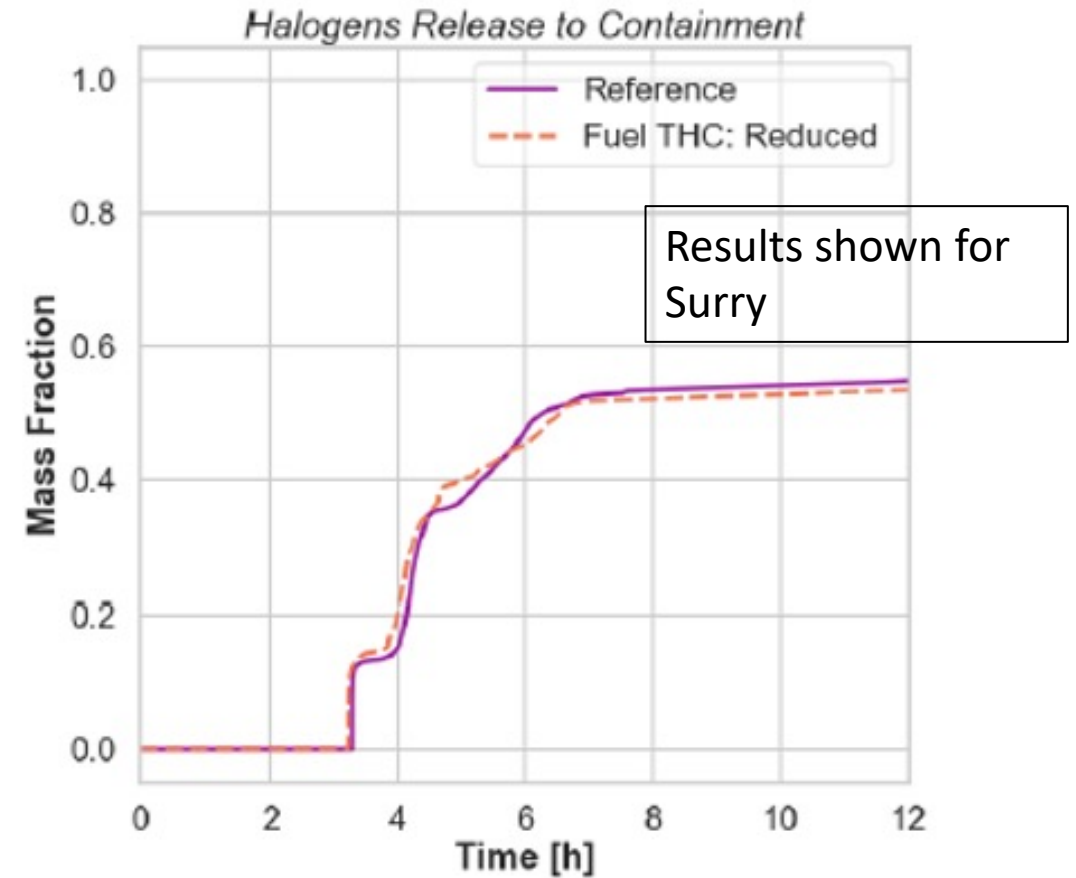
Study	BWR			PWR		
	Early In-vessel			Early In-vessel		
	2023	2011	NUREG-1465	2023	2011	NUREG-1465
Noble Gases	0.14	0.12	0.63	0.21	0.21	0.73
Halogens	0.11	0.059	0.17	0.16	0.082	0.27
Alkali Metals	0.047	0.016	0.13	0.15	0.051	0.19
Te Group	0.091	0.049	0.033	0.15	0.067	0.038
Ba/Sr Group	0.0009	0.0006	0.013	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



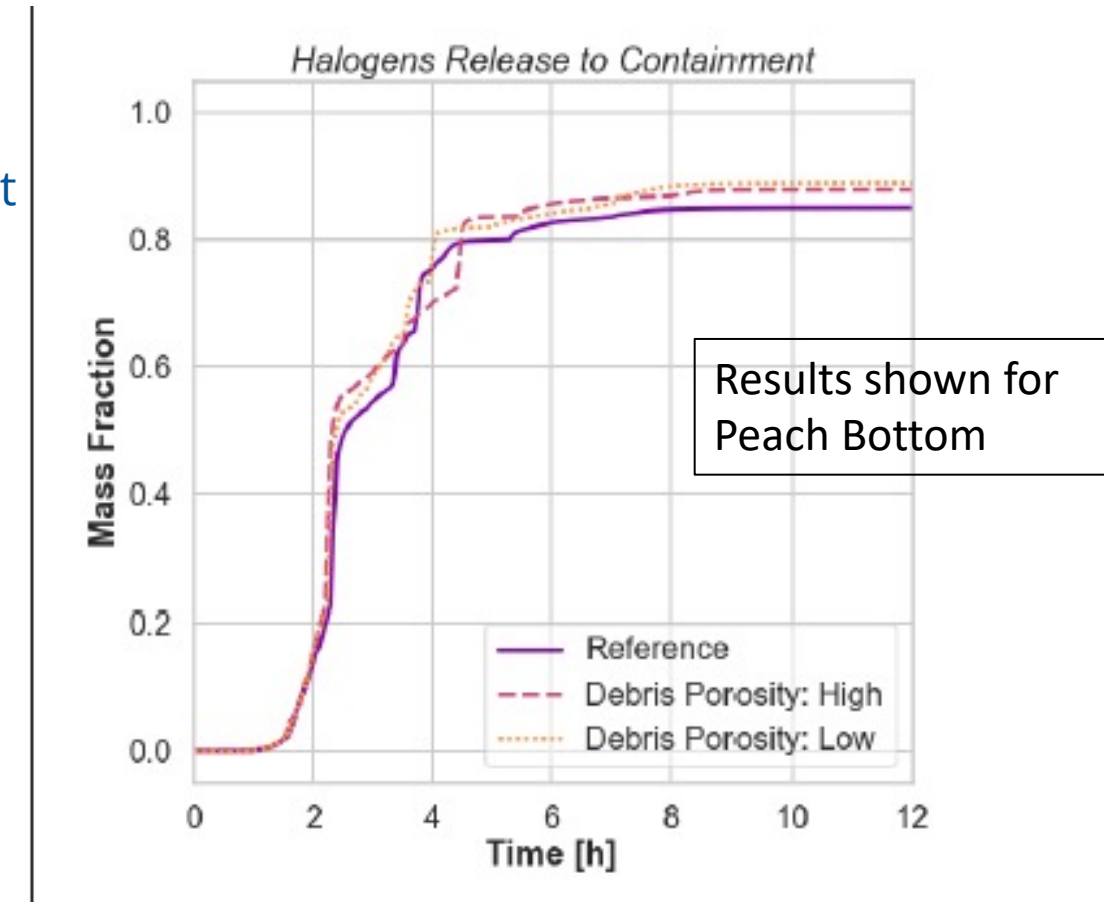
No impact from variation of fuel thermal conductivity

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

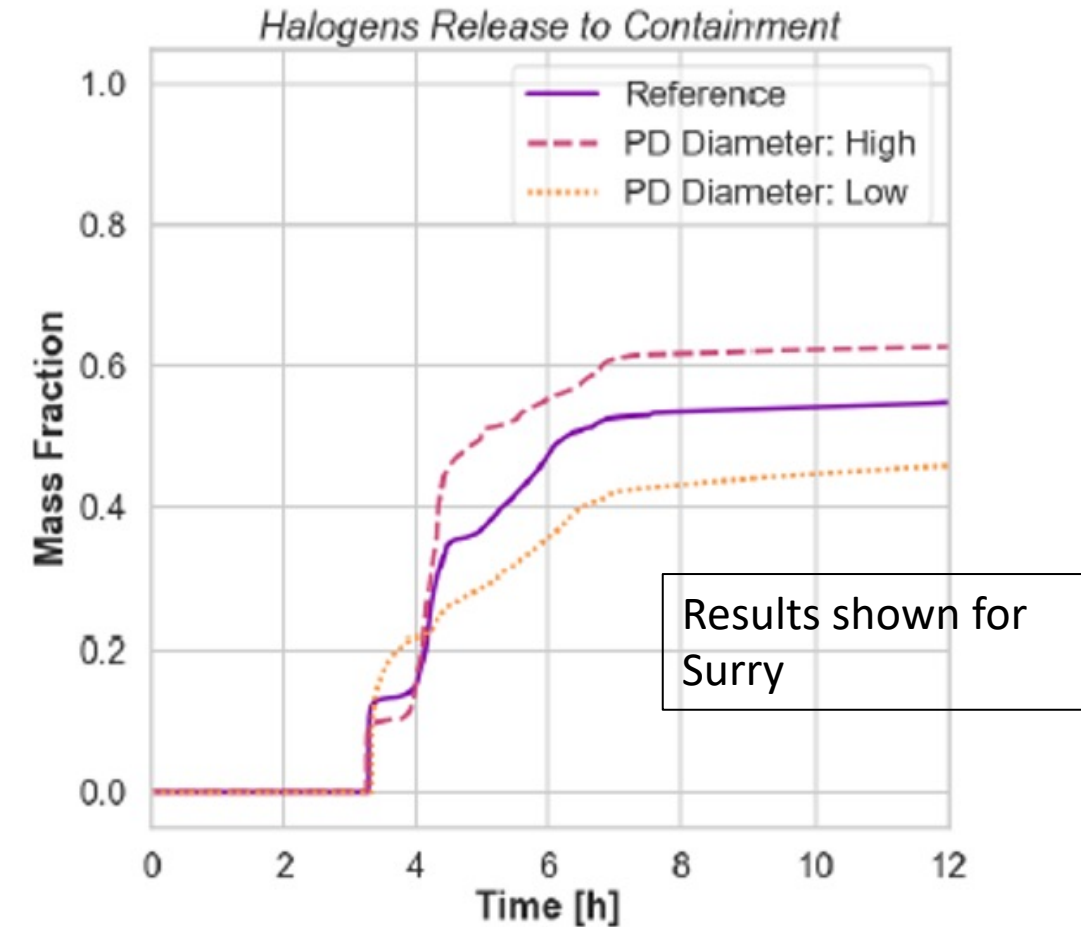


No impact from variation of in-vessel particulate debris porosity

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



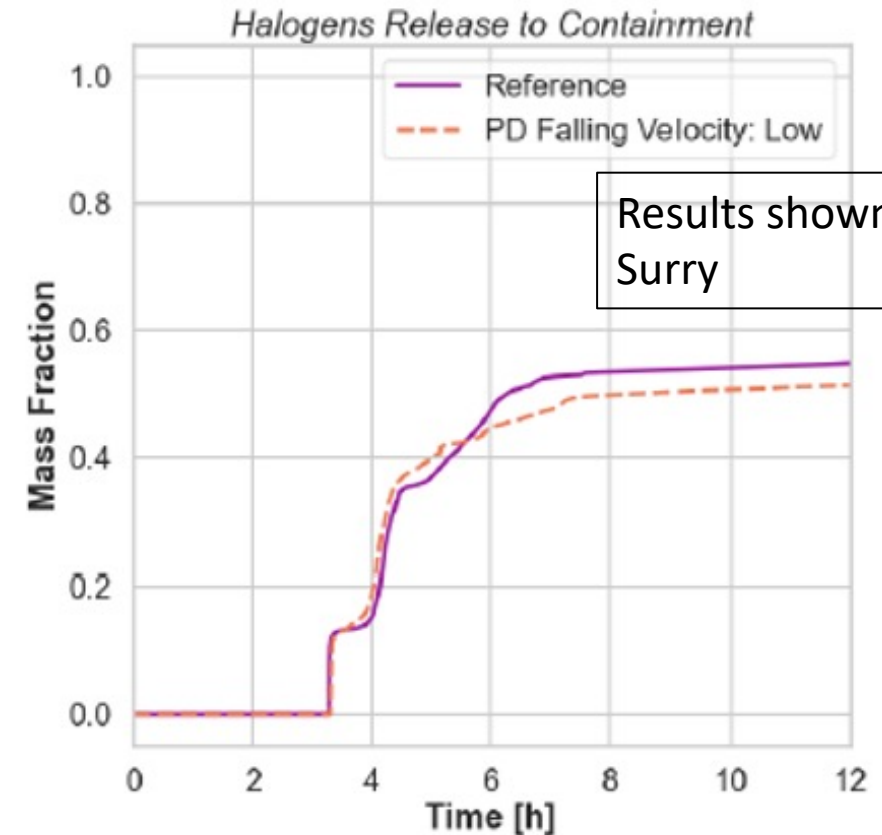
Variation in particulate debris diameter impacts in-containment source term
Impact smaller than changes across accident scenarios

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] <i>Peach Bottom</i>	In-Vessel Particulate Debris Fall Velocity [m/s] <i>Surry</i>
Reference	0.94	0.094
Low	0.094	0.064



No impact on source term due to variation in particulate debris fall velocity

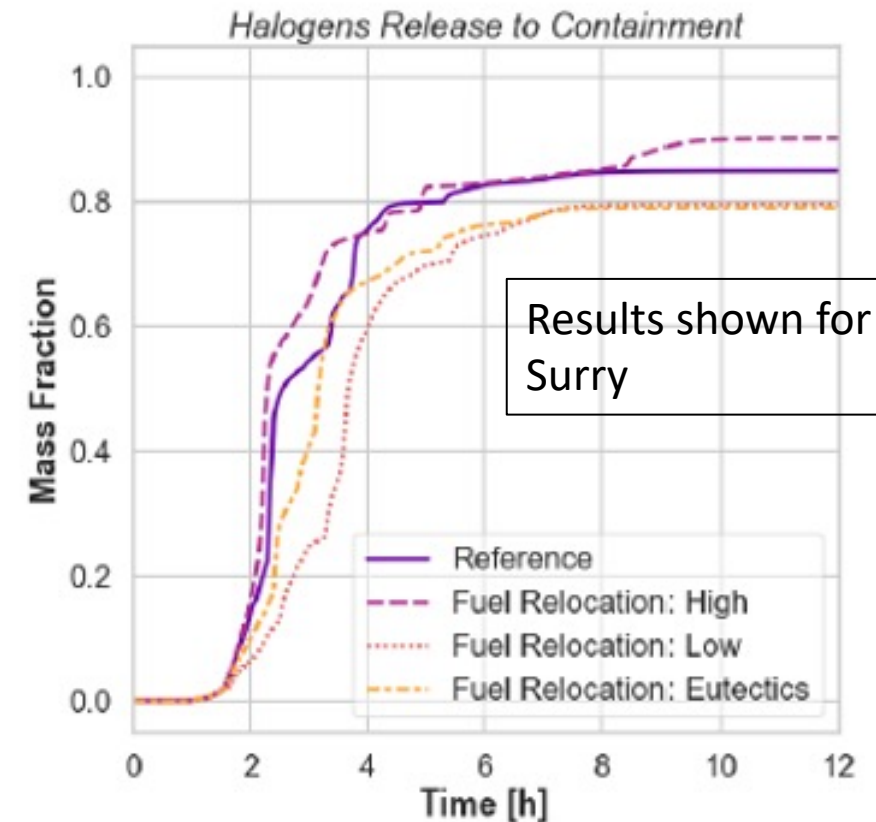
Fuel Relocation Temperature Sensitivity



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



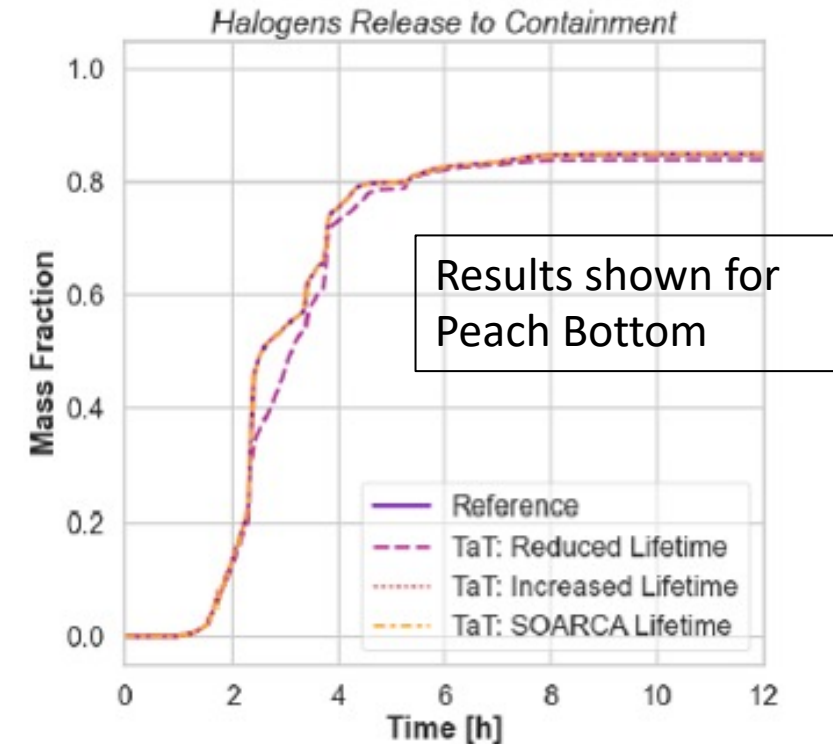
Material interactions that cause early fuel failure and can impact accident progression timings and in-containment source terms based on SOARCA uncertainty studies

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



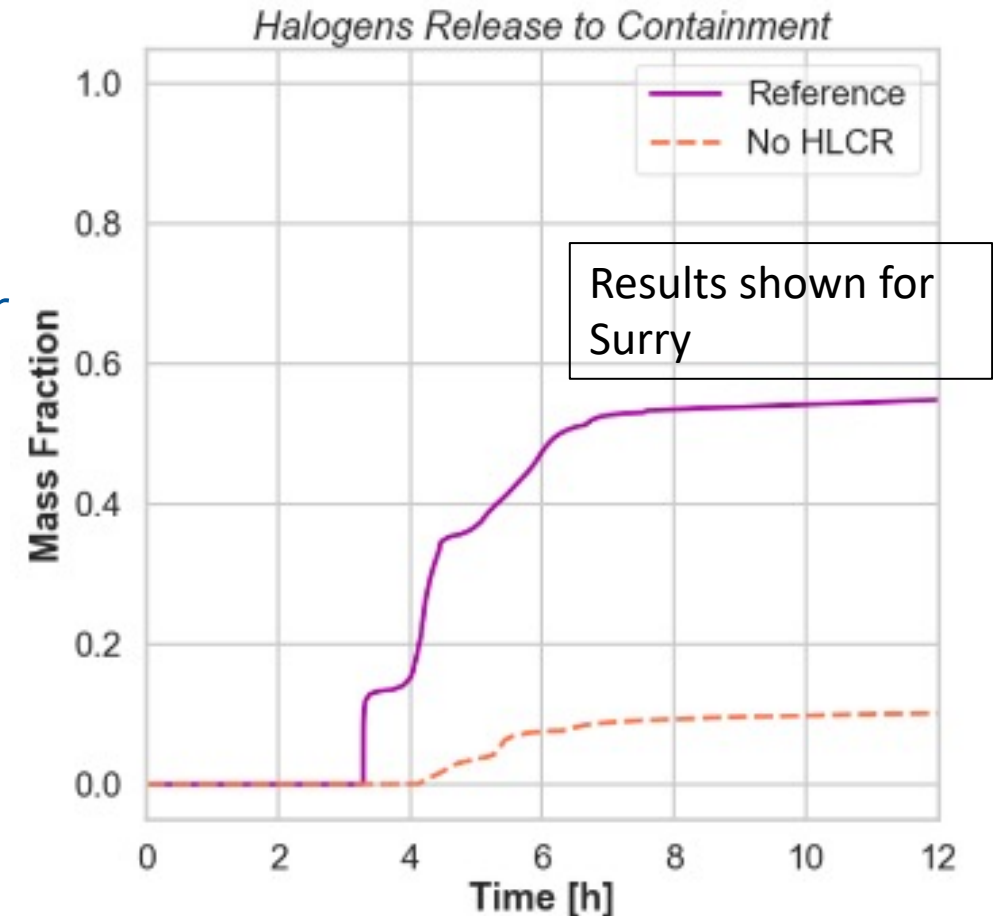
No impact due to variation of the fuel rod lifetime modeling on source term
Oxidized fuel assembly temperature failure model generally dominates

Hot Leg Creep Rupture Sensitivity

Key insight from SOARCA is potential for induced RPV pressure boundary failures

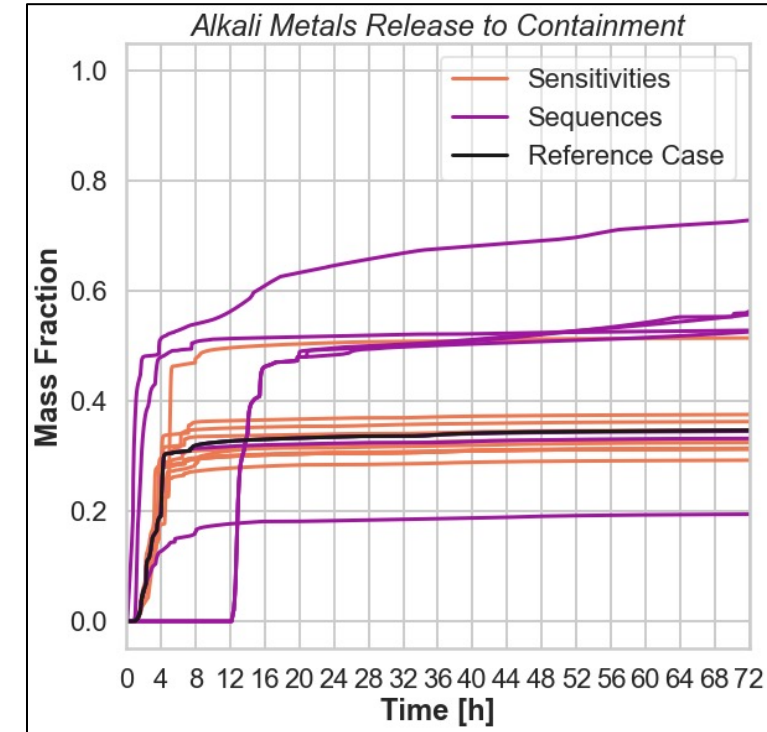
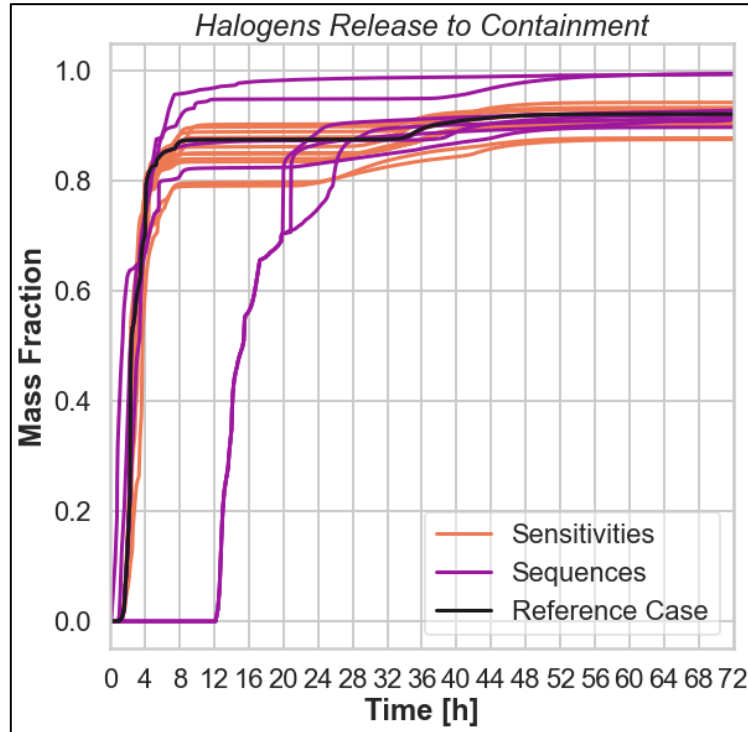
- Severe accident conditions lead to high pressure and temperature conditions at RPV boundary
- Thermally-induced hot leg creep rupture found likely for PWRs
- BWRs exhibited thermally-induced seizure of cycling SRVs

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 REPORTSAND2023-01313
Printed April 2023**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**

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



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

- 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 ingress 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



Release Category	Gap Release		Early In-vessel	
	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

**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

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

**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

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!



Backup Slides

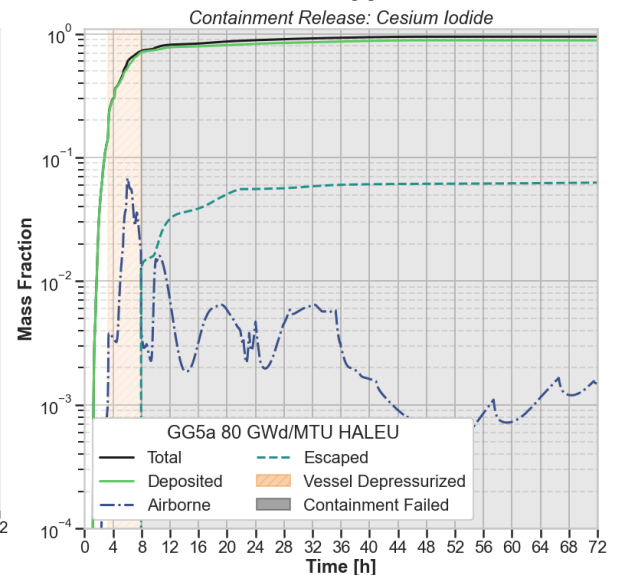
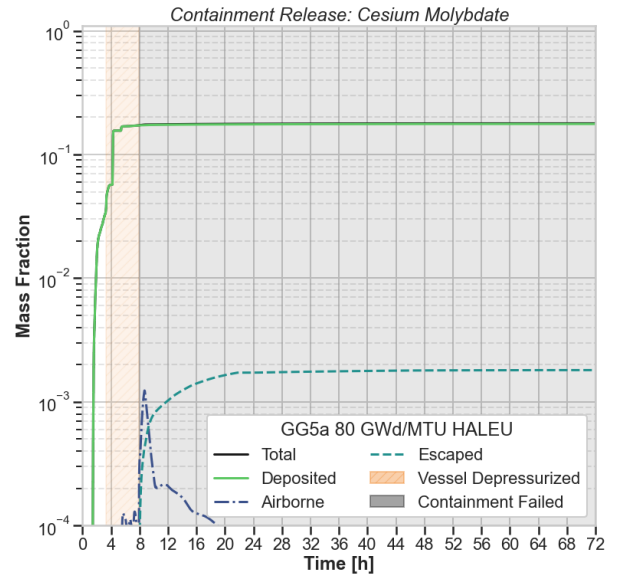
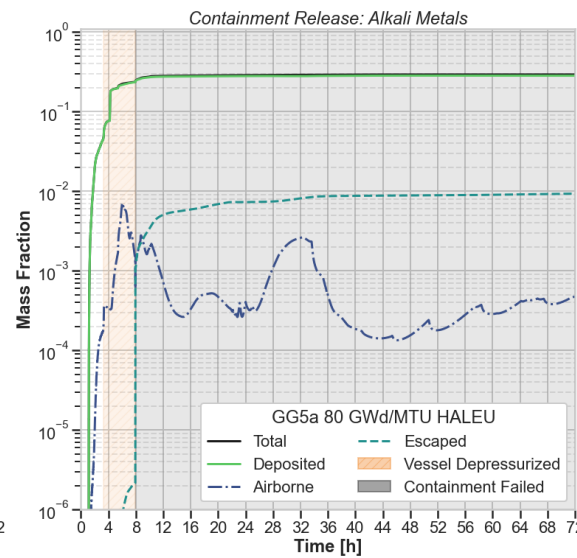
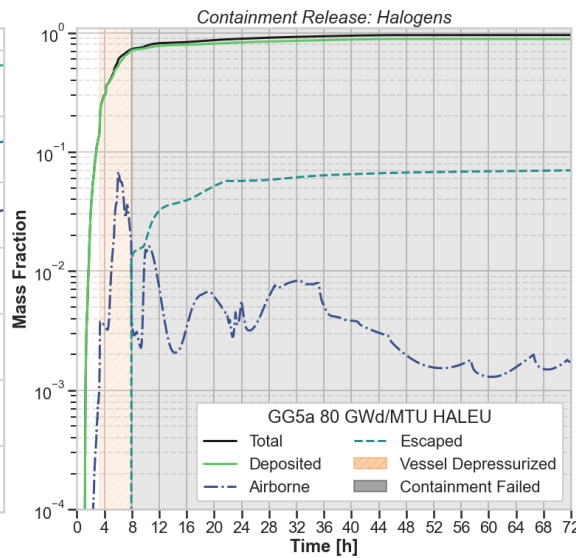
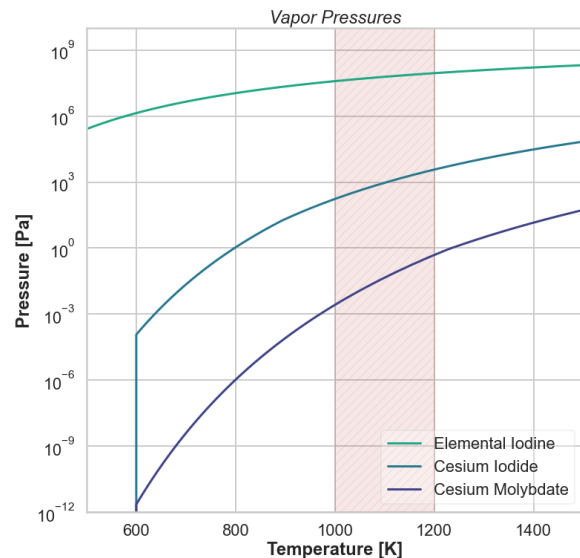
Acronyms



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



- 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 heat 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

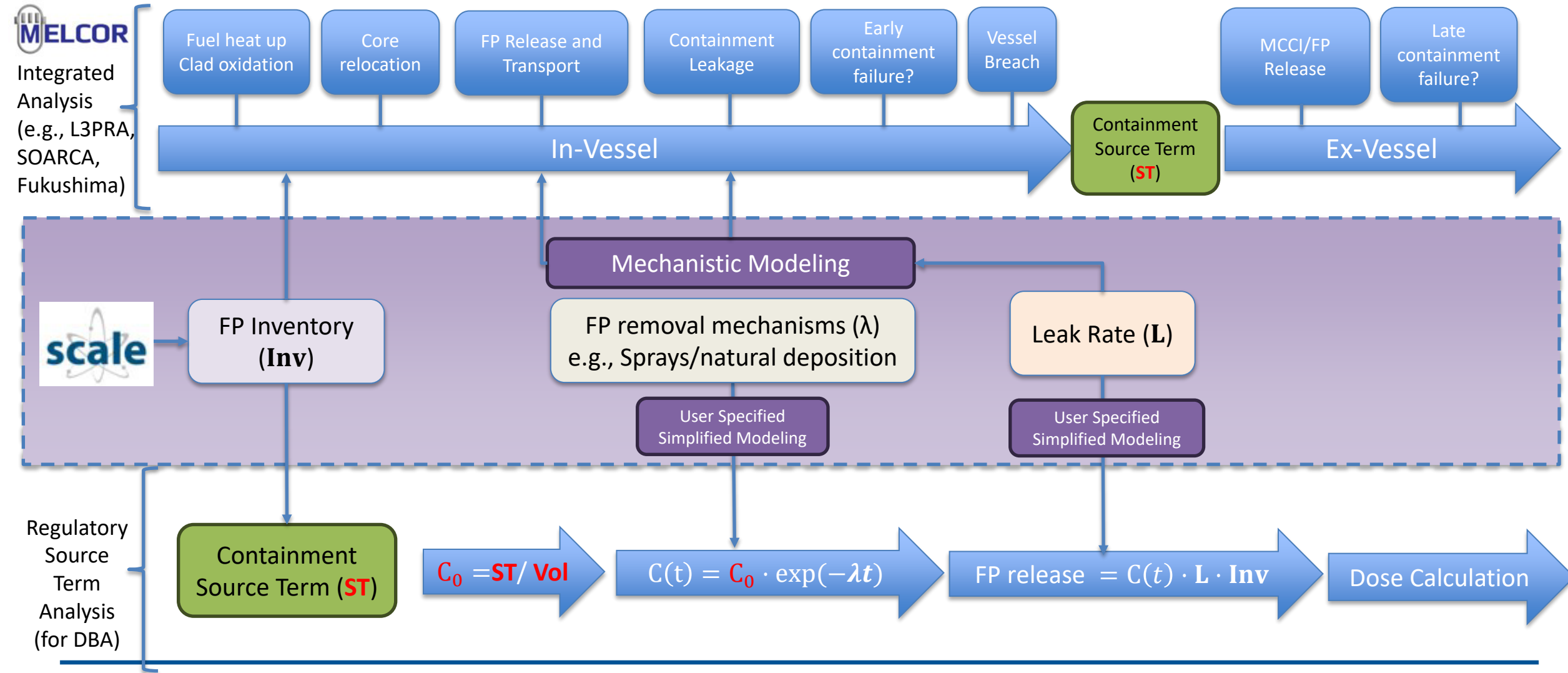
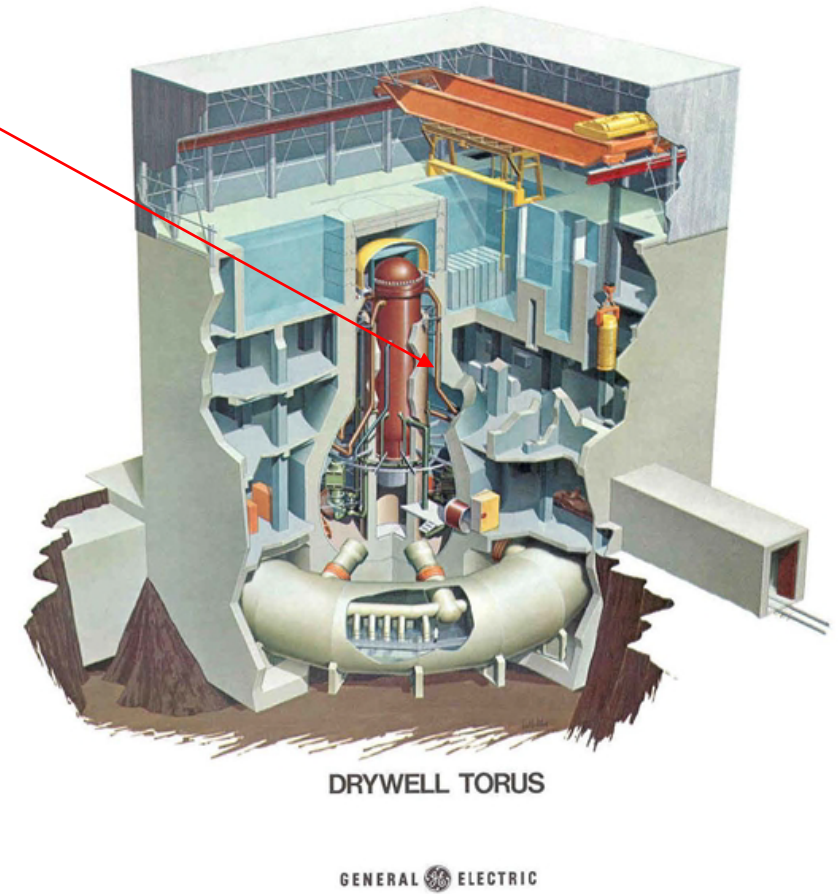
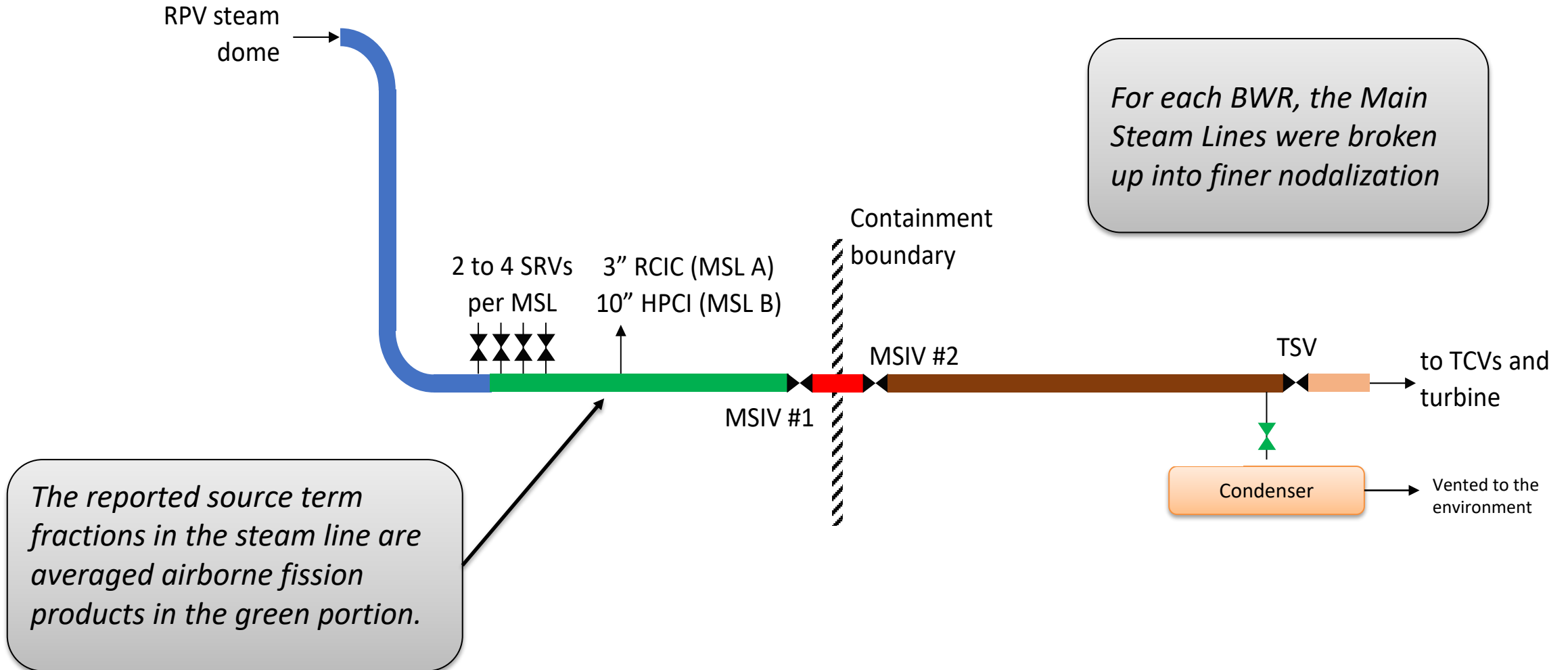


Illustration of BWR Modeling Practices

Area with refined modeling



New BWR Main Steam Line (MSL) Modeling



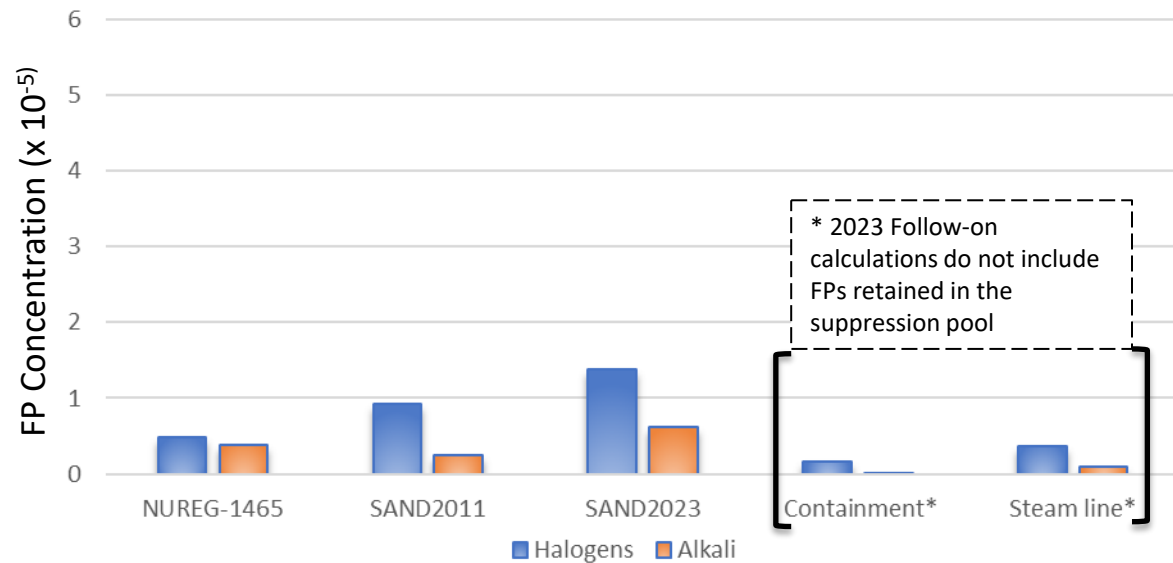
BWR Source Term (ST) Inventory Fractions – Early In-Vessel

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

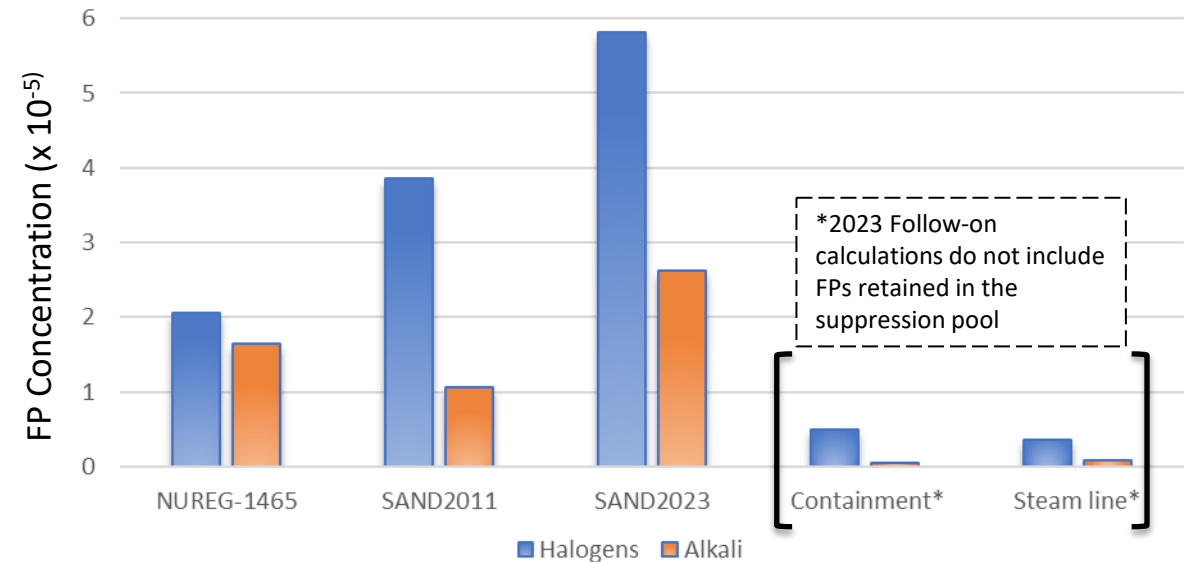
BWR Example Fission Product (FP) Concentrations (C_0)

$$C_0 = \text{ST} / \text{Vol}$$

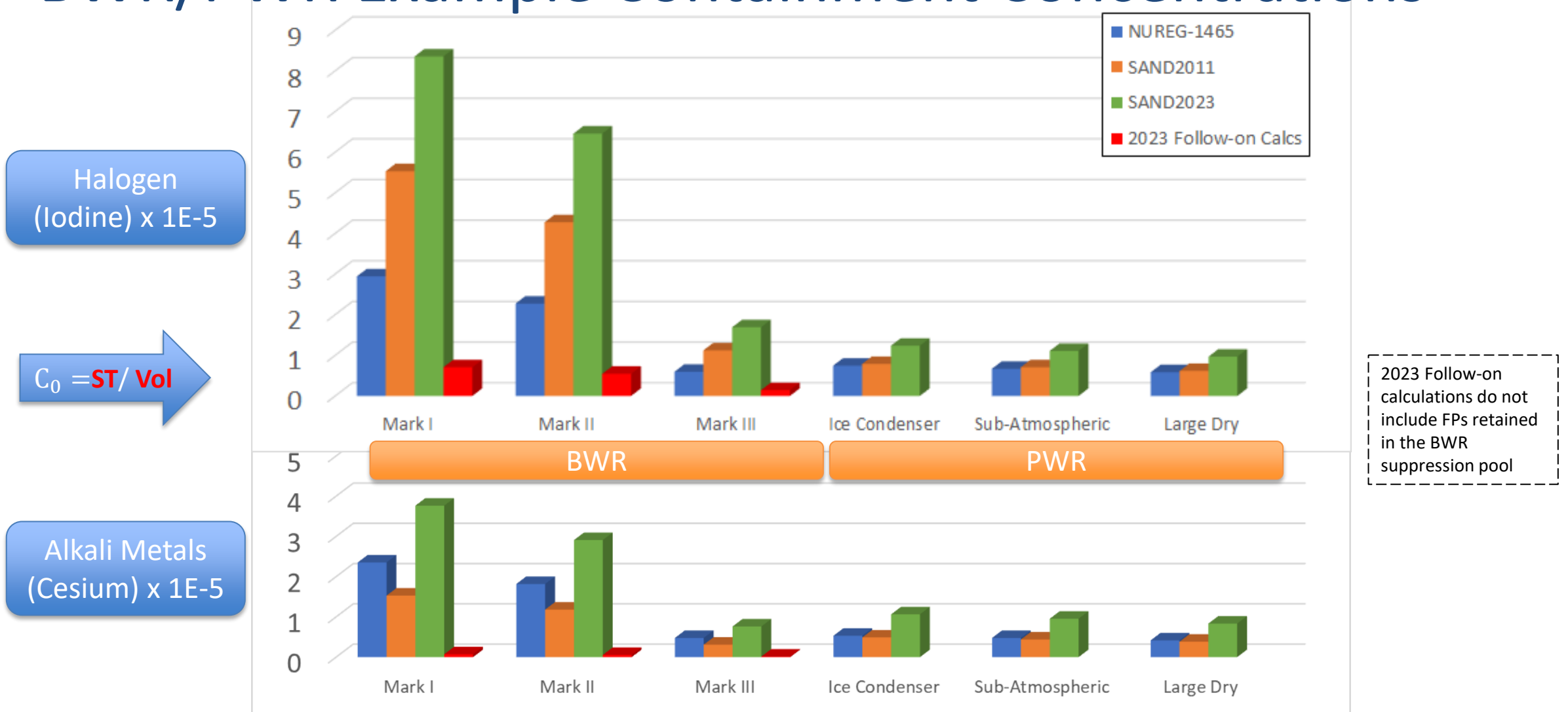
Grand Gulf Concentrations



Peach Bottom Concentrations



BWR/PWR Example Containment Concentrations



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).**

Release Category	Gap Release		Early In-vessel		Total (end of 72 hours)	
	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