Official Transcript of Proceedings

NUCLEAR REGULATORY COMMISSION

Title:	Advisory Committee on Reactor Safeguards Future Plant Designs Subcommittee
Docket Number:	(n/a)
Location:	teleconference
Date:	Thursday, February 17, 2022

Work Order No.: NRC-1859 Pages 1-251

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

(ACRS)

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FUTURE PLANT DESIGNS SUBCOMMITTEE

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THURSDAY

FEBRUARY 17, 2022

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The Subcommittee met via Teleconference, at 9:30 a.m. EST, David A. Petti, Chairman, presiding.

COMMITTEE MEMBERS:

DAVID A. PETTI, Chairman

RONALD G. BALLINGER, Member

VICKI M. BIER, Member

CHARLES H. BROWN, JR. Member

VESNA B. DIMITRIJEVIC, Member

GREGORY H. HALNON, Member

WALTER L. KIRCHNER, Member

JOSE MARCH-LEUBA, Chairman

JOY L. REMPE, Member

MATTHEW W. SUNSERI, Member

ACRS CONSULTANTS:

STEPHEN SCHULTZ

DENNIS BLEY

DESIGNATED FEDERAL OFFICIAL:

MIKE SNODDERLY

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

P R O C E E D I N G S

9:30 a.m.

CHAIRMAN PETTI: Okay, committee members, interested staff, and stakeholders. The meeting will come to order.

This is a meeting of the Advisory Committee, on the Active Safeguards Future Plant Designs Subcommittee.

I'm Dave Petti, lead member for the meeting.

Members in attendance today are Ron Ballinger, Vicki Bier, Greg Halnon, Jose March-Leuba, Matt Sunseri, Walt Kirchner, Charlie Brown, consultant Dennis Bley is on with us as well.

I do not see Steve Schultz yet, but I do expect him. And, I do not see Member Dimitrijevic.

MEMBER DIMITRIJEVIC: I'm here.

CHAIRMAN PETTI: Mike Snodderly is the designated federal --

(Simultaneous speaking.)

MEMBER DIMITRIJEVIC: I'm attending.

CHAIRMAN PETTI: Oh, thank you, Vesna.

MEMBER DIMITRIJEVIC: I don't know why you don't see me. They didn't see me yesterday, but I'm here.

CHAIRMAN PETTI: Ahh, great.

DR. SCHULTZ: This is Steve, Dave. I'm here.

CHAIRMAN PETTI: Okay, good. And, consultant Steve Schultz is here. Great.

Mike Snodderly is the designated federal official for this meeting.

The subcommittee will discuss with the staff the Integration of their Source Term Activities in Support of Advanced Reactor Initiatives.

This is a topic that is of interest both to the committee, and the staff. It's nice to see minds coming together and thinking that this was an important topic for us to talk, about in the context of Part 53.

The ACRS was established by statute, and is governed by the Federal Advisory Committee Act, FACA.

The NRC implements FACA in accordance with its regulations found in Title 10 of the Code of Federal Regulations, Part 7.

The committee can only speak through its published letter reports. We hold meetings to gather information, and perform preparatory work that will support our deliberations at a full committee meeting.

The rules for participation in all ACRS meetings were announced in the Federal Register, on June 13, 2019.

The ACRS section of the U.S. NRC public website provides our chat or bylaws agendas, letter reports, and full transcripts, of all full and subcommittee meetings, including slides presented there.

The agenda for this meeting was posted there as well.

As stated in the Federal Register notice, members of the public who desire to provide written or oral input to the subcommittee may do so, and should contact the designated federal official five days prior to the meeting, as practicable.

This is an MS Teams virtual meeting, and members of the public may listen in on the presentations, and committee discussion using the call in number and the conference ID number included on the agenda.

We have received no written (Audio interference) to make oral statements from members of the public regarding today's meeting.

There will be an opportunity for public comment, and we set aside 10 minutes in the agenda for comments from members of the public attending, or listening to our meetings.

Written comments may be forwarded to Mike Snodderly, the designated federal official.

A transcript of the open portions of the meeting is being kept, and it is requested that the speakers identify themselves, and speak with sufficient (Audio interference) so that they can be readily heard.

Additionally, participants should mute themselves when not speaking.

I just want to note that the staff has put together a tremendous amount of technical information to convey to the committee today, in, you know, basically our one day meeting.

I'm going to be extra mindful of the schedule, to make sure we get through all the information before the end of the day.

Now, I'd like to have Larry Burkhart provide a comment about MS Teams participation, followed by Arlon, who will provide us with an overview of the agenda.

And, then John Segala will make an opening statement (Audio interference) Larry.

MR. BURKHART: Okay, thanks. Just for

everyone's notice, the ACRS is now providing the MS Teams link for all open meetings, to all members of the public.

I do notice that we have a couple folks that are tying in via phone. So, if there are any of the public attendees who need the MS Teams link, please email me at the following address: L-A-W-R-E-N-C-E, Lawrence, dot Burkhart, B-U-R-K-H-A-R-T, @NRC.GOV.

So again one more time, if any members of the public who don't have the MS Teams link would like it, please email me at lawrence.burkhart@NRC.gov.

Thanks.

MR. ACOSTA: Okay, this is Arlon Acosta. I just would like to go over quickly, over the agenda.

We're going to as Dave already said, going to have the opening remarks from John, and staff introduction.

Following him, we will continue with the presentations as follows: the History and Evolution of LWRS Source Terms; in the NRC analytical tools and past studies; SCALE and MELCOR in non-light water reactors reference analysis.

And, then somewhere in between that presentation, we will have a break that Dave Petti will tell us.

And, we may need to continue with that item, which is item 4 is SCALE and MELCOR non-light water reactor reference plant analysis.

And, then subsequently, we will continue with the NuScale EPZ Sizing Methodology Topical Report Revision 2 presentation; light water reactor SMR design certification source term approach; source term approach for the early non-light water movers.

And then sometime as designated by Dave, we'll have a lunch break, and Accident and followed by the other presentation on Accident-Consequence-related regulatory, regulation activities.

And, again, place to be determined for a break, and four more, two more presentations on guidance and information for developing advance reactor source term.

And, guidance for developing advanced reactor source term as far as long-term is concerned.

And, as Dave mentioned, an opportunity for public comment and member discussion, and subsequently to adjourn.

John Segala?

MR. SEGALA: Thank you, Arlon. Hopefully you can hear me.

I'm John Segala, Acting Deputy Director of the Division of Advanced Reactors and Non-Power Production Utilization Facilities in the Office of Nuclear Reactor Regulation.

We're pleased to be here today for this important topic. Determining source term is a critical component in NRC's licensing of advanced reactor designs.

Over the past few years, we have discussed source term with the ACRS in a variety of contexts, including emergency preparedness rulemaking, licensing modernization project, Part 53 analytical tool development, the NuScale review, et cetera.

The ACRS has written several letters and raised a number of issues in this area, that development of a design specific source term will require substantial work.

The new reactor designers may need expanded guidance in this area, and this could help make the reviews more efficient. And, staff efforts need to be coordinated across the various source term related activities.

So, we proposed to have this separate ACRS meeting today, to focus on source terms to address these comments in a more holistic and coordinated way.

We understand that ACRS may want to write a letter on this topic, and have tentatively scheduled a discussion on this topic during the March 2nd ACRS full committee meeting.

Source term has evolved over the past 60 or so years. Historically for large light water reactors that are operating today, the NRC developed a source term for calculating offsite doses for citing, and provided it to the reactor designers.

For example, the TID-14844 are source term.

To facilitate the use of a risk informed performance based licensing approach such as the licensing modernization project, design and/or scenario specific source terms will need to be developed by the applicants.

With respect to guidance for source term development, the NRC staff has completed a number of source term related projects, and more work in ongoing. All of which we will be presenting to the ACRS today.

Although we acknowledge that developing design and scenario specific source term will require substantial work, the staff believes that there is sufficient guidance available for applicants to develop design and scenario specific source terms, and for the NRC staff to complete its technical review.

We will consider the need for additional source term guidance in the future, based on our experience with the ongoing pre-application and licensing interactions that we have, that we are having with applicants over the next couple years. Such as Kairos, X-energy, TerraPower.

We have recently developed a new public webpage focused on source term for nuclear power reactors, which contains information on source term, and provides links to a number of source term guidance documents, videos, presentations, and other associated references such as SECY papers, staff requirements, memorandums, NUREGs, contractor reports.

We plan to showcase the webpage for you today.

In addition to this available guidance and information on source term, the staff encourages applicants to engage in pre-application activities, to seek early NRC feedback on their source term methodology.

And, we have been having such engagements with several pre-applicants, which we will discuss today.

Pre-application engagement on topics such as source term, will support a more efficient application review.

If you could go to the next slide? Thanks.

At this point, I'd like to introduce the NRC team that will be making the presentations today. We have four NRC staff from the Division of Advanced Reactors and Non-Power Production and Utilization Facilities, in the Office of Nuclear Reactor Regulation.

That includes Michelle Hart, Jason Schaperow, Bill Reckley, Tim Drzewiecki. We also have Mark Blumberg, from the Division of Risk Assessment in the Office of Nuclear Reactor Regulation, and we have Hossein Esmaili, in the Division of Systems Analysis, in the Office of Nuclear Regulatory Research.

We are looking forward to hearing from the ACRS today on source term, and any insights and feedback you may all have.

That completes my opening remarks.

MR. BLUMBERG: Thank you, John. My name is Mark Blumberg, and I'll be giving you a presentation, which is a brief overview of the historical use, and the evolution of light water reactor source terms in the NRC regulatory process.

The NRC exists to protect the health and safety of the public from the accidental release of fission products.

One of the ways the NRC staff and licensees determine what measures and barriers are needed, is to perform dose analyses.

Specifically, these conservative analyses address the situation where we were wrong about the success of a facility's response to events or accidents.

Dose analyses provide an effective way to account for the uncertainties in equipment, and human performance.

In particular, these analyses account for the unlikely events that involve unknown, or unforeseen failure mechanisms, or phenomena, which because they are unknown or unforeseen, are not reflected in the PRA or traditional engineering analyses.

A critical component of the dose analysis is the source term. In the Regulations for Part 50, the NRC defines the source term as the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of fission product inventory in the fuel, as well as the physical and chemical form and timing of their release.

Please go to slide 6 now.

Since we're talking about history in this presentation, rather than providing an outline, a timeline is presented in this slide instead.

This slide presents a timeline of the documents and events, surrounding the development of the light water reactor source terms.

Since we are providing an overview today, this timeline highlights a few of the more important documents and events pertinent to our regulatory approach.

The text in the boxes are color coded. Boxes with text written in black describe important documents such as the NUREG-1465 document, shown here on the timeline in 1995.

NUREG-1465 supports our most recent source term and regulatory guidance.

Boxes with the text written in red describe the important events. The events are located below the timeline, and provide the timeframe for the first critical pile, the 1954 Atomic Energy Act that allowed civilian nuclear power development, and four accidents that influenced the development of our regulatory source terms.

I'll discuss many of these documents and events in this presentation, and this timeline can be used to provide a perspective of the timing of these events and documents.

Go to slide 7, please.

In the 1940s and the 1950s, the first government reactors were placed far from population centers, and therefore, siting was not a critical issue.

Over time when the utilities wanted to use reactors for power, transmission costs were a consideration. Therefore, utilities sought to put reactors closer to population centers that they served.

Because of their potential hazard that reactors posed, reactors with containments were proposed to provide defense and depth against accidents.

These reactor containments provide an extra barrier to prevent radioactivity from reaching the environment, in the unlikely scenario that the reactor coolant system failed, and the fuel melted.

To determine the acceptability of these designs, at the time the Atomic Energy Commission proposed using population densities to site reactors.

But public opinion judged this method was too rigid. The AEC, the Atomic Energy Commission, and the industry, ultimately decided that siting would be performed on a case-by-case basis, using dose calculations.

Please go to slide 8.

In 1962, the NRC issued 10 CFR Part 100. This performance based rule used dose calculations to evaluate the defense and depth provided by containment.

Nearly all the current operating reactors were licensed originally to 10 CFR Part 100, and TID-14844, which, as John said, provide guidance on a containment source term to be used for LOCAs involving significant fuel melt.

This source term was based upon heating fuel chips in a furnace and seeing what was released.

The source term involves the release of 100 percent of the Noble gasses, 50 percent of the iodine, and 1 percent of the other radionuclides as particles.

The iodine released is mostly elemental iodine. The remaining iodine is assumed to be particulate, and organic.

This deterministic source term is assumed to be available instantaneously, at the start of the accident.

Accident models credit mitigation of this source term by safety related systems, structures, and components. Non-LOCA source terms are also provided in Reg Guide 1.195.

You can go to slide 9 now, please.

Source term estimates under core melt conditions became of great interest shortly after the Three Mile Island, or TMI accident, when it was observed that only a relatively small amount of iodine was released to the environment.

Although the release of iodine in the containment may have been close to the TID source term, the releases to the environment were much smaller than suggested in regulatory models.

In 1981, the NRC began a major research effort to obtain a better understanding of the fission product transport and release mechanisms in LWRs, under these severe accidents.

This effort involved several national laboratories, extensive NRC staff in the NRC, and nuclear industry groups.

The cooperative research resulted in the development of the source term copackage to examine core melt progression, and fission product release and transport in LWRs.

In order to determine the accident source terms for regulatory purposes, a range of severe accidents were examined.

This work was based upon the work done in NUREG-1150. It provided an assessment of severe accident risks in five LWRs.

In addition, some source term code packets calculations and insights from the MELPOR code, were used.

These efforts provided research that confirmed that the source term release is highly dependent upon the nature of the accident, which included the accident pressures, temperatures, and release pathways.

If you want to go to slide 10 now, please.

In December of 1999, the NRC issued the final rule known as 10 CFR 5067. This rule is also known as the alternative source term, or AST rule.

It allowed plants that were licensed under 10 CFR Part 100, to convert to the AST, 10 AST. An acceptable AST based upon NUREG-1465 was provided in a document known as Reg Guide 1.183.

Based upon Commission direction, this AST used only the releases specified in NUREG-1465 up to the early end vessel release phase.

Commensurate with fuel utilization at the time, the NUREG-1465 source term was limited to core average burn ups of 40-gigawatt days per metric ton uranium, and five weight percent enriched uranium.

The AST rule was written to be flexible though so it did not specify in AST, a specific AST. Alternatives could be proposed and used, as technology and fuel utilization changed over time.

To facilitate this change, Reg Guide 1.183 included a list of significant attributes for future ASTs.

And, go to slide number 11 now, please.

The LOCA source term in Reg Guide 1.183, is specific to whether the reactor is a BWR or a PWR. This table provides a comparison of the BWR source term to the TID-14844 source term.

Significant differences in the release timing, release fractions, and chemical form of the source term exist between 14844, and NUREG-1465.

Of particular note is the chemical form of the halogens. Whereas the TID source term was mostly elemental, the NUREG-1465 source term is mostly aerosols.

You can go to slide number now, 12, please.

In 2009, after roughly nine years of AST reviews, the NRC proposed changes to Reg Guide 1.183, in the draft Reg Guide known as DG-1199.

Multiple changes were proposed, including significant changes to the non-LOCA source term, mostly to address changes in fuel utilization and design.

And, that was specifically because of a footnote in Reg Guide 1.183, known as footnote 11, which limited the burn up and the maximal linear heat generation rate, for utilization in the AST applications.

The NRC has decided not to finalize DG-1199 as Reg Guide 1.183 Revision 1, and is now issuing a replacement known as DG-1389.

You can go to slide 13 now, please.

DR. BLEY: This is Dennis Bley. Can you give us a little background on why that decision was made?

MR. BLUMBERG: Sure. So after the issuance of DT-1199, the staff received a number of public comments, and we spent a significant amount of effort in addressing those comments.

The efforts included an independent review of certain aspects of DG-1199, that were performed by the Sandia National Lab, report San 2086601.

In 2017, the staff received the final responses from SNL, associated with that independent review. But then when we resumed our efforts late in 2020, we had gained a significant number of insights since the issuance of that 1199.

And, including those 2017 Sandia responses that, where they looked at the San 2086601 report independently.

And, then further direction from the Commission in a SECY on back fitting. And, as a also additional research that we had, that had to do with fuel fragmentation, relocation and dispersal.

And, so as a result of that, we decided not to finalize DT-1199, and fold all that information into a new set of regulatory guidance, which we call DT-1389.

And, that's in process right now. And, in the near future, we'll be coming to the ACRS to provide details on that specific document.

DR. BLEY: Thanks, Mark.

MR. BLUMBERG: You're welcome.

So, as I said just then, DG-1389 builds upon the changes proposed in DG-1199, as modified by public comments.

Source term guidance for higher enrichments, which are eight weight percent and burn ups up to 68-gigawatts days per metric ton uranium, peak average burn ups, and near-term accident tolerant fuels, also known as ATFs, are provided in this proposed draft guidance.

These near-term ATFs include chromium coated cladding, and chromium dope fuel, but do not include iron chromium aluminum alloy cladding.

DG-1389 also addresses fuel fragmentation and relocation, and dispersal.

Can you go to slide 14, now, please?

MEMBER REMPE: Before you do that, we've not seen the version, the latest version of 1389. But we, anyone can find a copy of draft Guide 1199.

And, all of the changes you had on slide 13, as well as what I'm seeing here, are LWR based, correct?

MR. BLUMBERG: That's correct.

MEMBER REMPE: And, so are you planning, I mean I know we've done the pilot planned evaluations and you've got a website.

But is staff going to have some guidance on it, and I'm just kind of wondering about if something like what's done for the NPUFs would be, I mean you could give these new design developers some ideas by reflecting on what's done with the NPUFs a bit.

MR. BLUMBERG: So, it was my understanding that the ACRS had provide, been provided a draft copy of 1389. Were you, was that made available to you, Joy?

MEMBER REMPE: I've not seen it. Dave, have you seen it?

CHAIRMAN PETTI: No, I've seen a draft --

(Simultaneous speaking.)

MEMBER REMPE: I know we're having a meeting in March.

CHAIRMAN PETTI: -- of (Audio interference) but I, and I have the draft 1.183 from this past summer when it was earlier on the schedule, and then it got slipped.

I don't recall seeing anything on the fuel fragmentation, and how that could have potentially changed the write up. I don't think we've seen that yet.

But we're going to be, I'm assuming we're all going to, that's all going to get wrapped up in the next meeting on 1.183.

MR. BLUMBERG: That will be a future --

(Simultaneous speaking.)

MEMBER REMPE: Yes, well, that's --

MR. BLUMBERG: Go ahead.

MEMBER REMPE: -- yes, it's coming up in March so we need it soon, is the first question that I had because we do have it scheduled, and you're supposed to get us the documents 30 days in advance.

But it's still all LWR based, right?

MR. BLUMBERG: It is all LWR based, but it provides ideas on how to deal with future designs.

So in other words, it's not giving specific guidance, but it does provide advance guidance for LWR designs that could be potentially used for future designs.

MEMBER REMPE: So, it's a bit of stretch for them to come in and try and figure out how to do this, unless they look at.

Again, I'm thinking about NPUFs and some other interactions we're having, and I'm just kind of wondering if more could be done. But I guess we'll, that's what we're here to discuss today for the non-LWRs, right?

MR. BLUMBERG: That's correct. This presentation is an overview on the LWR source term development. And, then future presentations will go into detail about other source terms.

MEMBER REMPE: Okay, thank you.

MR. BLUMBERG: You're welcome.

CHAIRMAN PETTI: Steve, you have a comment?

DR. SCHULTZ: Mark, this is Steve Schultz. Question, which may be better in the next presentations to this, to the committee.

But you mentioned that some Accident Tolerant Fuel have been incorporated into the draft Guide coming up, and some have not.

Why was that, and what is the level of effort that would be required to expand that?

MR. BLUMBERG: So, Steve, if you don't mind, if possible I'd like to defer those type of detail discussions to the, to the presentation we're going to give on DG-1389.

But in general from a high level, what it boiled down to was lack, lack of data. We don't feel like we've got the data that's necessary to go beyond this 68-gigawatt day for metric ton uranium burn up limit right now.

And, also with respect to some of the

other ATFs, it's the same thing. There's data that needs to be synthesized for other than those near-term designs.

DR. SCHULTZ: Good, thanks for the overview. Look forward to the presentation coming up.

Thank you.

MR. BLUMBERG: Thank you.

Okay, so we are on slide 14. And, did we cover 14 already? Let's see here. Yes, and that's basically what slide 14 was about, too.

So, due to the limitations on data, currently we're unable to go beyond the 68-gigawatt days per metric ton uranium, and beyond the near-term ATF designs.

However, research is under way to accommodate higher burn ups and enrichments, and other ATF designs. And, after that research is completed, our anticipation is that we will come back with a future Reg Guide 1.183 update to include that research.

In addition, DG-1389 includes a method for calculating plant specific non-LOCA release fractions. And, includes a generic non-LOCA release fractions for BWRs and PWRs.

Please go to slide 15 now.

So, to conclude this presentation, I'd like to summarize some of the key points from the presentation.

So, one of the ways the NRC staff and licensees determine what measures, and barriers are needed to protect health and safety of the public, is to perform design basis dose calculations.

A key component of these analysis is the determination of the release source term. Over time, the NRC has developed regulations, source terms, and regulatory guidance to provide licensees and the staff, with efficient methods of performing these dose analyses.

And, then lastly, ongoing efforts by the NRC continue to revise these source terms and methods to address modern fuel utilization, and the use of Accident Tolerant Fuels.

So, this concludes my presentation. Are there any additional questions from the ACRS members, that I haven't answered in the presentation?

MEMBER KIRCHNER: Mark, this is Walt Kirchner.

Just for clarification. So, there's going to be an update to 1.183, as well as a release of DG, let me get the number correct, 1389 in the near future?

MR. BLUMBERG: So, DG-1389 is a draft revision to Reg Guide 1.183, and that's what we'll be presenting to the ACRS for their review in the very near future.

MEMBER KIRCHNER: Okay, they're one and the same then?

MR. BLUMBERG: They are, they are. And, it's my understanding that you should have access to that draft revision. If you do not, then I'll make sure that you get it.

MR. WANG: Hi, this is Weidong Wang, from the ACRS staff. And, we did receive those documents and there was a plan on the November 19 meeting, subcommittee meeting last year. Now it's moved to March 16.

So, the files is there.

CHAIRMAN PETTI: Okay, so it should be, the question in my mind was yes, I read the ones for the November meeting that then didn't happen in November.

It hasn't changed since then?

MR. BLUMBERG: Not substantially, no?

CHAIRMAN PETTI: Could you check on that?

(Audio interference.)

CHAIRMAN PETTI: Okay, thanks.

MEMBER REMPE: So, what are we supposed to review for March 16? The November document --

(Simultaneous speaking.)

CHAIRMAN PETTI: Yes. I think so.

MEMBER REMPE: -- or, and there won't be anything about the fuel fragmentation, and FFRD stuff?

MR. BLUMBERG: Right. The only change that we considered, and we're looking at it right now, was potentially due to the definition of the near-term ATF designs to clarify that. To make sure that it was understood that, that it including the FeCrAl designs.

But we haven't actually done that change yet.

CHAIRMAN PETTI: Okay.

MEMBER REMPE: Okay. And, then --

(Simultaneous speaking.)

CHAIRMAN PETTI: Because that was an issue that I had raised. And, I thought (Audio interference) like when I heard that, I thought I misread it.

But you guys are actually doing a little bit of clarification then, which I think will be good for the committee to hear.

MR. BLUMBERG: Good.

CHAIRMAN PETTI: So.

MEMBER REMPE: And, then --

(Simultaneous speaking.)

CHAIRMAN PETTI: So, yes, so members, I think that old, that document we got that Weidong talked about is, is fairly current and they'll just, there's just a couple things that they'll probably bring up new in the meeting.

MEMBER REMPE: If someone had theoretically, a small modular LWR and they did not want to use the Reg Guide, they wanted to go with a maximum hypothetical accident, does the guidance, this revised guidance, does it help point out what is done? Or needs to be done?

MR. BLUMBERG: Could I call on some help here? Michelle Hart, are you on the line?

MS. HART: I am.

MR. BLUMBERG: Would you like to speak to this?

MS. HART: So, I think there is information in Reg Guide 1.183 right now, that talks about the attributes of an acceptable alternative source term.

And, so you could use that information to help inform what would be necessary for a maximum hypothetical accident.

If you wanted to use the guidance that's out there for non-power reactors, which is more focused on a maximum hypothetical accident as a, as a licensing aspect, you could also use that information. That is NUREG-1537, I think.

But there is nothing specific in Reg Guide 1.183 that would give you a different licensing source term for that kind of assessment, if that's your question.

MEMBER REMPE: That helps. And, I'm just thinking again --

(Simultaneous speaking.)

CHAIRMAN PETTI: Yes, and --

MEMBER REMPE: -- we're supposed to think about things that aren't there yet.

I'm wondering if maybe, is it on the website this NUREG-1537 so that, I mean the website's supposed to give everybody all the different references to consider. Is that out there on that website?

MS. HART: So, we will talk some more about the website later. I think, you know, right now I can't recall if it's there.

But we are intending on keeping that website as up to date as possible, as we figure out new things that need to be on there. Or even old things that we, we don't recognize. It's a list of links. So, we'll talk about that some more later.

But certainly, it is on the public website now as the standard review plan, and guidance for non-power reactors.

MEMBER REMPE: But again, we're trying to help the folks that are coming new, and they may not think about it.

So anyway, it's just an idea that I was thinking about.

MS. HART: Yes, I think this is --

(Simultaneous speaking.)

CHAIRMAN PETTI: So --

MS. HART: -- to event selection kind of discussion.

And, so once you figure out the events, you know, you develop your source terms for that. So it's a related topic, or it's an entangled topic we'll say.

CHAIRMAN PETTI: So Michelle, in terms of 1.183, this guidance, is that the Appendix A as I remember?

MS. HART: So, we're --

(Simultaneous speaking.)

CHAIRMAN PETTI: That talks about the

attributes?

MS. HART: So, the attributes of an acceptable alternative source term are in position 2, which is in the main body of the text. It's not in an Appendix.

CHAIRMAN PETTI: Ahh.

MS. HART: The appendices go over each of the accident types, and the source term that would be acceptable. Or, one source term that would be acceptable to meet the regulations.

CHAIRMAN PETTI: Okay.

So members, I think we're going to hear a lot about LWR stuff, you know, in that meeting here in March. So, I think we should just keep going so we can get through everything.

But I see a hand raised. Elijah?

MR. DICKSON: Hi, Dr. Rempe. This is Elijah with the staff.

We did do some FFRD research in regards to the MHA and the LOCA source term that Mark was discussing.

And, in the DG draft guidance that you'll be seeing here shortly, it does refer to some work that the Office of Research had done for us.

And, so you can look at that document,

that memo and the analysis that Dr. Mike Solay (phonetic) put together for us, that addresses FFRD for this one particular source term.

MEMBER REMPE: So, there's a lot of different draft Guides and things being juggled around here.

Again, to make sure we are, because it wasn't in the files that we were given to review for this meeting, and that's why I'm a little puzzled. I didn't know to go back to November.

But what we're going to be reviewing in March, will have not only the version of the draft Guide that we're supposed to review, but this additional FFR, I mean I know we've discussed and we actually wrote a letter on FFRD research recently.

But it's going to have some additional position or guidance, that will be part of what we're supposed to review, or not?

MR. DICKSON: We'll make sure that you have that information, yes.

MEMBER REMPE: Okay, thank you.

MR. BLUMBERG: And, if you refer to slide 9, that's the memo that Elijah was speaking to, the footnote there at the bottom. That provides the reference.

It's not slide 9 on your presentation. Just a second, that is slide 13 in your presentation.

(Pause.)

MEMBER REMPE: So we'll look at that, as well as what we are given for Reg Guide 1.189's update.

Thank you.

CHAIRMAN PETTI: Yes, so if (Audio interference) that by Dr. Weidong so that we can put it on our SharePoint site, and will all be update.

Thank you.

Dennis, you had your hand up but then maybe you put it down?

DR. BLEY: Right, that's true Dave. I was going to save this for the end, but I'll say it now.

This was a nice presentation. It kind of not only showed this history well, but showed the linkages among so many different things people have to tie together.

It strikes me, and I'm sure the rest of the presentations will go into the newer material in more detail.

Today's transcript and slides might be a really good source for people who have to deal with this issue. Especially some of the new designers.

And, I was wondering if among all your other tasks, the staff is thinking of putting together something like a white paper that sort of replicates today's presentations, to give people a good map if they're going to have to do this for themselves.

CHAIRMAN PETTI: Yes, Dennis, the word that keeps bouncing in my head is consolidation. Because it's in lots of different places, and lots of different piece parts.

And, yes, I worry that the new designers, you know, it's harder to get to that. And, something that would, a road map like you say, could be quite useful.

DR. BLEY: Yes, the consolidation is really the key to, well, I think a key for many people on this, to see all the pieces in one place.

And, we tracked down a lot of this after we went on what we, we knew from our experience. And, I think it would be very helpful.

And, maybe today's meeting slides would serve that same purpose.

CHAIRMAN PETTI: I see John Segala has his hand up. John?

MR. SEGALA: Yes, thank you.

Good comments there. I think that was one of, that was in our thinking as we were putting together this, this new webpage on source term, was trying to provide, you know, members of the public sort of a here's where all the information is in one place.

And, so to the extent that what we do have on the website are our presentations, videos. We have a whole lot of different information up there.

So, if today's presentation, slide material, you know, that's all publicly available, I think that's something we could look at adding to, to that website as well.

But yes, I think we'll have more discussions as we go through the material today.

But thanks for the comment.

DR. BLEY: Yes, thanks, John.

CHAIRMAN PETTI: Yes, to me --

(Simultaneous speaking.)

DR. BLEY: I hadn't seen the, have not yet looked at that part of the website, and that might be the perfect place for all of this. And, I look forward to looking at it.

CHAIRMAN PETTI: Yes, I mean to me, something like a lead me first, right? The first document you should look at, you know, like the years ago when you used to get software, that was the thing you had to do. So that they don't, the staff doesn't get, people don't get overwhelmed, and kind of guides them.

Because I think there'll be a lot of information on the website, so, thanks.

Okay, let's keep moving along; we're doing good.

(Pause.)

MR. ESMAILI: Good morning. So, my name is Hossein Esmaili. I am the Chief of Fuel and Source and Code Development Branch, in the Office of Research.

I'll be talking briefly about NRC analytical tools and past studies, specifically addressing severe accident progression and source term.

Next slide, please.

So, before I get into some of the details, I just want to leave you with some high level messages that will be the theme of my presentation.

First, we have decades of experience in developing our computational tools that we need to predict the source.

These tools are under active development assessment. They are considered state of the practice tools that is used worldwide, by many regulatory bodies, and research organizations.

And, we have a strong relationship with international community to our code sharing programs. This helps us identify knowledge gaps, and participate in experimental research at a fraction of cost to us, to improve our tools.

I will go through some of the examples of the more recent supplications of the code, in terms of, especially the work that we have done following the Fukushima accident. And, how these tools have helped us in resolving regulatory issues and decisionmaking.

I note that Dr. Petti and his introductory remarks said that there was lot of technical material in this, but my intent is not go into the details of these technical issues, since this have been discussed and presented to ACRS before.

But to paint you an overview of the complexities of the issues, and how we go about resolving them using our best state of practice computational tools.

Next slide, please.

So, first I will give an overview of the code MELCOR. This is our severe accident code. I will talk briefly about international collaboration, and this is important in how we understand severe accidents, and how we improve our code MELCOR.

And, some of the examples as I mentioned in terms of regulatory decisionmaking. This is design certifications for new advanced reactors, state of the art reactor consequence. Now, this is SOARCA, and how it help us you know, move forward.

And, some of the post-Fukushima activities. These are the activities that they have been going on for the past 10 years or so.

And, finally, I will, Jason will talk about the applications to new advanced reactors. And, this is the work that we have done, the scale MELCOR demonstration calculations.

Next slide, please.

So, well, next slide.

So, the importance of regulatory source term is well established. It finds, I think it's way, into many of our regulations for safety and environmental reviews. Mark was talking about this for LWRs.

This slide shows the basics of the source term development process. This is the process that was followed in NUREG-1465,Reg Guide 1.183.

And, this is exactly the same process that we are following for advanced nuclear technologies, including ATF and high burn off fuel. And, non-LWRs.

So, the first step is the identification of important phenomena, or you know, what will be establish at PERT, phenomena identification, and ranking data.

For example, we recently completed a PERT for ATF high burn up fuel, with the aim of understanding knowledge gaps in MELCOR. This is what you see at the, in NUREG 72.83.

Of course, we need experimental basis for some of these models in the code. So, we need specific data for fission product diffusivity. And, we model different phenomena.

The next step is identification of risk significant accident scenarios. And, this is typically informed by PRAs. NUREG-1150's input to the NUREG-1465, and what accident scenarios we should, we should model.

So, what is important in terms of release characteristics is duration of the release, release fraction, and radionuclide species.

So, if you are interested in the in containment source term, then we can synthesize the results. This is what we see on the right hand side of the slides.

Otherwise, the same process and tools can provide environment release for different accident scenarios. This is typically the NPRAs, or design certifications, or review of chapter 19.

Next slide, please.

So, this slide shows the relationship between the phenomena we are trying to understand and model, and their experimental basis. This information is then incorporated into MELCOR, by adding and improving modeling capabilities.

SCALE is also an important input. It defines decay, hidden ratings like inventories for any MELCOR accident progression analysis.

The outputs from the MELCOR is then input into MAACS for offsite consequence.

And, I just want to point out a few items here. So, we rely on experiments to understand the phenomena, and validate our models. This is what you see in the dark blue boxes.

We have used the codes for regulatory applications. I will go, as I said, I will go through some of the examples in the green boxes.

The staff is familiar with the code

extension to new technologies, including non-LWRs required minimum training, and we have great flexibility for uncertainty analysis.

This is becoming more of a standard practice, and this is what we followed in the latest SOARCA analysis.

What you see in the red boxes are our focus, non-LWRs and how we're improving our code for non-LWRs and high burn up on Accident Tolerant Fuel.

Next slide, please.

So, I'm just going to give you a brief overview on history of MELCOR.

Next slide, please.

So, before MELCOR we had separate effect codes. These where we deemed the source term code package. They were run independently. Results were then manually transferred between the codes.

But this led to a number of challenges for transferring data, ensuring consistency in data and properties, and capturing the coupling of physics.

For light water reactors over the years, various stand alone codes were integrated into MELCOR. The project actually began in 1982.

We had the first release of MELCOR, this is MELCOR 1.80. This was a domestic release in 1986, and then followed by an international release in 1989.

In 1991, LANL did a comprehensive peer review was conducted, and all the recommendations are now dealt with. And, code development and maintenance is ongoing, and we typically have annual release of the code as you can see on the slide.

At the bottom, I'm just showing you a timeline of the MELCOR development. What models we have put into the code over the past 20 years or so.

Some of the milestones include introduction of more mechanistic models. For example, formation of a molten pool in the core. This was introduced in MELCOR 1.86.

And, spent fuel pool models. We started looking at spent fuel pools more closely after the 9/11.

And, new models capturing physics for HDGR started more than 10 years ago in support of NGMP.

We also started putting in models for other non-LWR designs, and we were able to conduct public workshops on non-LWR application this past summer, and Jason will talk about those in more details.

Okay, next slide, please.

All right, so MELCOR is an integral system level code. It models all phases of a severe accident from accident initiation to core heat up, and degradation leading to vessel and containment failure; tracks fission products through the RCS, and containment to the environment.

So, it does everything.

Back in the '90s, NRC was developing other codes, such as containment SCDAP. However, these other codes only partially model some of these phases.

For example, there are no models for RCS in contained. And, SCDAP it doesn't model containment phenomenon associated with fission product deposition and removal.

Research went through a code consolidation phase, and decided to focus on MELCOR. So as you can see on the upper right hand side, all these phenomena are well captured by MELCOR.

So, I want to make a very important point here. The question is: what are the requirements for the level of details in a severe accident code?

So, when it comes to severe accident, uncertainties in the accident progression and available experimental data for model validation, does not support more detailed model, modeling approach.

So, we are able to capture some of this

phenomena through simpler models.

The other point is MELCOR is flexible. It allows the user to build the entire model of the plant from basic building blocks, such as control volumes, flow path, and structures.

That's why it's easily adoptable to new reactor designs, and implication to spent fuel pools as I showed on the right hand side here.

Am I live? Can everybody hear me?

CHAIRMAN PETTI: Yes, we can.

MR. ESMAILI: I'm not hearing any questions.

Okay. Next slide, please.

So, again, I have shown a lot of these slides before. So, you're very familiar with it.

In this picture, I'm trying to capture the importance in accident progression that can affect the source term. The aim here is to identify similarities and differences within various reactor technologies. So, I also provided a comparison to LWR, and see if we can leverage what we already know, and how we can use our existing tools.

I would break it down to three main levels. So, going from the top, when it comes to the containment, there are phenomenon processes that are common. This has to do with high aerosols and vapors evolve and are removed. For example, aerosol-shaped factors may be different under dry and wet conditions in the containment.

But when it comes to the primary system, there are, obviously, differences. This is high radionuclides are released from the fuel, but the underlying physics and data is diffusivity to the fuel, and other components can be similar. And we have the coding infrastructure to deal with that. In terms of aerosol dynamics, we also believe similar processes are occurring.

There are some ex-vessel phenomena that are different. For example, we have four concrete interactions in the case of LWR and sodium fire in case of SFRs that are unique, that have to be treated separately.

Next slide, please.

So, code verification and validation is an important element of the software quality assurance. This is the program that we have at Sandia. MELCOR documentation is extensive. There are separate volumes for users' guides, reference or theory manual, and code assessment.

Validation is targeted to a relevant

physics model, and many of the models that are represented in MELCOR are agnostic to the particular reactor technologies that I showed you before for containment phenomena.

Here are a few examples from non-LWR applications. AB1 was the assessment of the codes with AB1 test. This was a test conducted at the Containment System Test Facility in Hanford, Washington. This was a sodium-cooled fire under dry conditions, providing data on aerosol behavior. And as you can see, MELCOR does a reasonable job of predicting the evolution of suspended sodium aerosols over a 50-hour timeframe.

We also participated in an IAEA benchmarking looking into releases from the TRISO particles. This is what you see on the upper righthand side of the slide. This is documented in TECDOC-1674.

As we move on, we are going to look at the other validations; for example, MSRE in sodium reactors and HTGRs, as we move on.

Next slide, please.

So, here I'm showing you the evolution of core and RCS nodalizations and the modeling details that we needed over the years.

The first input, source input package. This was the model that we had back in the '80s. It has a simple core with one hydrodynamic cell, but multiple core cells in both radial and natural directions.

So, when we started in MELCOR, as I mentioned before, in the early '90s, we improved the hydrodynamic modeling for the RCS and the core. By the mid-90s, we had more details on the RCS to model hot leg natural circulation. It became important, you know, when we started looking at steam generator tube rupture. By the end of the '90s, we had more details on the RPBN core to model in-vessel natural circulation, and RCS and in-vessel natural circulation can impact timing of the core damage, hydrogen production, fission product deposition inside the RCS, and potential for RCS piping failure and rupture.

So, as we moved through the years, we have updated our practices and our code nodalizations, et cetera. And this is --

CHAIRMAN PETTI: Hossein?

MR. ESMAILI: Yes, sir.

CHAIRMAN PETTI: Just a question. I assume -- I think it's true -- that, as the modeling sophistication has increased, those source terms have, I guess, reduced. So, the estimates are more accurate, the uncertainty less. Is that a fair assessment generally?

MR. ESMAILI: Well, I don't say it's reduced the source term, because I cannot say that because it really depends on a lot of factors. But what I can tell you is that the accident progression affects the source term. So, the more accurate that you model the accident progression -- you know, like the reasons, for example, we have multiple hydrodynamic core cells inside the core was to get a better understanding of how you have oxidation; how that fuel heats up, and how it moves once the core relocation process occurs.

So, yes, we have better estimates of the source term.

CHAIRMAN PETTI: Yes.

MR. ESMAILI: Yes. Does that answer your question?

CHAIRMAN PETTI: Yes. I'm just thinking about now you go to these advanced systems, where, hopefully, the physics is simpler; that, you know, the sledgehammer when you may only need to put the tack in. It may be more than is necessarily needed. But MELCOR is scalable in the sense that you could do something simple, if that's all you felt you needed, right? I mean, it's got that flexibility?

MR. ESMAILI: That is true. That is true. And, you know, there's a limit to what we can do. So, when I'm showing this core nodalization, you know, we didn't study it to say that, if you had more models, if you had more nodes, does it improve our modeling. And it turned out that, no, at a certain point, the number of rings, the number of actual levels is sufficient to capture. And this has to do with our inability to know everything about the core relocation process.

You are right in terms of some of these non-LWRs, because they don't go through this cliff-edge effect, right? I mean, you're not melting. You're not draining this molten. And so, it could be simpler, yes.

CHAIRMAN PETTI: Thanks.

MR. ESMAILI: Yes, thank you.

So, next slide, please.

So, for the past few years, we have conducted research to modernize MELCOR code with the key goal of enhancing the efficiency of the development and maintenance of this large, complex code base. Oh, yes, the code is huge. You know, in the earlier presentation, I always refer to it like it has almost 2 million lines of code. It's probably increasing by the day.

So, frankly, much of the code architecture reflects programming practices that are several decades old. This does not allow MELCOR to incorporate and benefit from the significant evolution in modern programming languages, operating systems, and compilers.

And, in addition, as I mentioned before, the code involved integration of separate other codes. So, these were all integrated into MELCOR, and sometimes using their own solvers. You know, like the other codes that I was mentioning--Maros, VANESSA, et cetera--they had their own solvers because they were standalone codes.

But, then, they were integrated into MELCOR in what they call "code packages." The communication between these different packages is also complex, as you can see on the upper righthand side. You know, the hydrodynamics has to communicate with the core package. The core package has to communicate with the radionuclide package, et cetera. So, that communication is pretty sophisticated.

So, the idea in modernization, we still

wanted to have the code as a state of the practice, too. So, the basic idea is to separate the physics from the numerics. You know, numerics is evolving. We don't have to touch all aspects of the code.

So, this is done in stages. Last year, we completed the hydrodynamic package. FY22-23, our focus is on the core modeling. You know, this also includes ex-vessel and in-vessel melt progression. And in FY24, we plan to finish with the fission product release and transport modeling, and hopefully, we would have a modernized code by early FY25.

All right. Next slide, please.

So, this is the slide that I used when I was briefing the ACRS on our, quote, "readiness plan" for non-LWR applications. I'm adding it here. As I mentioned, you know, most of these slides you have seen before, but I'm just putting everything together.

So, we have modified the code. We have developed reference models. We have run calculations and conducted workshops. You're going to hear a little bit about that later.

So, just like light water reactors, the model that we have developed requires some input data, and the data can come from all sources. It can come from codes, experiments, et cetera. So, I have listed some of the data needs in terms of what we need to put in the code. But I want to make sure that, as far as the code infrastructure is concerned, it can accept -- you know, as data becomes available, we can put it into the code. We do not need to go and change basically the models. We can just change the input parameters.

So, as an example of the data needs, the models that are implemented -- but they need data -- are fission product diffusivity. Here we have an example of diffusivity through a TRISO particle. We are modeling it; it is simple, one-dimensional diffusion equations. But I need to know what that diffusivity is in terms of calculating the buildup of the fission products through the different layers.

SCALE will provide us with the data that we need in terms of generation of the radionuclides into TRISO particles.

MEMBER REMPE: Hossein, this is Joy.

MR. ESMAILI: Yes, ma'am.

MEMBER REMPE: It looks like you wanted a question. So, I'll ask this one now.

(Laughter.)

I know when we talked about this a while back with the code readiness reviews documents, I thought we brought up the point that was you do the pilot plant applications. You'll also gain some insights about which data are very important and have more risk impact. And you can wait and answer this later in the pilot plant or reference plant discussions.

But I'm curious if you've been able to make much progress in that area, so that it might give some insights to the design developers on, yes, we've got to have something, but plus or minus a factor of two doesn't make much impact on the results. And so, we can use some sort of bounding value for some of the parameter needs.

Do you understand where I'm going?

MR. ESMAILI: Yes, yes. So, Joy, I don't know whether you have seen the slides. I think in a little bit later slides, I'm talking about some of this uncertainty. And one of the uncertainty analyses that we have done for HTGR, maybe when I get to that, maybe I can answer your question better. Is that --

MEMBER REMPE: That's fine, but it's not just that you've identified it. I'm also curious if you're communicating that back. Is there a document on the website that says that -- you know, how do you inform the folks coming in? In this case, you're mentioning the gas reactors. Do the folks that are engaging in the pre-application discussions have a good feel for where the staff is with the progress they've made on the reference plant evaluations?

MR. ESMAILI: So, in that, you know we have conducted public workshops, right?

MEMBER REMPE: Uh-hum, right.

MR. ESMAILI: And we shared information of what we know, you know, what is important. But these are like reference plant models. These are like publicly available information that we can gain.

But, as a matter of fact, we are starting to release some of these input models to anybody who wants them. This could be the industry or anybody. And they can follow the same procedure, right, that we have done, to find out what it is that we found important.

And so, you know --

MEMBER REMPE: You don't have to answer it now --

MR. ESMAILI: Okay.

MEMBER REMPE: -- but I'm just thinking about readiness and guidance, and some of the things like Dave and Dennis brought up earlier, things that the staff has learned from these evaluations that not only make them ready for incoming applications, but guidance, so that they focus.

You know, if you say that parameters X, Y, and Z are very important -- again, you did a reference plant evaluation -- their design may be different, but they'll know to focus and say, okay, that may be important for what you did, but our design is so different, we don't have to do it. And it might facilitate and make future reviews more efficient. It's just an idea I had. Anyway, just a thought.

MR. ESMAILI: I think that is very important, Joy. I really appreciate your asking this question. Now I'm just trying to put this into -- you know, because we are looking at different accident scenarios, you know.

But one thing I can say, now that you brought it up, is what I'm showing here in terms of TRISO. For example, we are talking about fission product diffusion coefficients. But the other thing is fuel failure, right? That's another important -- so, our experience has shown that we can go ahead and look at these fission product -- and I know that they are doing this under the AGR program with DOE; that they are trying to use their tools to do experiments, et cetera; find out what this fission product diffusivity is, right? So, this is a large effort to gain that.

But it's more important to know what the fuel failure is, right? Because that silicon carbide layer is a good fission product retention, right, layer? So, if you want to know more about this, we would like to know more about the fuel failures. That dominates the source term, as opposed to fission product diffusion coefficients, right?

So, these are the insights that we gain from running calculations, doing uncertainty analyses, doing sensitivity. And you're absolutely right, as we go about doing some of these things, we can communicate that, so people or whoever is using our codes understand.

Does that --

MEMBER REMPE: That helps. It's just a thought to consider, and maybe you've already done it, but something to think about.

Thanks. Go ahead.

MEMBER BIER: Hi. I have another question, if I can interrupt at this point. This is Vicki Bier.

So, I should preface this; I'm not a physical scientist at all. I come from the PRA world.

And in the PRA world, there is a certain level of detail beyond which the results really become meaningless. So, like if we tried to expand the level of detail in a PRA to the level of resistors and capacitors in a plant, it would be impossible to create that level of detail of model with any accuracy.

And so, as you talk about building more features into MELCOR and more level of detail of what can be modeled, what do you do about validation, and are you creating situations where you are, basically, kind of giving users the ability to hang themselves by giving them so many parameters that may be difficult to estimate or so many models that may be difficult to build accurately?

MR. ESMAILI: So, I think, as I mentioned before, you know, we are resisting. We are resisting, and I think of some of the areas mentioned, that we are resisting going into a lot of details. We don't need to do that because, again, MELCOR is a system-level code. And, you know, it has been -- it's used consistently for PRA applications. As a matter of fact, we are using it in a Level 3 PRA that we are conducting at NRC.

It is -- sorry?

MEMBER BIER: No, go ahead. Sorry.

MR. ESMAILI: Okay. So, we are very careful in terms of what needs or does not need to be. I will get into that, but there are lots of people all over the world that they use this for their PRA applications. And some people, they go into more details than is necessary. But, through our guidance, we conduct workshops; we meet with them. We are informing them of the level of detail that is needed; the level that you don't really need to get into. And so, we have interaction in terms of what is needed and what is not needed.

And we don't ever go into that level of details that you are talking about; that modeling every nut and bolt in the plant. We just model whatever is necessary to gain an understanding of how the accident progresses.

MEMBER BIER: Great. Thanks.

MR. ESMAILI: Okay. Thank you.

Next slide, please, "International Collaboration."

So, I'm going to go a little bit -- I don't know how much time I have.

But, next slide, please.

CHAIRMAN PETTI: Yes, Hossein, we would

probably like to take a break at the top of the hour.

MR. ESMAILI: Okay.

CHAIRMAN PETTI: So, just to give you a sense.

MR. ESMAILI: All right. So, let's see if I can finish by the top of the hour.

So, in addition to the domestic use, the code is provided by the NRC to international organizations through bilateral agreements. This is under the Cooperative Severe Accident Research Program, or CSARP.

This is an international program. Severe accident, you know, knowledge, code research and code development. It provides access to experimental data for code development, modification, assessment.

It's an NRC coordinated program with participation from a lot of countries. Actually, I think that the program started back in the '80s. We have limited experimental programs sponsored by the NRC. The current thrust is on development, assessment, and modification of our tool MELCOR, and we host a meeting once a year, usually in June, to exchange progress in severe accident research and to report code development and assessment status.

And so, what you can see in the map here is that we have approximately about 30 countries that are participating in this program -- you know, from North America to Europe to Asia.

There are two user group meetings, one for Asian MELCOR users. It's called Asian MELCOR User Group meeting. This typically meets in the fall, and we had just had one in November. It was hosted in Singapore, and it was still virtual, but hosted by them.

And a European MELCOR User Group meeting that meets in the spring. This is sponsored by European countries. So, every year it's different. This year it's going to be Poland who is hosting the meeting. Again, it's going to be virtual and it's going to be sometime in April.

These meetings allow more interaction between the code developers and the code users, so they can get more access to what's going on. So, since we are thinking of guidance, when it is time to actually apply the code -- and also, the MELCOR workshop -- this is where we tell people, you know, what it is that they can focus on; what is the important modeling practices; what's the best modeling practices, et cetera.

There's also a large user base worldwide.

I listed nearly 1,000. So, as you can see from the map, somewhere in the world someone's light is on and probably running one of our codes. So, it has world usage.

Next slide, please.

So, this slide shows some of the relevant international severe accident projects that the NRC uses to develop and validate the MELCOR code. I'm listing some of them.

The PHEBUS, of course, is a very important program involving the knowledge of the tests of fission product release and transport from irradiated reactor vessel, reactor fuel. This was organized by IRSN at the Cadarache facility in France.

It consisted of five tests involving releases from irradiated fuel and steam transported to a model RCS, including a steam generator tube and behavior in the model containment. And we use the PHEBUS experiments to validate the MELCOR in NUREG-1465. So, this was one of the first large-scale integral experiments that we were participating in.

And from this experimental program, the international severe accident community did a series of separate effects experimental programs to study phenomena for which the code did not capture the behavior. And we are constantly monitoring, and we are involved in some of these studies.

These are the Behavior of Iodine Project, for example; Source Term Evaluation and Mitigation, STEM.

And after Fukushima, there were additional programs initiated to better understand the experimental progression with severe accident codes because, as I said in the previous slide, the international community uses our code. So, there's a larger user base of MELCOR, and once you participate, it improves our modeling and predictive capabilities.

And recently, we are conducting experiments to study underwater melts. This is especially for composition with high metal content. This is representative of those in Fukushima. This is the ROSAU program.

Finally, we are looking into the potential sources of delayed radionuclide releases. This is, you know, where we are talking about revaporization -- this was observed at Fukushima -- focusing on the revaporization, the RCS and formation of organic iodides in the containment. This is the ESTER program. They are also participating in an exercise. It's very similar to what we have done under SOARCA.

They are looking at uncertainties in severe accidents.

Next slide, please.

So, last year, we formed a panel of experts. They collaborated to create a Phenomena Identification and Ranking Table. This was to address significant phenomenological issues impacting core degradation and radiological releases for various ATF designs. We also looked at the impact of burnup in enrichment and compared with conventional fuel.

The aim of the PIRT was to help the NRC to focus attention on how these concepts change our existing understanding and provide information on how we can use this to improve MELCOR.

The final meeting was held, actually, last April, and we published two NUREG documents that you can see on the righthand side of this slide. One was a literature review. This was literature review of what we do know about ATF and high burnup. And the other one was PIRT itself.

So, the PIRT really consisted of 10 actually internationally recognized experts. So, it's a good document. I highly recommend reading it, and especially the fact that we are looking at these new technologies and comparing it to our conventional fuel.

We also worked on calculating the source term for high burnup fuel, and we are planning a peer review for later this year.

We are also participating in the QUENCH-ATF program. This is using an experimental facility at the Karlsruhe Institute of Technology, KIT. This is a joint international program through the OECD/NEA. It involves large-scale bunker experiments for design basis and beyond design basis conditions.

So, the data that we are getting from these experiments, again, will be used to develop and validate models from MELCOR and FAST. FAST is also our fuel performance code.

The project is supported domestically by the NRC, EPRI, and Westinghouse, and internationally by 15 organizations from seven countries.

The focus of the first phase would be on chromium-coated zirconium alloys. So, I just learned that the samples have been shipped from Westinghouse to KIT this past January, just about a month ago, and we are hoping to have the first test as soon as April of this year.

Next slide, please.

So, NRC joined a cooperative research

program -- this was hosted by IRSN in France -- to investigate loss of coolant or cooling accident in spent fuel pools. This is called the DENOPI project. It's composed of experiments, modeling, validation of computer codes, even CFD codes. These are designed to extend the knowledge about various phases of a loss of cooling accident in the spent fuel pool.

The project provides experimental data to validate spray -- I think it's cooling spent fuel, cooling bundles, and cladding oxidation under a mixture of steam and air environments. Specifically, we are interested in rate of spray droplets penetration into the fuel assembly and air and steam oxidation of the cooling cladding to address uncertainty with our current predictive capabilities.

We met with IRSN. We looked at some of the results of the experiment, and we are planning to have follow-on to better understand the experimental conditions and how we can improve our code.

And this is in line with what we state in SECY-16-0100, following the "National Academy of Sciences Study of the Lessons Learned from the Fukushima Nuclear Accident" for plant users. Also, it is an effort to enhance MELCOR capabilities. So, we are on target with that.

Next slide, please.

So, I'm going to go look at some of the MELCOR applications.

Next slide, please.

So, MELCOR is used extensively in the design certification of new reactors. I'm showing you some pictures of what we have used in the past and what are the newer designs. It's used for both severe accident response -- you know, this is mostly when you're talking about the LWRs, Chapter 19, and source term, and the containment response to a design basis accident. This is what we cover usually under Chapter 6.

The application of MELCOR to these new reactor designs requires some code enhancement and validations because of the special design features. For example -- and this goes back to the '90s -- for example, for AP-1000, we started using a fuel tracking model for the containment shell, because the containment is being cooled on the outside.

And as far as severe accident mitigation is concerned, there are design differences between some of these designs. For example, the EPR and APWR. EPR, what you see here, has a special core catcher and spreading compartment, and the containment is equipped with passive autocatalytic recombiners for hydrogen control. And the APWR, it's more active systems. The debris shifts through a combination of cavity flooding and enhanced area for debris spreading, and hydrogen control relies on the igniter. So, MELCOR provides us a tool that we can look at these different systems.

Next slide, please.

This is a good time if you wanted to take a break, because the four slides are SOARCA-related. Or do you want me to keep going?

CHAIRMAN PETTI: No, we could probably take a break now. So, thank you.

Let's pause here and be back at 15 after the hour.

Thank you all.

(Whereupon, at 10:56 a.m., the foregoing matter went off the record and went back on the record at 11:15 a.m.)

CHAIRMAN PETTI: Okay, Hossein, I have 15 after. So, let's keep moving. Thanks.

MR. ESMAILI: Okay. Sorry. Dave, do you want me to go a little bit faster in the interest of time? I think I'm a little bit behind, but --

MR. SNODDERLY: This is Mike Snodderly.

I think you're like four slides behind.

I think the goal would be to start the next presentation at noon. But cover the material you need to.

MR. ESMAILI: Okay. I'll go a little bit faster because Jason still has to talk.

All right. So, back in 2006, we started the State-of-the-Art Reactor Consequence Analysis. This is the SOARCA project. We looked at potential consequences of an unlikely severe accident in Surry and Peach Bottom that involved significant quantities of radioactive material, release of radioactive material into the environment.

These two plants are a BWR and a PWR and have been analyzed before in NUREG-1150 and WASH-1400 before that. And following completion of the original SOARCA in 2012, we documented the results, and then, we took a systematic look at a potential source of uncertainty.

We always acknowledged there are uncertainties in severe accidents, whether in modeling or boundary conditions, and while sensitive analysis is helpful, a formal uncertainty analysis, we thought, could shed more light on the expected behavior.

And so, a few years ago, we started on performing the UA analysis for selected accident scenarios in Surry, Beach Bottom, and also, for Sequoyah.

For the Sequoyah -- this is the picture that you see on the righthand side -- we focused more on the containment performance because of potential for hydrogen combustion and early containment failure. And the picture also shows the details of containment, especially the ice condenser region itself.

Next slide, please.

So, these are typical results from the SOARCA back in 2012, in terms of source term and release to the environment, as well as containment performance with the ice condenser. The 2000 results show the range of source term for a number of accident scenarios. The UA analysis showed variation in the source term, given uncertainties in the MELCOR modeling frame, as well as boundary conditions, such as safety valve failure for a short-term station blackout in Sequoyah.

So, that's what we see in the four steps. There are a few early releases of the iodine due to early failure of the containment, but the majority of the cases resulted in late failure of the containment. So, these are the typical results that you are seeing right now in the uncertainty analysis and detailed modeling with SOARCA.

Next slide, please.

So, when we were doing the Sequoyah, we started looking more carefully at the UA results and, in particular, trying to find out if we could better understand the response of the system, do a little bit of data mining, and trying to see if we can cluster the points that are important; you know, trying to find out what are the important figures of merit.

So, in this case, we were looking at hydrogen generation from the time that the hydrogen started generating after the first deflagration. This is important because it characterizes the accident progression, since only the first hydrogen burn determines early versus late containment failure.

And so, on the left, you notice the clustering of the points. So, the yellow are the beginning of cycle for a short-term station blackout, and the red are the long-term station blackouts. So, we were able to cluster these points and superimpose them on top of each other, even though the scenarios were completely different. And this was because, at the time that they started the hydrogen generation, these realizations had similar decay heat and they showed similar behavior.

The differences between the middle of cycle and end of cycles are less pronounced. So, the response in terms of hydrogen generation is also similar. So, the point is that, when you look at this clustering, you see that the response is -- you know, the uncertainties are very established. You know, the points are not all over the place and you can find that clustering of the points.

During that same Sequoyah, there were also questions of how we modeled the pressurizing safety valve and what impact it has on the results. So, this is what you see on the right figure. And following a briefing of the ACRS on the draft, we revised the model. And so, then, our new model and most recent model showed a probability of 80 percent that the safety valve fails to close. In the revised UA, that probably was reduced to about 11 percent.

But both UAs exhibit the same generally characteristics. You know, these are the blue points and the orange points. So, what you see, even though we changed the probabilities, what it did was that it redistributed the points. So, more of the points that are orange now moved into the blue.

And this is understandable because the hydrogen that's produced in-vessel is generally more in the time to hot leg failure and the hydrogen burns longer. Then, the pressurizer safety valve fails to close. And we need a longer time to allow more efficient venting of the hydrogen to the containment.

Next slide.

So, when we are looking at the Fukushima, we are looking, also, at the clustering of the points. So, we have a true MELCOR that gives us something. I'm not showing initial boundary conditions and event scenario.

And some of the questions we can answer are: how well do these different accident management strategies reduce the potential or the magnitude of the release to the environment? What are the critical events in an accident? What are the knowledge gaps in understanding plants' response and various phenomena that have the biggest impact on the accident progression?

So, we did this clustering on the righthand side. This is an example of the calculation in the certainty analysis that we did for the AGR that Jason is going to talk in a little bit, and we'll look at some of the, you know, the immersivity of the RCCS, for example; the graphite conductivity; heat transfer from the reactor cavity cooling system, and also, the blockage; you know, how well we get air cooling of the cavity. And so, we can understand how these parameters affect the results.

Next slide, please.

So, we also used MELCOR; it played a critical role in forensic evaluation of the three core melt events at Fukushima Daiichi, the events at Fukushima. So, they showed significant uncertainties on our understanding of the boundary conditions.

So, what you see here is, with a code like MELCOR, we were able to perform forensic evaluation of the uncertainty boundary conditions and compare available measurements and code estimations to better understand what possible unmeasured and unobserved plant states there were. So, it's very useful in terms of reconstructing the accident that happened at Fukushima. I guess this is the takeaway.

Next slide, please.

So, we started looking at the spent fuel pool modeling since 9/11, and MELCOR flexibility allowed a relatively straightforward application to spent fuel pool analysis. Over the years, we introduced different components, like, for example, racks and developed a general standard radiation model. And so, this would allow us to better model

MELCOR for accidents in the spent fuel pools.

And like you see, the inserts are some of the NUREG/CRs in terms of zirc fire experiments that we conducted, an international PIRT and status report on the state of modeling for spent fuel pool that I added on the slides.

Next slide, please.

So, this is an example of the experiment that was done for a PWR assembly. This was an OECD/NEA experiment. This was 12 international partners conducted the experiments.

And so, what you can see is that you can see the fuel assemblies. This is the hot channel before the start of the experiments. The hot channel is the first, and it's surrounded by four cooler assemblies. What you can see is that, as it goes through the heatup, the central assembly, you know, it heats up and propagates to the other assemblies, and finally, you can see the state of the assemblies.

So, it is also a highly sensitive. So, we tried to model this with MELCOR. Another thing, I think, in general, the MELCOR is capable of predicting the conditions of the experiments. The results were highly sensitive to both oxidation kinetics and transition to breakaway. But, in general, we showed good agreement as far as peak cladding temperature, ignition time, and ignition propagation is concerned.

Next slide, please.

So, we did a major study in NUREG-2161 to look at spent fuel pool modeling. Our focus was on a BWR with a Mark 1 containment. We studied the impact of spent fuel configuration, high-density versus low-density loading, and we did a very detailed analysis.

You can see some of the results here. This is an example of a high-density pool. This is actually at Peach Bottom and it shows where the hottest assemblies from the last two offloads are in a 1x4 configuration and constantly surrounded by colder assemblies.

So, we looked at different accident scenarios, a small leak and a moderate leak. When you are like 13 days, the hotter assemblies are hot enough that they can cause serious damage. And as you can see, it heats up and it goes through a zirconium fire, and then, actually, there is fuel relocation.

For a moderate leak, this was a case where the break was at the bottom of the pool. So, there was some mitigation associated and natural circulation of air through the fuel assemblies that kept the fuel temperatures relatively lower.

Next slide, please.

So, in terms of the magnitude of the release, it depends on many factors: rate of draindown, time after out of load, and mode of mitigation. So, you can see this figure shows the environmental release of cesium-137 during that operating cycle. The operating cycle usually for a BWR is about 730 days. And so, we looked at different snapshots during its operating and the range of releases that we can find.

In general, they have low-density releases. You know, these are the blue boxes. This is where two-thirds of the assemblies have been removed from the pool. It shows that two orders of magnitude are lower compared to the high-density case, where we had like the fuel completely with the assemblies.

What we also found out is that, when we were doing the study, that Peach Bottom is actually not doing 1x4; it's doing 1x8. So, it's one hot assembly surrounded by eight colder assemblies. And this 1x8, it actually was a sensitivity because they're not required to do so, but it was very effective in dissipating the heat through the core assembly. So, this is what we can see in the green boxes here.

But the figure also shows that, during the first week, you know, during the defueling process, the hot assemblies are still pretty hot. So, even though we started doing spray cooling of the assemblies, there was not sufficient water to keep it cool. So, during the first week or so, we still have some releases because we were uniformly spraying throughout at the rate of 200 gpm over the whole pool.

Next slide, please.

So, during the Fukushima event, it became apparent that reliable vent operation was needed to provide containment integrity. Then, the Commission directed the staff to modify an earlier order on reliable hardened vent to make it be a function of a severe accident. So, we looked at these things carefully under NUREG-2206, "Containment Protection and Release Reduction." We used the model that we developed under SOARCA for Peach Bottom, and we had a run matrix of about 50 runs.

We looked at different conditions: the boundary conditions; availability of DC power; how operators control reactor pressure; operation of RCSI from what we learned from the Fukushima accident, and the mitigation during post-core damage. You know, water injection, is it into the driver or the RPV? And how do we vent the containment?

So, sometimes you hear the words "severe accident management," "severe accident water addition," "severe accident water management." This is how we do water addition to reduce the releases after lower head failure.

Next slide, please.

So, in terms of the releases to the environment, we did not explicitly model an external filter, but the effect of filters was considered in the consequence analysis. So here, in the top figure, you see release to the environment. The blue lines are cases without water injection. The red lines assume injections at lower head failure. So, it's clear that, you know, injection is arresting the further releases from the fuel.

But what is important also is what you see on the bottom. So, on the bottom, what you see is that, during the release, most of the releases that are occurring, the particles that are being released, it is in submicrons. Like 80 percent at least are in the .5-micron size. Because the release has already gone through the suppression pool, so it becomes more difficult to further scrub.

So, this provided some insights into NUREG-2206 and external filtration rulemaking.

Next slide, please.

So, in terms of the magnitude of the release, these are the range of releases that you find, depending on whether you do RPV injections, driver injections; how you do water exit management or water addition, and if you cycle the vent or not.

And as you can see, there's not much different between various post-core damage strategies, whether it's RPV or driver injections, and how you manage the water.

One thing is that the highest releases occur for a main steam line break, but this is an unlikely event because they do depressurize the vessel. So, we actually had to work hard; we actually had to induce a steam line break, just to see what the releases looked like.

Okay. Next slide, please.

So, I just want to close by saying that we have decades of experimental and analytical research in severe accident progression and source term.

We know how to develop codes. We understand a lot about how fission products are released and move about and go into the environment, go into the containment.

We have a validated state-of-practice MELCOR code. This is ready for application to a wide variety of nuclear technologies, including advanced designs.

And it has been an essential tool for resolving safety issues and informing regulatory decisionmaking.

Thank you.

MR. SCHAPEROW: Okay. I would like to talk to you about MELCOR application in new reactors.

Before that, though, I would like to mention that, in case you hadn't noticed, we started MELCOR in 1982. So, this is the 40th anniversary of MELCOR, and MELCOR, I believe, has come a long way.

I was there for some of the earlier analyses. The very first MELCOR analysis back in the '90s for steam generator tube rupture, severe accident-induced tube rupture, where we didn't have countercurrent flow in the hot leg, and somebody asked, well, how are the things going to get into the steam generator? How is the steam generator going to heat up? And fission products? So, anyway, this really has come a long way, I believe.

So, next slide, please.

For new reactors, we do independent analysis using MELCOR. That's part of our review. We have guidance for that. The guidance is given in the Standard Review Plan.

The guidance includes: the staff will do an independent assessment of plant response and source term. The staff will do this for select scenarios from the PRA. And then, the staff will sit down with the applicant and discuss any differences or issues the staff might note between the staff's analysis and the applicant's analysis.

Next slide.

So, we have built a lot of MELCOR models for new reactors. We have MELCOR for all of the large light water reactors. I've listed them here on this slide. I think I've got them all, eight of them. I guess it's maybe more correct to say U.S. EPR and U.S. APWR.

The graphic here shows accident progression. This is kind of a graphical depiction. It starts on the upper left and moves around to the bottom left. The first one is the beginning of the accident. The next one is dryout of the steam generator and the core starts to crumble. The one on the bottom right shows the core pretty much all in the lower plenum of the reactor. And then, the one on the lower left shows the core after it's left the reactor vessel and it's sitting at the beginning of the core catcher. And this is the EPR.

Next slide.

We have also built and applied MELCOR models for SMRs. These are light water reactors. NuScale, mPower, Westinghouse SMR, and more recently and currently, the BWRX-300. We actually use the NuScale model quite a bit, and I'm going to talk about that in the next slide.

Next slide, please.

So, one thing that we saw with NuScale is that they developed their own source term from the reactor into the containment for the purpose of demonstrating that they meet the EAB and LPZ dose criteria offsite. So, the source term that NuScale came up with was a replacement for the source term in Appendix A of Regulatory Guide 1.183.

NuScale did this with MELCOR. They used MELCOR to estimate source term from the reactor into the containment. And part of their analysis and, typically, one of these analyses is the deposition in containment. And for that, actually, NuScale turned from MELCOR and actually decided to use STARNAUA for that. It's a separate code. It's a aerosol code. I understand it was developed years ago in Germany, and they use that.

So, when it was our turn to do our independent analysis, we, again, pulled out our tool MELCOR and we performed independent assessment for a number of scenarios from the PRA. And we also used RADTRAD to understand what that meant for offsite doses.

Next slide. Okay, next slide, please.

So now, I'm going to turn to the MELCOR Scale Non-LWR Source Term Demonstration Project. So, this is where we are in our work for non-LWRs.

As Hossein mentioned, we actually started non-LWR work a long time ago. For example, back about 10 years ago, we were doing some work on HTGRs. So, here's where we are today:

I'm going to talk, starting off with the first bullet here, I'm going to talk about the strategy for non-LWR source term analysis.

I'm going to talk about the objectives of this project to develop a MELCOR for non-LWR analysis and to apply it for several reference plant models.

I'm going to talk about some public workshops we did this past year.

And then, I'm going to show some sample results for the three plants we did this year. I'm only going to show a small sample of the work we did. We did a lot of analyses because we really wanted to give MELCOR a good shakedown, so that we would know that we have a tool that can be used for these reactors.

Next slide.

So, our strategy for determining source term for non-light water reactors involves applying SCALE in MELCOR. And this is really what we have been doing for many years, pretty much since I've been at the NRC back in the early '90s.

In the early '90s, we were using what is called ORIGIN, ORIGIN and MELCOR. And now, it's called SCALE. ORIGIN is part of SCALE. So, we've been using these codes to do source term at least since I got here in the early '90s.

Before that, in NUREG-1150, we tended to use -- well, we had a source term code package which was there were separate codes that we had to feed the output of one code into the next code manually.

And also, during NUREG-1150, we developed something called the EXOR code package, which, again, wasn't really a -- it was not a phenomenological code like MELCOR is. It was factors, you know, release fractions from the core, release deposited in the RCS, et cetera.

Anyway, so what we're doing is we're continuing what we've been doing. This is an extension of what we have been doing for many years. We're applying SCALE and MELCOR to the source term. And our strategy for non-LWRs is described in detail in Volume 3 of the code strategy. The cover of that is shown on the left side of this slide.

We also plan to use SCALE and MELCOR for safety analysis for fuel cycle facilities for non-LWRs. And that's shown on the left side as well. That is so-called Volume 5 of our code strategy.

Today, though, I'm going to be focusing on Volume 3 and the work we're doing for the reactors and not for the fuel cycle facilities.

Next slide.

DR. BLEY: Hey, Jason, this is Dennis Bley.

Something both you and Hossein talked about raises a question for me. And that is, over time, as you develop these codes, various people would raise questions about what's there and what's not there, and does this apply to the accident randomizing. And it caused more evolution as a code in development.

Now, for a new design and an applicant who's developed that new design who hasn't been involved in the development of these codes, has access to the website we've heard about today, I'm wondering what keys them to question whether the code's properly handling the physics of anything that's unique in their design. Is there any guidance or help with that that they'll have? Or is it just, when they submit it, you folks might find those things?

MR. SCHAPEROW: Yes, well, I think you raise a good point, which is that it is a state-of-the-art code. It is a large code. It's sophisticated. It models a lot of phenomena in different levels of detail. And it will be a challenge for a user to code. They need some experience.

And various vendors have handled that. One vendor -- I won't name them -- they hired one of the folks at Sandia who's been involved with development in application, when they got started. And I think they trained other people along the way.

But, yes, you're right, it is, it will be a challenge for them. But --

DR. BLEY: We may run out of those people.

MR. SCHAPEROW: Well, that's -- what do they call it these days? -- knowledge management. You know, people are apprenticed, and then, they learn how to do stuff. And then, they develop their skills.

Yes, we have MELCOR analysts and developers. At one point, the MELCOR project was really huge. We had people working on MELCOR development at Oak Ridge, Brookhaven, and Sandia. Now Sandia has always been the central spot, and that's where it is now. But, yes, there's a big community.

And people that use codes like MAAP, they understand this business quite well, too. They could probably jump right in, I imagine.

DR. BLEY: Okay. Well, it's something just for the overview. I mean, this is a session -- today's session is on how all of this stuff integrates together, and it's a pretty massive amount of integration.

I think we need some kind of guidance that helps people know to look for the kind of things that you folks have talked about that's evolved over the years. Anyway, you don't need to answer that now, but we need to have a focus on that at some point.

MR. SCHAPEROW: Well, it looks like

Hossein would like to add something.

MEMBER REMPE: To make it real clear, the best practices volume of the User Guide for MELCOR is something that would be useful for the non-LWR applications, right?

MR. SCHAPEROW: Hossein is ready to help out with that one.

MR. ESMAILI: So, can I just add something to what you are discussing? So, we have MELCOR workshops, right? So, as long as you are registered code users, you can come to the workshops; you can participate in these codes. You know, this is where Sandia is teaching you how to use the code, if there are additional models for non-LWRs.

And this is nothing new. You know, we started having workshops on HTGR and SFR during the pandemic back in the 2018 or 2019 timeframe. So, we always have those training classes for registered users.

As I mentioned, we have the theory manual, Users' Guides. You know, the users can go to those to find out exactly what the models are. They can always ask questions. So, they know what the theory behind what we put into the code is.

For the surface plant models, we are

working on reports, right? So, those reports are going to become available, right, to see what they have done; you know, when we are using the code, what we found out. And in addition, we can show the models. So, there's a whole bunch of things that are going to come together.

But I just want to go back to what Joy said earlier. I think it would be good to put everything in one place in terms of the non-LWRs, to have some type of best practices, et cetera. And I'm sure we can do that.

MR. SCHAPEROW: Okay. So, I guess I'd like to turn to the SCALE-MELCOR Non-LWR Demonstration Project. We set up three objectives for this effort.

The first is to develop our understanding of severe accident behavior for non-light water reactors, and that would help provide insights for regulatory guidance.

Second, we would like to have a dialog with the stakeholders on what we're doing with MELCOR and SCALE and the staff's approach for accident progression and source term. Because we're going to be doing independent analysis, as we always do, I imagine.

And finally, we're demonstrating how one could use SCALE and MELCOR for non-LWR analysis, and how they can be used to help identify characteristics that are important in an accident, and also, uncertainties that need to be addressed, or maybe that don't need to be addressed.

As part of this effort, we have publicly available input models that we're just wrapping up now, at least for three designs. This is new. We've never given plant models out before because the plant models that we have have proprietary information in them.

So, these plant models are based on conceptual designs of new reactors that are publicly available, supplemented by things that weren't in the reports. Like the reports, typically, don't specify what building is around the reactor, these design reports for these new reactors. So, we specified a building.

Next slide.

Regarding the scope of this effort, as I said, we've been doing a lot of work to add new models to MELCOR, such as a heat pipe model. And so, we are developing five full plant models for non-LWRs. We finished the first three this year, this past year. And this current year, 2022, we're doing the last two.

This slide shows how we've binned the new reactor designs. We have taken all the new reactor designs and we bin them, roughly, into five classes. And we're building a full plant MELCOR model for each of these five classes. And the individual models that we're building are shown here on this slide.

The first one is what was originally called the Megapower reactor that Los Alamos designed, but INL tweaked it and made some changes to it. So, we're modeling the INL Design A version of that.

The second reactor, the HTGR, we're modeling the PBMR-400 design, which goes back, again, about 10 years.

A newer design is the next one, the University of California Berkeley Mark 1 FHR. This is a pebble-bed reactor with molten salt cooling it.

And then, the last two reactors we're going to be doing are the molten salt reactor experiment and the advanced burner test reactor.

Next slide.

DR. BLEY: Jason, Dennis Bley again.

A sort of related question. This is wonderful you're going to have these, and they would be good starting places for many people.

Can you talk about -- thinking about the issues of what might be really unique and are we missing anything, more comfort in that area comes with a broader range of peer review and comment. And I don't know the best way to get that. Maybe you can talk about that if you put together an international panel or if you're submitting papers on these that would get international review? Anyway, if you can talk a little bit about the peer review that's going to be applied, or likely will be applied to these models, it would be helpful.

MR. SCHAPEROW: Yes. First of all, I think we spent a lot of time developing the code strategy. And the code strategy lays out the models that are needed for these new designs. And so, that's one thing we do factor in.

Second is we do have quite a bit of international collaboration going on. Actually, one of the first applications of MELCOR in the non-light water reactor area was by -- I think it was somebody in Hungary. Part of our annual severe accident research meeting, we met and they presented their work on how they applied MELCOR for non-light water reactors. So, yes, there is cross-fertilization, and there is, I guess you might call it, peer review going on there.

We've presented the work at public meetings this past year. The vendors came, some of the vendors came. We got some comments. Some of it was more on scenario selection than it was on phenomenological per se.

But we're going to be publishing reports, Sandia and Oak Ridge reports, ORNL reports, as part of this project. So, there will be other opportunities. We're presenting results at different conferences.

We're producing a new ref conference this year to present one of these results. Our lead analyst, Casey Wagner, is going to be presenting that.

So, I don't know if that fully addresses your question, but --

DR. BLEY: It's helpful. Thanks a lot.

MR. SCHAPEROW: Yes, I think, as with most things, the more we work on it, the more we're going to learn about it and the better our models will get. We're still finding things wrong with the Peace Bottom MELCOR model, little things, you know, at this stage, but there's still little things that get debugged. I guess you might call it bootstrapping. There will be some bootstrapping going on, I think.

Anyway, the approach for each of the five designs we're modeling, first, we start off with

SCALE. SCALE is used to calculate what's going on during operation, not during the accident per se, but during operation. So, as the reactor is operating, the gauge is building; radionuclide inventory is building in. So, we're using SCALE to calculate that because that's the starting point for the accident: what's the decay heat? What's the radionuclide inventory in the core?

Also, SCALE is providing us reactivity feedback coefficients, which we're using in MELCOR point kinetics models, if we have a reactivity transient.

So, the second step is to build a MELCOR full plant model. We're building that based on publicly available designs, design concepts, I should say, that have been proposed over the last number of years.

In some cases, we've had to supplement that because these design concept models, they're really are focused on the reactor and not so much on the building. So, running the reactor, and the building surrounding the reactor, of course, is an important place for fission particle deposition.

So, the third part is to select accident scenarios. Again, we're turning back to our LWR experience. We're doing the usual suspects, as you say, the ATWS scenario, the station blackout scenario, and the LOCA with no makeup scenario, those types of scenarios, the, quote, "severe accidents" where everything goes wrong.

The fourth step is to perform MELCOR simulations for the selected scenarios, including lots of sensitivity calculations to shake the models down. And also, we've done some uncertainty analysis. We've used MELCOR and we've developed MELCOR input models with -- I shouldn't say that. We varied some of the input parameters with using Monte Carlo tools, and we're running many as MELCOR simulations to look at things with a Monte-Carlo-type approach.

And finally, when we finish all that, then we sit down at these public workshops and we discuss what we've done, including the modeling and the results.

Next slide.

Okay. This is kind of our advertisements for the workshops. We just plopped them here on this slide. The top half shows the workshops we did this year, and then, the bottom half is going to show the workshops we'll do next year.

We have a QR code here. If you use your

camera, it plops you right to the website where it is. And also, if you want to see about these workshops some more, there's a link at the bottom.

I see a hand raised. Joy?

MEMBER REMPE: Yes, in the last slide, I didn't get my hand up in time.

MR. SCHAPEROW: Oh, I'm sorry.

MEMBER REMPE: And you said, well, we did do sensitivity analyses. And so, I asked the question earlier of Hossein, did you determine which parameters have the most impact? And where there's gaps? And are you communicating that to the folks in the workshop? I listened to some of some of the workshops, and I don't recall that discussion, but maybe I wasn't online at the time it was discussed.

MR. SCHAPEROW: No. No. Well, I mean, the first goal of the demonstration calculations was to show that the models we put into MELCOR worked; they work together.

MEMBER REMPE: Yes, and I think that's great you guys do that.

MR. SCHAPEROW: And, you know, that's a low bar, but MELCOR is a big code and it's a hard job.

The second part is to say, well, okay, how do you use the code to look and see maybe what's important, what you really -- what are the results sensitive to? And so, we did demonstrate, I think, how you would do that; how you would use MELCOR to show what the results are sensitive to.

But the next step is to show, okay, for this design, what are the results sensitive to? We did a little bit of that, but I think that's kind of the next step. I wouldn't characterize that we've done much of that at this point.

And one reason is the designs aren't finished. We had to like make up things. Maybe I shouldn't use the word "make up."

MEMBER REMPE: Sure.

MR. SCHAPEROW: We had to imagine what the building around the design would like, the building around the reactor. And again, the building can be an important factor in some of these analyses. You know, what happens when the coolant leaves the reactor and enters the building?

So, I hope that characterizes it. And maybe, Hossein, I don't know if Hossein had anything to add on this that I would characterize as well.

MR. ESMAILI: No, that is beautiful, Jason.

MEMBER REMPE: So, again, when I see some of the applications coming through, I see a lack of data because we all know getting the data is a long lead time and very expensive process. And so, communicating to the design developers what data you had to make up, because they are, also, by the way, making up.

MR. SCHAPEROW: Yes, yes.

MEMBER REMPE: This input is important, and you'd better have data or show us why you don't have or don't need that data. I think it's a very useful conversation to have with the design developers earlier on, so that we all have as efficient and effective licensing process as possible.

MR. SCHAPEROW: And I would suggest that the ACRS actually pointed that out with a recent review of a Source Term Topical Report, that they felt that someone needs to think about that a little more. I think it was release of fission particles from molten salt. So, yes, point well-taken.

MEMBER REMPE: I think we're trying to point that out, but it might as well be on this website, or whatever, a roadmap that you guys are developing early on, just to avoid problems in the latter end of things. But, anyway, just a thought.

MR. SCHAPEROW: No, thank you.

Next slide, please.

Okay. So now, I have about, gosh, I don't know how much time. Let me do the first one. I have sample results for three plants. So, I'd like to at least go through the first one. If that's enough, then we can skip the other two.

So, I would like to point out, though, before I get into that, that this was definitely a collaborative effort between the NRC, Oak Ridge, and Sandia, Oak Ridge being the experts and the developers of SCALE and Sandia being the experts and developers of MELCOR. And Oak Ridge did the development and running for SCALE for this project, and Sandia did the development and running for MELCOR.

Okay. Next slide, please.

So, this is the first of the three designs we did this past year. This is the INL Design A of the Megapower reactor. And what I'm showing on this slide is our models for this reactor. The left side of the slide shows the SCALE model which encompasses the reactor. And on the right, the other two graphics are the MELCOR model. All the way on the right is the MELCOR model of the reactor, and it even shows the reflector. This is a fast reactor, so it's got a reflector, and the external B4C shield.

And the center of the slide shows the MELCOR nodalization of the reactor building, which, again, we had to kind of make this one up. We decided that we would use something that looks a lot like a BWR Mark 1 reactor building, not the containment, but the reactor building, which has a specified kind of leak rate type of thing.

Okay. Next slide.

So, we did a few different accident scenarios with MELCOR for the heat pipe reactor design. The one I'm going to talk about today involves a reactivity addition accident with a delayed scram. So, those are the two basic assumptions, scenario assumptions, I would say: that (a) somebody's adding reactivity to this thing unintentionally, and then, the scram doesn't happen right away; it's delayed for about an hour.

So, I'd like to direct your attention to the upper righthand graph, the power graph. So, the power of this reactor is 5 megawatts. It's a small, it's a tiny reactor.

And so, what happens at the beginning of the accident, we have an inadvertent addition of reactivity by rotating of something called the control drums. So, this is analogous to control rising in an

LWR. And so, as the drums turn, reactivity gets added and the power goes up from 5 megawatts up to about 8.

And if you look at the bottom graph, you'll see the temperature of the fuel going up, starting around 1100 degrees up to about 1300. And then, there's a knee in the curve at 1300 Kelvin. Because what's going on there is the potassium inside the heat pipes is starting to boil, and the heat pipes aren't working as well now because they're not meant to work with boiling potassium in the pipes. That is kind of the end of them.

So, what MELCOR predicts at that point is the temperature really goes up even faster. At 2200 K, we have an assumption that the control rods are -- the shutdown rods are inserted. So, that's it. So, the rods go into the core and all that's left is decay heat. And so, after 2200 K, you see the temperature going down because the heat from the fuel is now moving outwards, and the fuel into other materials in the reactor, and then, finally, into the cavity surrounding the reactor. This is a passively cooled reactor. The heat just goes from the outside of the reactor vessel into the cavity, and then, there's air circulation in the cavity.

Next slide.

So, this is, again, just to give you a flavor of what our MELCOR analysis for a non-light water reactor looks like. We had a three-hour public workshop on this. So, this is really a highlight.

Anyway, turning to fission products, this slide talks about fission products. Well, first of all, the top righthand graph shows the pressure in the heat pipes. So, when the heat pipes heat up to 2200 K, the pressure goes up from 1 bar to almost 6 bar. And then, as the heat pipes fail, then the pressure comes back down to 1 bar.

The bottom righthand graph shows what's going on with the iodine. That's, typically, the nuclide of most interest in reactor safety analysis. So, about an hour into the accident, the cladding reaches 1650 K, and that's what we assume at full cladding failure temperature. It's stainless steel. And so, then, we have a release from the fuel, and that's the top curve, the blue curve in the bottom graph.

And then, as you can see, the next curve down is the iodine that's in the reactor vessel. So, as you can see, most of the iodine that's released from the fuel stays in the reactor vessel.

The next curve down is the green curve.

So, that shows how much of the iodine escapes from the reactor vessel into the reactor building.

And then, the bottom-most curve is how much escaped into the environment. Now, to have escaping from the vessel to the building, and the building to the environment, you have to have leak paths.

So, if you look at the center, the bottom of the center, I show what our assumed leakage was. Again, these are engineering assumptions. We don't know what the tech specs are for the reactor vessel leak rate or the reactor building leak rate. We had to just pick one, so we could demonstrate that MELCOR works and how you might do such an analysis.

Next slide.

CHAIRMAN PETTI: Jason? Jason?

MR. SCHAPEROW: Yes?

CHAIRMAN PETTI: Just a question.

MR. SCHAPEROW: Sure.

CHAIRMAN PETTI: When that heat path from that potassium goes into the secondary system, I don't know if it's an air-cooled system or if it's a water-cooled system, but you would know the reaction of the liquid metal?

MR. SCHAPEROW: Okay. Yes, so to answer

your question, I guess I would like to draw your attention to the center diagram. So, this is our fairly detailed cartoon of our model. The gray area is the fuel. The gray areas are the fuel, and the heat pipe is the green rectangle, the tall, green rectangle.

So, for this scenario, we found that the heat pipe failure was in the bottom region. That's where it was hottest. And there was no heat pipe failure up in the top region, where the connection is to the secondary system. So, the release was just into the reactor vessel. And then, the way the fission products made their way into the environment was through the leakage in the reactor vessel wall, and then, the leakage in the reactor building.

Now, regarding the question about oxidation of the potassium that left the heat pipe when the heat pipe got a hole in it, I don't believe we modeled it in the simulations. We do have models for metal oxidation in MELCOR. So, if we thought it was an important phenomena, we could certainly activate it or adjust it for potassium. We have one for sodium in there, I'm pretty positive, because we added it a while ago for sodium pool reactors.

So, I don't know if that addresses your

question.

CHAIRMAN PETTI: Yes. No, no, it's, as you talked about the progression of the accident, and this is a really different type of design.

MR. SCHAPEROW: Yes, really.

CHAIRMAN PETTI: And so, I always worry about that, that material getting out. You know, liquid metal getting out might be the worse thing that happens compared to the fuel.

MR. SCHAPEROW: Yes, I don't know.

CHAIRMAN PETTI: Yes.

MR. SCHAPEROW: The first for me, it was really interesting that we didn't really have what I would call a reactor coolant system anymore. We've got these heat pumps, which kind of is the analog. I should say, instead of one reactor coolant system with one hot leg, you know, now we've got -- I don't know how many of these -- 100 of these or 1,000 of these heat pipes. It's really different.

CHAIRMAN PETTI: Yes.

MR. SCHAPEROW: So, first of all, we had to add heat pipes to MELCOR. So, we did that, actually, a little over a year ago, a year and a half ago, I guess. Thanks.

So, the next two sets of slides I have

talk about similar accident analysis for HTGR and for the FHR. Does the Committee want me to go through these? I could. It's just it's going to take more than a few minutes, though. I'm not sure what the Committee would like to hear, since I'm about at the end of my time.

CHAIRMAN PETTI: Just hit what you think, if there's certain highlights, you know, that you think are important.

MR. SCHAPEROW: Okay. I'll go really fast through them, but I won't go into any detail.

So, this is the MELCOR model we put together for a high-temperature gas-cooled reactor and, also, the SCALE model, shown on the left. So, the SCALE model appears on the left. The next one over is the MELCOR model of the reactor, and the one on the upper righthand side, that's the reactor coolant system outside of the reactor. And the bottom diagram is the reactor building.

Next slide.

So, for this reactor, to demonstrate MELCOR for HTGR, we assume the loss of coolant accident. And that was really it; we assumed the loss of coolant accident.

So, the pressure dropped, as you can see on the top, and then, the bottom viewgraph shows the countercurrent flow in the pipe between the reactor and the break location.

Next slide.

We assume the control rods went in. But it was a large-break LOCA, so all the heat, all the reactor coolant, all the gases went into the containment.

So, the top righthand of the slide shows the flows in the core. The flows were slow initially in the core, but, eventually, they picked up a bit. And this is natural circulation. Because it's a loss of coolant accident, there's no pumping of the gas around the system. The bottom graph shows fuel temperatures predicted by MELCOR at different levels in the core, at different locations.

Next slide.

Tying to fission products, these are fission product results. Again, the releases are very small. The core didn't get that hot. I mean, these are 10 to the minus 8 iodine releases. I would characterize that as small. The bottom graph is for cesium.

Next slide.

The third and final MELCOR model we

developed last year was for a pebble-bed reactor with molten salt coolant. The SCALE model is shown on the left. The MELCOR model is shown on the right. So, this shows the core, the reactor coolant system, and the reactor building.

Next slide.

For this reactor, again, we did a variety of scenarios. We did ATWS. We did station blackout, and we did LOCA. We also did failure of all the -- we pushed this one pretty hard.

So, for the ATWS scenario we ran, it was an ATWS. So, the salt pumps shut off, but reactor fails to scram. So now, you're sitting there with an unscrammed reactor.

And we also looked at different amounts of decay heat removal from zero up to three full trains. We also included reactivity effects because, again, it was an ATWS and we have what I would characterize as a preliminary analysis, including xenon transient, because xenon is a big deal with you have reactivity accidents. At least that's what we found here.

Next slide.

On the left side, I show the reactivity predicted in the MELCOR point kinetics model using, again, the SCALE inputs. The first thousand seconds, the reactor is heating up because of the accident and you're getting negative reactivity insertion. And then, after about a thousand seconds, you really see an effect of xenon. The xenon really takes over. And for the next 100,000 seconds, the reactivity is really down because the xenon is adding all kinds of negative reactivity.

And so, the next graph over to the right shows the core power going down the first thousand seconds, and then, continuing to go down. At about 100,000 seconds -- well, 85,000 to be more precise -- the xenon is kind of gone. And guess what? The reactor's power starts coming up again.

So, if you look at the bottom righthand side, you'll see the power going up to up to about 20 megawatts-ish, and then, the core heats up and the power comes back down again. And we get oscillatory behavior, which kind of ends up the core power at around 7 or 8 megawatts thermal. That's equal to what the decay heat removal system has taken out. So, that's the new steady state for this reactor, assuming no rods in.

Next slide.

CHAIRMAN PETTI: Oh, for these sorts of systems, that's exactly what I've seen in the HGTRs, too. You know it's right when the power and the heat removal balance. Then, it's the new steady state until the rods get in, yes.

MR. SCHAPEROW: Okay. Next slide.

So, this just shows some sensitivities we ran. So, the reactor has a decay heat removal system called DRACS. It's got three trains. So, we ran the accident with all three trains, as I just showed you. And now, we also ran the same accident with two trains working, one train working, and zero trains working.

And so, these just show the end trends that you see from that. Maybe one of the more obvious things is on the right side. The fuel temperatures for the case with no trains just keep going up. That's the red line.

The initial transient, though, it stops at about 800 degrees C, but, eventually, there's no heat removal. This reactor, without the decay removal system, is kind of in trouble for a severe event like this.

Next slide.

Summary. We demonstrated the use of SCALE and MELCOR for three classes of non-LWRs. We're working on two more classes this year.

And again, to be a little redundant with what Hossein says, MELCOR simulates the entire course of the accident starting with the initiating event, with the hydraulic response, the fuel heatup, and then, the heat transfer out of the reactor into the reactor building, and then, into the environment. And then, last, but not least, the radiological release.

And certainly, MELCOR is unique in this regard. MELCOR has got the necessary aerosol modeling and similar modeling to handle tracking of fission products. It was one of the reasons that the SCDAP/RELAP went away eventually. SCDAP/RELAP did track what left the reactor, what left the core -- excuse me -- what fission products, but that was it. After the fission products left the core, they didn't track it. So, we really needed a tool like MELCOR. I'm going back, again, to the '90s.

Also, as part of this, we showed that you could use MELCOR to evaluate effectiveness of passive mitigation features, including things such as the DRACS system.

That's it.

CHAIRMAN PETTI: Hey, Jason, just a comment here. To me, the results also show the importance of the design of the building, right? And there's a message that has to go back to designers, because I think a lot of the advanced reactor designers, you know, everything's passive and I've got this functional containment, and I've got to make it by seismic, but there's an important fission product mitigation capability there that, as they are designing it, they should think about.

You guys have to assume something here to just show the calculations, but I think that they could be thinking about this sort of stuff when they're working on those designs. That's important, which they may not appreciate.

MR. SCHAPEROW: Also, and going back to maybe the NUREG-1150 days, I understand that the analysis, they included the containment, the deposition and containment, but they didn't always include deposition in surrounding buildings. Because a lot of times, well, the Mark 1, it's got a building around it. So, some of the fission products are going to end up in there. Or Surry, it has an auxiliary building. You know, if there's a pipe break in the auxiliary building, the fission products -- so, even if it's not, quote, "formally credited," or formally whatever, I mean, these buildings exist. And a lot of times, though, they're going to survive the accident. You'll have deposition there.

But thank you very much. Anyway, I don't have anything else to present today. So, I guess the next presenter, which I believe is Michelle Hart.

CHAIRMAN PETTI: Yes, that's right, Jason.

MS. HART: All right. So, yes, my name is Michelle Hart. I work in --

CHAIRMAN PETTI: Hold on, Michelle.

Just before we start, Arlon, you have the agenda? I don't have it open right now on my computer. When are we supposed to break for lunch? At one o'clock Eastern?

MR. COSTA: It's about this time, Dave.

MR. SNODDERLY: One o'clock. O'clock Eastern. So, yes, you're 15 minutes behind.

CHAIRMAN PETTI: Okay. So, keep going, Michelle.

MS. HART: All right. So, I'm Michelle Hart. I work in the Division of Advanced Reactors and Non-Power Production and Utilization Facilities.

Next slide.

And I'll be talking about the accident source term and near-term in recent applications.

So, the outline of this is:

First, I'll talk about some recent experience with SMRs and non-light water reactors.

Then, we'll talk about the Emergency Planning Zone size justification consequence analyses.

I'll give an example of an SMR design certification source term approach.

And then, we'll talk about some non-light water reactor early movers.

Next slide, please. Slide 82.

So, some of our recent experience is that we have been seeing some Topical Report for the SMR for NuScale, for example, and the design certification for NuScale. We've also had some advanced reactor pre-application interactions, which include Topical Report reviews and some license applications, as well as some meetings on special topics. And we have also contracted with the National Labs to develop some Source Term Development Reports.

Next slide, please.

So, when we talk about the Emergency Planning Zone size justification consequence analyses, the concept is based on NUREG-0396, which is the technical basis for the current regulation which says you should have an Emergency Planning Zone for plume exposure of about 10 miles in radius and an ingestion pathway Emergency Planning Zone of about 50 miles in radius. And that is in the regulation, and it is based on the analysis in NUREG-0396. And it's based on identification of an area within which prompt protective actions may be necessary to provide dose savings in the event of a radiological release.

And the feature of the analysis in NUREG-0396 is that it calculates dose at a distance for a spectrum of accidents. These analyses include the design basis accidents and severe accidents.

Next slide, please.

There were no separate or unique source terms developed especially for EPZ size analysis expected for this. In other words, you kind of re-use the source terms and accident release information that were developed for a safety analysis report or your PRA.

Next slide, please.

So, we've had some interactions on this with applicants and licensees so far. There's a methodology that TVA developed for the Clinch River ESP Emergency Planning Zone size justification in their Site Safety Analysis Report, and it supports exemptions to the 10-mile plume exposure pathway EPZ side requirement. It did not address ingestion pathway Emergency Planning Zones. And it was purely a methodology. It has not been exercised yet, and they have not selected an Emergency Planning Zone for the Clinch River site.

We've also had some interactions, and are still undergoing interactions, with NuScale on a Topical Report for EPZ sizing methodology. And that's a methodology to support plume exposure pathway EPZ size determination on a case-by-case basis for reactors under 250 megawatts thermal, which is currently allowed in the regulation.

And we also have a rulemaking underway for Emergency Planning Zone size determination for emergency planning for SMRs and other new technologies. And that's in a proposed new regulation 50.160. We have just issued, for Commission review and approval, the SECY that goes along with the final rule, and there guidance on analysis in Appendices to the related Regulatory Guide 1.242.

I will discuss the NuScale methodology next, and then, I'll talk some more about the rulemaking later.

Next slide, please. This is slide 86.

So, the NuScale EPZ Sizing Methodology Topical Report, which, as I said, is currently under review, it was not part of the design certification review, but it was submitted at the same time. It's currently on Revision 2. It's applicable to light water reactor SRMs such as NuScale, although it's not limited to the NuScale design.

And, in fact, in a response to a recent RAI, the scope of Revision 2 was reduced to advanced light water reactor SMRs. So, it's not for any advanced reactor, as originally envisioned in Revision 2.

Revision 3 is coming. Some recent discussions focused mostly on consideration of seismically initiated events and the probability of exceeding the acute dose criterion.

The applicability of this method to technologies other than light water reactors has still been a topic of discussion with the applicant, but, like I said, they just reduced it to advanced light water reactor SMRs.

And the Topical Report is a methodology for the analysis to determine the plume exposure pathway EPZ size. It also does not address ingestion pathway Emergency Planning Zone size.

Next slide, please.

So, in this Topical Report, source term refers to the fission product release to the environment as a function of time. Unlike with discussions of Reg. Guide 1.183 or TID-14844, in which the source term is released to containment, it includes the containment transport in the determination of the source term.

The Topical Report methodology uses source terms from the design basis accidents -- for example, from the NuScale design certification FSAR Chapter 15 -- and PRA severe accident scenarios that have been scoped into the analysis according to the methodology.

As I had talked about earlier, there's no separate or unique source terms that were developed especially for Emergency Planning Zone size analysis, and the NuScale methodology does use core damage frequency from the PRA to categorize severe accidents and select accident sequences to evaluate against the relevant dose criteria. And so, that's one of the features that makes it maybe not appropriate for non-light water reactors which may not have a core damage frequency in their PRA.

MEMBER HALNON: Michelle, this is Greg Halnon.

MS. HART: Uh-hum.

MEMBER HALNON: My understanding is there was a question about the threshold of that CDF and what was below regulatory concern, since there was such a large spectrum of accidents looked at. Was that ever resolved or is that still a question on the table?

MS. HART: So, that is still a question on the table. The review is ongoing. And in fact, some of the attempt to resolve some of that issue should be in Revision 3. Unfortunately, I'm not a direct reviewer of that anymore. I'm aware of what's going on, but I can't talk about any details. And because it is still under review, we can't talk about any details.

MEMBER HALNON: Okay. Because, during the 50.160 discussions, that was one of their sticking points on trying to get the Reg. Guide and some guidance there. We kind of punted towards this Topical Report coming up with some resolution to that issue. So, hopefully, that will be a resolution, but we'll be looking at that in the future, I'm sure.

MS. HART: Right. I think there should be a caution, though, about the methodology resolution of the issue may not be a general resolution, because we are taking into account the specific design and the specific design interaction with the environment that may affect that. Because the major concerns that we have are about how to account for external hazards and the probability of those events that are initiated by external hazards. And that may not be a general concern for everyone. And so, there may not be a single way to look at that, but, like I said, it is still something that's still under review.

MEMBER HALNON: Okay. Thank you.

MS. HART: Okay. So, next slide, please, which should be slide 88.

So, a different example -- also for NuScale, though -- is the design certification source term approach that they used. In August of 2019, we did write SECY-19-0079, which describes our review approach to evaluate accident source terms for both the Accident Source Term Topical Report that they had sent in, and we were reviewing at the time, and the design certification application for the NuScale SMR.

That SECY talks about how we're going to evaluate their change that they have just recently proposed -- "recently" being at the beginning of 2019 -- which moved away from mostly doing what was in Reg. Guide 1.183 with some specific information from the plant to something a little bit more unusual, in that they wanted to use a design basis source term without core damage for environmental qualification of equipment, and for other purposes in the plant. But they were still using a Reg. Guide 1.183-like source term for the offsite analysis and control room habitability analysis.

Next slide, please.

So, to talk about the source term methodology, it is an approved methodology for the NuScale SMR design. Its method is to develop accident source terms that are consistent with the Reg. Guide 1.183 guidance for PWRs, except for these two special events that they included, which is the core damage source term for the core damage event, which is the equivalent to the maximum hypothetical accident LOCA that we do in Reg. Guide 1.183, Appendix A, and the iodine spike design basis source term, which included no fuel damage, but release of the coolant to the containment.

This Topical Report is a reference in the design certification application FSAR, and COL applicants will incorporate by reference the design certification analyses that are listed in the FSAR, if they reference the design certification itself. The Topical Report is not necessarily likely to be implemented separately for COL applications that reference the certified design.

The ACRS has reviewed the staff's

evaluation of both the Accident Source Term Methodology Topical Report and the accident radiological consequence analyses in the design certification application. Therefore, I'm not going to go into a lot of great detail about it, but I'll summarize some of the important features of the source terms.

Staff found that the methods described in the Topical Report to develop design basis accident source terms were acceptable and consistent with the guidance in Reg. Guide 1.183 for PWRs. The two source terms, like I have listed here on the slide -- the core damage source term used to develop the core damage event and the iodine spike design basis source term for use in the evaluation of environmental qualification of equipment -- do not follow specific guidance in Reg. Guide 1.183, but they are generally consistent with the guidance in Reg. Guide 1.183 for similar types of analyses.

On the next few slides, I will describe NuScale's methodology to determine the core damage source term and implementation of the design certification FSAR analysis.

Next slide, please. So, this should be slide 90.

The core damage event was developed to show compliance with 10 CFR 50.47(a)(2)(iv), "The Offsite Dose Criteria" -- you know, the 25-rem that we're all familiar with -- and also address the discussion of fission product release in footnote 3, which is my favorite footnote in the regulation. You know, when you talk about doing this accident analysis to look at offsite consequence analysis, the fission product release should be assumed, for this evaluation, "should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accident events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products."

NuScale's core damage event is similar to a maximum hypothetical accident, in that it is not a specific scenario, but is intended to represent an accident with major damage to the core. NuScale's methodology is, in concept, similar to that used to develop NUREG-1465 and the Reg. Guide 1.183 LOCA source term. It is derived from a range of accident scenarios that result in significant damage to the core, informed by the PRA, and it is intended to be representative or bounding of a dominant majority of intact containment core damage events for the NuScale nuclear power module.

Next slide, please.

The core damage event dose analysis in the design certification application FSAR implemented the NuScale Topical Report methodology to determine the core damage source term. And so, it gave the actual release values.

The core inventory was calculated using the SCALE code for the core for the NuScale SMR design.

And they selected scenarios based on the NuScale SMR PRA, internal events only, and there were five surrogate scenarios that were various failures of ECCS, with decay heat removal system available and an intact containment.

Next slide, please.

In the NuScale design certification application and in their Topical Report methodology, there was one release phase, unlike in Reg. Guide 1.183, which has a gap release phase and an in-vessel phase.

MELCOR was used to estimate the release timing and magnitude for each scenario of those five scenarios, and the release onset and duration from the scenario with the minimum time to core damage was used for the core damage event source term.

And the core release fractions were taken from the median of the scenarios for each of the release -- I was going to say, "release categories"; that's not the right term -- the species.

Time-dependent aerosol removal rates were, then, calculated using the STARNAUA code. They did not use MELCOR for that portion of it. And those are aerosol removal rates within the containment for the SMR.

They did use design-specific input from thermal hydraulic conditions that were calculated by MELCOR for a surrogate scenario with the minimum time to core damage.

Next slide, please.

So now, to move on to source term approaches for non-light water reactor, the ones that we're talking to right now.

CHAIRMAN PETTI: Michelle, before we shift there, NuScale kind of lays their own path. But you, the staff, found that it was generally consistent with the methodology in 1.183. I'm struggling with, is the guidance there adequate enough for other SMR vendors?

Because there's all sorts of stuff in here they did that is consistent, but is it captured in a functional away? Because there's a lot of design-specific stuff that doesn't make sense. Is there enough there that another SMR vendor could read that and figure their way through to get something that would be acceptable?

MS. HART: Right.

CHAIRMAN PETTI: So, take what NuScale did and think about it functionally in line with the attributes of a source term, or whatever you want to call it. Is that all there clearly enough, do you think?

MS. HART: So, I think, in my opinion, if a light water reactor SMR, or even non-SMR, vendor would look at NuScale's Topical Report methodology, because it is a methodology and it talks about how to go about developing the source terms, that would give them an idea about how to go about developing source terms. And the choices that NuScale made, which is not necessarily true that another SMR vendor may have to make the exact same choices, but it would tell them what they did. And then, they could also look at the information in the design certification application and in our SE to see how it was implemented and the results that came out of that.

So, the methodology, when we reviewed the methodology -- and I do admit that we were reviewing the methodology at the same time as we were reviewing its implementation. So, that did help us determine the importance of some of their assumptions.

You know, Jason talked earlier about we did use MELCOR and RADTRAD to evaluate some of the questions that we had about the methodology. And we could do that in a more structured kind of way because we had more detailed information on the design itself, because we were reviewing the design at the same time.

CHAIRMAN PETTI: Yes, sometimes methodologies can be abstract, right?

MS. HART: Yes.

CHAIRMAN PETTI: And it's hard to know what it means? Right. I understand that.

MEMBER KIRCHNER: Michelle?

Okay. Go ahead, Walt.

MEMBER KIRCHNER: Yes, thank you, Dave.

I thought what was important in the NuScale approach was, consistent with the requirements of 10 CFR 52, they did assume MHA, as you said, that led to core damage. What happens when someone else picking up this methodology does not assume an MHA or assumes that there is no core damage?

MS. HART: Right.

MEMBER KIRCHNER: Then, how do you proceed?

MS. HART: Right. And so, I think, certainly, if somebody is going through development of their plant and development of their licensing strategy, it would really be helpful if they came and talked to us about that and tried to get an early read. You know, they could send in a white paper or they could send in a Topical Report. And even if they wanted to go to the extent of providing us a Topical Report with the source terms results in it, not just a methodology, to get really early "buy-in," quote-unquote, on their approach, I think, in general -- and this is something that we're talking amongst ourselves about and kind of developing our position on this -- you know, if it's not as obvious that it's a core melt source term with a maximum hypothetical accident, like we're talking about in the footnote, they may have to request an exemption.

And depending on the information that they use to justify that, you know, if they have a full structured analysis of the accidents and their facility, and the risk from their facility, they may be able to justify an exemption from that particular requirement. Because, as we were talking about earlier, you know, there is this idea that, with like the Licensing Modernization Project, that you would do that full assessment, and in conjunction with functional containment, the features of the assumptions behind the regulation, that you would have this large release into a containment, but with a demonstrable leak rate that you would evaluate for this assessment, you know, none of those features may be there for your particular design.

Does that help answer the question or? I know it doesn't necessarily give any certainty per se, but it is certainly --

MEMBER KIRCHNER: Michelle, there is a statement that what was key, I think, when we looked at this was they did assume an MHA and core damage, and then, proceeded.

MS. HART: Right.

MEMBER KIRCHNER: The problem I see is that you may have applicants that say, "We don't have a source term as NuScale designs it."

MS. HART: Correct.

MEMBER KIRCHNER: And then, you're at a juncture where, does the staff have in its evaluation -- like you say, the pre-application meetings between the applicant and the staff will be very important in resolving it. But I see that as the stumbling block going forward with some of the non-LWR concepts.

MS. HART: And I take that as a good point, yes.

Are there any more questions about this?

CHAIRMAN PETTI: Yes. Because I think this idea of core damage, I mean, could mean just a large fission product release. Because some reactors don't have the same equivalent damage as a light water reactor.

MEMBER KIRCHNER: Yes, that is what I mean.

CHAIRMAN PETTI: Yes. I mean, think of an HTGR. They're going to come in and they're going to show you really low releases under anything inside the design basis because the fuel is designed to handle all that. And they come in and go, "We don't even have damage, but we are going to postulate something where we get more release." That might, hypothetically, be acceptable because it's kind of meeting the intent of that footnote without the exact detail.

MEMBER KIRCHNER: Precisely, Dave, yes.

CHAIRMAN PETTI: Right. And I guess the

question in my mind is capturing some of that thought process in some paper, the value of doing that for some of these advanced systems. Because sometimes, really, they read it and they go, "Oh, well, that's LWR stuff. We don't need that."

But, no, hold it. Step back. Go through that thought process. Figure out what the real intent is behind it, and is the intent behind it well enough documented on the staff side, so that you could have productive discussions going forward? That's sort of a thing in my head.

MS. HART: Yes, those are the discussions that we have been having with some of the folks so far. I mean, those exact points are the points that I try to raise and I think we try to raise, that you need to evaluate what potential releases are from your facility and you would use your bounding one.

And there may be different ways to go about that. It may be a purely mechanistic analysis. It may be a conservative maximum hypothetical accident, much more like you would see from a non-power reactor, where you make unphysical assumptions because you know it would bound anything physical that could happen.

I say that guardedly because we have to

evaluate what potential things may actually appear.

Yes, Dr. Rempe?

MEMBER REMPE: Just call me Joy.

You've heard several of us mention it would be nice to have more guidance, a roadmap, or whatever. Are you hearing that from the design developers coming in?

MS. HART: No, we haven't actually heard that, and I think a lot of them -- I mean, we've only talked to a few. They all have plans to do things like PIRT. They all have plans to do things like evaluate what potential accidents are capable for their reactors. I think they all, the ones that we have coming soon, do plan to talk to us about accident source term and how to go about that.

So, I don't know that we're necessarily hearing everything. You know, I can't say that. But we haven't had any specific questions about this, I will say, not to say that maybe they just haven't brought it up because they might not be in that portion of their preparing for licensing yet.

MEMBER REMPE: Yes, we're all kind of saying the same thing, but they're not saying it. That's kind of interesting. Thanks.

CHAIRMAN PETTI: Yes, I agree with you,

Joy. I'm just wondering if it ought to be an item on the Part 53 or to put it on the list to try to get more direct feedback from a broader set of stakeholders. That might be interesting.

MEMBER REMPE: But a lot of them are using Part 50 and 52. So, I'm not sure, you know, if Part 53 is the only place to think about it.

CHAIRMAN PETTI: But whatever is the relevant setting to get that broad stakeholder feedback, I guess that's -- to be more direct and ask that question of them, instead of waiting until you get a pre-application.

Okay. Let's keep going then.

Oh, hold it. Sorry, I see some hands.

John Segala?

MR. SEGALA: Yes, this is John Segala from NRR.

I was just going to say, you know, we have our periodic advanced reactor stakeholder meetings and we do ask at those meetings if there's need for guidance, and whatnot. So, I think that is something that we could explore at a future meeting. I think the next one is coming up in mid-March.

CHAIRMAN PETTI: Okay.

MS. HART: All right. So now, to move on

to non-light water reactors. The people that we've talked to recently and people we're going to talk to soon -- of course, one of the most recent interactions we've had is with Kairos Power. They did provide us a Topical Report for the mechanistic source term methodology. It's still under review. We did come to the ACRS with a presentation on our evaluation of their Topical Report methodology just last fall.

And it's a methodology for applicants to develop event-specific radiological source terms for offsite consequence analysis. It does not include control room habitability or other purposes like normal effluents, or things like that. It's design basis accidents for the siting and safety analysis and AOOs and design basic events for the Licensing Modernization Project process.

We are also right now going through a review of the Hermes construction permit application. It evaluates a maximum hypothetical accident, which is a deterministic analysis, and it refers to the Kairos Power Mechanistic Source Term Topical Report for some of the details about the phenomena and how the phenomena are modeled.

Kairos, in their construction permit application, did use the SRP for non-power reactors, NUREG-1537, in preparing its application. And the maximum hypothetical accident which they developed is intended to bound all potential accident source terms.

I will say, in the construction permit, they are not providing specific consequence analyses or specific highly detailed analyses; that that is something that there is an analysis, but this maximum hypothetical accident may be more bounding than would be expected for an operating license application. It's not refined for the specific design at this point. And like I said, it's under review. We haven't asked initial questions about it yet.

Next slide, please.

We also have talked to X-energy. They propose to use a developer-made source term code, which they call XSTERM, which includes modeling of radionuclides from generation in the fuel to release and dose.

The Topical Report was submitted in April 2021, after they had previously submitted a white paper on the concepts that they were talking about for the source term verification and validation for the code suite, which they had submitted back in 2019 and we had given them comments on, or not comments, but feedback. When they submitted their Topical Report in

2021, they did address some of those comments.

However, we had some questions about the Topical Report, which was to provide the approach taken by X-energy to event-specific mechanistic source terms for their Xe-100 plant. We had questions or discussions with them about the scope of the report, and subsequently, they withdrew the Topical Report in order to revise it and submit a proposed revision to the review scope request. So, we're not currently reviewing anything for X-energy at this time.

Next slide, please.

As we know, the Oklo Aurora COL application review has recently ended without a resolution at this time. They had proposed a maximum credible accident without release.

With respect to the accident source term topic, staff was focusing its review on Oklo's determination of the maximum credible accident scenarios for the Aurora. And so, we had not finalized our evaluation of that at the time we ended our review earlier this year in January.

The staff is in pre-application interactions with several other designers, but we do not currently have white papers or Topical Reports on accident source term for review at this time.

In a public meeting on January 13th of this year, TerraPower presented a description of their development of source term methodology for the Natrium reactor, and stated they plan to submit a Source Term Topical Report in April of next year. They also stated that they are planning a presentation to the staff on functional containment in April of this year.

Source terms and methodologies for other designs have not been submitted to the NRC yet. However, we do have some information on what submittals to expect. For example, the regulatory engagement plan for the Westinghouse eVinci microreactor shows that Westinghouse plans to submit a report on mechanistic or accident source term development that would describe the computer code, the code qualification plan, and outline the methodology that would be used to generate the mechanistic source terms. The submittal timeline is not public.

Regulatory engagement plans and other information on pre-application activities, including meetings and submittals on source term, may be found on the NRC's public website under "Nuclear Reactors," and then, "New Reactors," then "Advanced Reactors," then "Licensing Activities," then, finally "Pre-application Activities". And if you do have a copy of the slides, the blue underlined is a link to that site, and it will tell you what information we have in-house or are expecting.

Next slide, please.

And that ends my presentation on this particular topic.

CHAIRMAN PETTI: Thank you, Michelle. Very helpful.

Anybody have any questions before we break for lunch?

MEMBER DIMITRIJEVIC: Yes, I do.

CHAIRMAN PETTI: Go ahead, Vesna.

MEMBER DIMITRIJEVIC: I have a very simple question. What's the relationship between maximum hypothetical accident and maximum credible accident?

MS. HART: You know, that's a very good question. Neither term is explicitly defined in the regulations or even in any of our guidance.

Maximum hypothetical accident is a term that has been used widely in non-power reactors, in the test reactor world. And in there, if you look in NUREG-1537, it kind of implies that it's an accident that is something greater than would be credible. It may be a non-physical accident. You know, assume that all of the cooling fluid just immediately disappears;

there's no specific reason for it.

Whereas, maximum credible accident, which is a term that may have been used in the past by some applicants to refer to like the TID-14844 source term, although some people have called that a maximum hypothetical accident as well, is it's not an accident that would be considered to be exceeded by any accident considered credible. I mean, it should be your most credible or the worst thing that could happen in credible space.

But credible is not, and has not been, defined by any specific likelihood or probability of the event or release category, or anything like that, unfortunately. They may or may not be the same accident, but I think there is a flavor to maximum hypothetical that would say that it's something that you think is something that would not occur in the facility.

MEMBER DIMITRIJEVIC: So, is it fair to say the maximum hypothetical accident is probably not credible? And what I want to say is, is it fair to say that "credible" is not well-defined in maximum credible, and "maximum" is not well-defined in maximum hypothetical accidents?

MS. HART: I think maybe the only thing

that may be well-defined is "accident," but, you know, I wouldn't even say that much.

(Laughter.)

I understand the question, because it is a struggle that we've had, you know, certainly when we were going through the Oklo review, because they were calling their accident "maximum credible." It's a struggle to determine how have they determined that and what do we think are the right things to consider in determining that. And so, I think it's easy to say if something is obviously bounding of anything that could happen in the facility. It's harder to say, when you try to refine that more and get closer to what you think may be an actual expected credible accident, whatever that may mean.

MEMBER DIMITRIJEVIC: Thank you. That's a very interesting discussion, especially if you look at these surrogate accidents around NuScale, you know. Are they credible or maximum, or whatever?

But thanks.

MS. HART: Yes, I think the thing that we're aiming for in that regulation, in that discussion, in that footnote is, you know, let's try to see what would be a bad thing that could happen to test the containment. And if you don't have a specific containment you're testing, well, what would be the bad thing, the worst thing that you would expect to use in your design basis for your siting and for your safety systems that you're crediting in retaining the radionuclides? It's a topic that is very nuanced and may depend on the information that you have in front of you.

MEMBER HALNON: So, wouldn't it behoove the staff to try to define those terms, so that it's not across the map in how people use them?

MS. HART: I mean, I don't know. I mean, it is something that we are discussing among ourselves. It is something that has come up several times. I don't know that there's a specific definition; like if you use a specific frequency of the event, you know, if that would capture everything that's necessary, or if it's widely acceptable for all potential uses.

MEMBER HALNON: Okay.

CHAIRMAN PETTI: I see some hands.

John? And then, Steve.

MR. PARILLO: Yes, this is John Parillo.

I just wanted to point out to the Committee that the Draft Guide 1389 does contain a definition of an MHA which is similar to the wording that Michelle has used, but I can read it.

"The maximum hypothetical accident, also referred to as a maximum credible accident, is that accident whose consequences, as measured by the radiation exposure to the surrounding public, would not be exceeded by any other accident whose occurrence during the lifetime of the facility would appear to be credible. As used in this Guide, the term `LOCA' refers to any accident that causes a loss of core cooling. The `MHA LOCA' refers to a loss of core cooling resulting in a substantial meltdown of the core with a subsequent release into containment of appreciable quantities of fission products. These evaluations assume containment integrity with offsite hazards evaluated based on design basis containment leakage."

So, that's now in the Draft Guide which we will be presenting, I believe it's March 16th.

MEMBER HALNON: Okay. So, the argument is going to be on "credible," not "maximum" or "accident".

CHAIRMAN PETTI: But I also think that it depends a little bit on -- that may be a fine definition for water-cooled reactors, but, for advanced reactors, it probably needs some noodling.

Because if you have functional containment, it might be a degradation of containment in some way, not necessarily failure, but some of the values somehow get degraded. I mean, there is a lot of, as Michelle says, nuance when you look at it through the lens of an advanced system, I think.

MEMBER HALNON: You're going to have to couple some external event to it.

CHAIRMAN PETTI: Steve?

MR. LYNCH: Sure. Hi. This is Steve Lynch, the Acting Chief of the Advanced Reactor Policy Branch.

I just wanted to add a little bit of how we've thought about maximum hypothetical and credible accidents in the non-power world, and how that might be able to be extrapolated into some of the advanced reactor concepts.

So, while we don't have an official definition for an MHA or an MCA, we do have a discussion of what to consider when developing one of these accidents in NUREG-1537, both the format and content guide for applicants developing applications, and the Standard Review Plan used by NRC staff to review these applications.

And typically, when we're thinking of a

hypothetical accident, rather than trying to go down the path of is it credible or non-credible, we've found it helpful to think of it more as, is it a non-physical event that is what we are calling the MHA? So, for example, it could be an assumption of a fuel pin exploding in air, where there's not a physical mechanism that would allow for that event to happen. But it does provide a release of fission products that is greater than any design basis accident.

And really, when we're looking at MHAs, where that concept is most useful is when you're looking at facilities that have very low consequences to begin with, where you're looking at your suite of design basis accidents, and in the case of non-power facilities, you're not finding a design basis accident that's exceeding 100 millirem at the site boundary. So, then, in order to reduce the burden of some of the analysis that needs to be done, you may look for something that is hypothetical that still demonstrates, even in a non-physical way, that, hey, we're still not exceeding this very low consequence for any accident that could be anticipated at the facility.

Then, when we start looking at facilities that might be somewhat larger, and just using examples of some of the medical radioisotope facilities that are designing to be less than 1 rem at the site boundary, that's when we've started looking at considerations for maximum credible accidents. And we've looked at that as being derived from the design basis accidents at the facility, where there is still an expectation that an applicant has looked at initiating events and all of the accident sequences within families, and then, they select a bounding event from those. And that credible accident is still, more or less, used as a conversation piece and a way of demonstrating the overall risk of the facility, but does not replace any other accident analysis that would otherwise be expected for the facility.

So, that's just to give some insights of how the staff has been treating those in some other reviews; that while it hasn't relied on a specific definition, we have had general practices in place for considering these.

Thank you.

CHAIRMAN PETTI: Joy?

MEMBER REMPE: Sure. I'm glad you brought this up, Steve, and I'm glad you went and actually mentioned the fact that sometimes a design basis accident that's been evaluated by an applicant might suffice for the maximum hypothetical accident. Because I think there are some good insights, and I'm not sure that those insights are yet on the web page, or whatever guidance the staff's developing. And I am not sure that the design developers that are coming for Part 53 will be looking at the non-power applications, the impact applications. And so, I do think that there's a synergy that could be explored here, especially, again, if they do a systematic approach to come up with the initiating events.

But, anyway, I'm glad to hear you mention these insights.

CHAIRMAN PETTI: I've captured all this. I think there will be a recommendation coming. Good.

Anyone else?

(No response.)

Okay. Then, let's pause her for lunch and see everyone at 2:00 p.m. Eastern time.

Thank you all.

(Whereupon, at 1:02 p.m., the foregoing matter went off the record for lunch and went back on the record at 2:00 p.m.)

CHAIRMAN PETTI: Okay. It's two o'clock Eastern. Let's continue.

Michelle?

MS. HART: Hi. Welcome back.

So now, for the next topic, we'll be talking about the accident consequence-related regulation activities, and there are several regulation development activities ongoing where source terms will be used. And that's in addition to Part 53, which, of course, is happening.

Next slide, please.

So, the first thing we want to talk about is there is a petition for rulemaking. It was received at the end of 2019 and docketed at the beginning of 2020. And there is The Federal Register notice for it. It's under evaluation. There is no disposition on it yet.

The petition requests a voluntary rule to allow power reactor licensees to adopt an alternative to the accident dose criteria specified in 50.67, "Accident Source Term." The petition proposes a uniform value of 100 millisieverts, or 10 rem, for offsite locations and for the control room. So, they would be equal to each other, instead of 25 rem offsite and 5 rem in the control room.

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CHAIRMAN PETTI: Michelle, does it say who petitioned this?

MS. HART: So, the petitioner is actually an NRC employee, John Parillo. He is on the line.

CHAIRMAN PETTI: Okay. I just wanted to know if it came from industry or where it came from. That's good. Okay.

MS. HART: Yes.

CHAIRMAN PETTI: Thanks.

MS. HART: Yes, there is more detail online. If you go to the rulemaking website, you can see the actual petition that he sent in.

CHAIRMAN PETTI: Okay. Thanks.

MS. HART: So, the next topic is the emergency preparedness for small modular reactors and other new technologies rulemaking. The stage we're in now is the final rule is in development. And as I had mentioned previously, there is this new section, 50.160, and related and conforming changes to other portions of Part 50.

The staff presented the rule text and supporting guidance to the Subcommittee in September and the full Committee in November of last year. For the purposes of this topic meeting, I won't be discussing the rulemaking in detail today, but I will describe how the supporting guidance in Reg. Guide

1.242, to be issued with the final rule, currently discusses accident consequence analysis and information on source terms.

We did present the draft for Reg. Guide 1.242 to the ACRS, which, when issued for public comment, had two appendices: Appendix A, on the methodology for analyses to support determination of plume exposure pathway EPZ size, and Appendix B, which provided information on source terms.

Reg. Guide 1.242 includes further clarifications and consideration of the public comments. The changes incorporated into Reg. Guide 1.242 were presented to the ACRS last fall.

Next slide, please. This should be slide 101.

Appendix A to the Reg. Guide, "General Methodology for Establishing the Plume Exposure Pathway Emergency Planning Zone Size, provides general guidance on the consequence analysis and discusses event selection and consideration of accident likelihood in determining the scenarios to be used in Emergency Planning Zone size evaluation.

In selecting events for the Emergency Planning Zone size evaluation, an applicant should use licensing basis events, such as design basis events, beyond design basis events, and DBAs that they would determine through the LMP process, or the MHA, if they chose to use that in their licensing scheme, and severe accidents, as candidate radiological release scenarios.

Considerations include internal and external initiators, all sources of radioactivity release, multi-module and multi-unit considerations, and the event likelihood, including uncertainty, as well as the timing of the releases.

Next slide, please.

Appendix B to the Reg. Guide, "Development of Information on Source Terms," provides high-level guidance on how to develop source terms for plume exposure pathway EPZ size evaluations. It does not provide specific source terms, but it does give reference to information that could be used to develop source terms. And like we had said previously, the source terms that they would use for these Emergency Planning Zone size determinations would be reuse of source terms they've already analyzed for safety analysis and/or the PRA.

Next slide, please.

Another rulemaking activity that's ongoing right now is the alternative physical security for advanced reactors. It is a draft rule and guidance in development right now.

The proposed rule would establish alternative physical security requirements for advanced reactor technologies that would include SMRs and other non-light water reactors to protect against adversary attack resulting in radiological consequences to public health and safety.

So, voluntary alternative physical security requirements would be commensurate with potential consequences to the public health and safety and the common defense and security. The idea is that there would be site-specific analyses demonstrating how the identified alternative physical security requirements meet the applicable performance requirements.

This rule has not been issued as a draft rule yet. It's still under development. It should be coming out soon. And so, I can't really describe the details yet, but they will be available for public comment before long.

However, I can give you some flavor as to --

CHAIRMAN PETTI: Michelle?

MS. HART: Yes?

CHAIRMAN PETTI: Soon? A month? Six months? What is "soon"?

MS. HART: So, it's in the final concurrence kind of status right now. So, I don't have the schedule up in front of me, I'm sorry to say.

CHAIRMAN PETTI: Okay.

MEMBER BROWN: This is Charlie Brown.

What's driving a physical security change for advanced reactors? I mean, a plant is a plant; barbed wire is barbed wire, and fences are fences. I mean, they're still reactors.

MS. HART: Right.

MEMBER BROWN: That's all I'm curious about.

MS. HART: So, you know, the rulemaking plan for this, and the Commission direction on this, was to do a limited-scope rulemaking to address the concerns that some small modular reactors would have in the near term. Before we develop Part 53, there may be a further investigation of how you would include considerations for design in your security for your facility.

MEMBER BROWN: I guess I still don't understand. I mean, you're still going to have guards. You've got to have some type of administrative requirements in buildings.

MS. HART: So, I am really only here to talk about the source terms.

MEMBER BROWN: Okay.

MS. HART: I can't really speak to the alternative requirements.

MEMBER BROWN: All right. I've got it.

MS. HART: That's not what this meeting is about.

MEMBER BROWN: Thank you. Okay. I'm done.

MEMBER REMPE: Michelle?

MS. HART: Yes, Joy?

MEMBER REMPE: I'm sorry, maybe you're not allowed to answer the question, but there are Reg. Guides and SRP changes, or is it just Reg. Guides? And can you say which ones?

MS. HART: So, we are developing a new Reg. Guide to go along with this rule, and when there are some revisions to a current Reg. Guide on like target set analysis. And so, I don't recall the numbers of the Reg. Guides right off the top of my head, but there are two Draft Guides that would go along with this rulemaking.

MEMBER REMPE: Okay. That helps. Thank

you.

MS. HART: Uh-hum.

To give you a little bit of flavor about the stuff that we're thinking about, though, it is that, you know, kind of similar to --

MR. SEGALA: Hey, Michelle?

MS. HART: I'm sorry. Yes?

MR. SEGALA: Yes, this is John Segala.

I just wanted to say, the proposed rule publication date is November 14th, 2022.

MS. HART: Thank you, John. I couldn't remember.

MR. SEGALA: But it's just now starting to go through the concurrence process as of right now.

MS. HART: Yes. Thank you, John.

So, with regard to source terms and consequence analysis that may be used -- and, of course, you know, it still hasn't been issued -- but, like with the emergency preparedness rule, we are reusing safety analysis source terms and just determining which scenarios we need to use for that assessment.

So, I think you would start with that same basis for the security rule, but there are other considerations you would have to bring into play, such as potential different release pathways and different initiators. So, that will be something that we would expect to discuss in the Reg. Guides that would go with that rule.

Next slide, please.

So, that is all the slides I had on this particular topic. Are there any other questions?

(No response.)

CHAIRMAN PETTI: Hearing none, let's just keep on moving.

MS. HART: All right. So, the next topic that we have is guidance and information for developing source terms for non-light water reactors. And in this one, I will have some coworkers that will help with the presentation.

Next slide, please.

So, the outline of this particular presentation is that we are going to, first, talk about accident consequence analysis for advanced reactors in a general sense.

Then, I'll describe mechanistic source term.

Next, I'll talk about recent reports on non-light water reactor source term development, those National Lab contractor reports.

We'll talk a little bit about the non-light water reactor PRA standard and the source term discussion in there.

And then, Bill Reckley will talk about the Licensing Modernization Project and the source term that goes along with that.

Tim Drzewiecki will talk about an overview of the method in NUREG-2246, "Fuel Qualification for Advanced Reactors," and how that may interact with source term. We'll also discuss that.

And then, the last topic, we will talk about this non-light water reactor accident source term information web page that we are developing.

Next slide, please.

So, when we talk about source terms, including development of mechanistic source terms, we're talking about the source terms for accident assessment. Just as a reminder, there are other source terms and other radiological sources that are part of regulatory assessment, such as those for shielding design or effluents and rad waste system design evaluation, worker protection, things like that. So, when we're talking about source term today, we have only been talking about the accident assessment, and mainly, the offsite consequence analysis.

The siting and safety analysis requirements are the same as used in previous licensing of large light water reactors. We have not changed the regulations for advanced reactors and other new technologies, and the offsite dose consequence reference values and the evaluation of plant design and siting remain the same currently.

Newer uses for advanced reactors including the Licensing Modernization Project process for non-light water reactors to select licensing basis events; classify system structures and components, and evaluate the adequacy of defense-in-depth as described in NEI 18-04 and endorsed in Reg. Guide 1.233.

As we discussed earlier, consequence analysis may be used to aid in plume exposure pathway EPZ size determinations; to support exemptions, or to provide a case-by-case basis for gas-cooled reactors or reactors with rated thermal power less than 250 megawatts thermal, in compliance with the current regulations; or, to support applications of the alternative framework for emergency preparedness for SMRs and other new technologies, the 10 CFR 50.160 rulemaking, once the rule is issued as final.

And as I just discussed, there's an

ongoing limited-scope rulemaking for alternative security requirements that use a consequence assessment to say that use of alternative security requirements is acceptable. And the proposed rule and guidance has not been issued yet.

Next slide, please.

So, some considerations in the development of accident source terms is, of course, you have to determine which events you're going to look at, which scenarios you're going to use for your particular analysis that you're looking for. And I know we've kind of alluded to that a little bit before. Like what scenarios do you use for maximum hypothetical accident? If you're going to use the maximum hypothetical accident, should you analyze radiological consequences for all of your accidents and provide that as your safety analysis? So, the determination there is a part of the basis for determining which source terms you need to develop.

When you go through determining which phenomena you need to model, and which systems, structures and components, you would credit in these analyses, there is a balance of prevention versus mitigation of the event. So, certain SSCs are safety-related or must be managed in a different way because they prevent the accidents from occurring. Then, other SSCs are there to reduce the radionuclide release or increase the containment of radionuclides. And so, they have special treatment as well.

There's a relationship in determining source terms to the functional containment concept, and the functional containment, as taken from the SECY paper, is a barrier or set of barriers taken together that effectively limit the physical transport of radioactive material to the environment. And that's in place of like a leak-tight physical structure necessarily being required.

There's also relationship to PRA, the decisions that you would make with a PRA. You have to develop source terms for some of those decisions. Like if you were to use the non-light water reactor PRA standard, it goes all the way to consequence. You're not making decisions based on proxy metrics such as CDF or LERF. And, of course, in all of this, you need to have a good determination of uncertainty around the physical phenomena and the amount of release and the likelihood of the event.

Next slide, please.

So, as I had just kind of discussed, there's no requirement that you have a mechanistic or deterministic valuation. That's a choice that a designer can make. If you do use the Licensing Modernization Project, that assumes that you would develop a mechanistic source term and use a PRA.

However, there are some non-light water reactors that may choose to provide a postulated maximum hypothetical accident, similar to non-power reactor licensees. The staff acknowledges that there's no specific guidance on development and review of non-light water reactor source terms or mechanistic source terms.

However, there is useful information available. Reg. Guide 1.183 has a section, "Attributes of an Acceptable AST," that may be useful, and it describes what attributes are expected for an acceptable alternative source term for light water reactors. Though some aspects of the discussion may not be applicable to some non-light water reactor designs, the guidance is mostly general in nature and should help them be able to address regulatory requirements, if they do look at that.

I also note that the guidance in Reg. Guide 1.183 on radiological assessment, in other words, the portions that do not describe the light water reactors source term itself, such as guidance on dose conversion factors, briefing rates, atmospheric dispersion, things like that, is technology-inclusive in manner, and non-light water reactor applicants may find it useful in performance consequence analyses. And we are planning to clarify that in the revision to Reg. Guide 1.183.

SECY-93-092, which is a discussion about issues pertaining to advanced reactor designs and the relationship to current regulatory requirements, included staff recommendations on non-light water reactor source terms. I'm going to discuss these recommendations starting on the next slide.

If there are any questions?

(No response.)

If not, next slide, please. It should be slide 110.

So, this is the SECY-93-092 definition of mechanistic source term that the staff put forward. "A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from specific accident sequences being evaluated. It's developed using best-estimate phenomenological models of the transport of fission products from the fuel to the reactor coolant system for all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs."

Applicants can choose to provide either a mechanistic scenario -- I'm sorry. In the SECY, we recommended that advanced reactor source terms should be based upon a mechanistic analysis, and based on the staff's assurance that certain provisions are met. The SECY defined a mechanistic source term in this manner. As you can see, this definition has a clear relationship to the discussion of functional containment and the barrier assessment that you would do there.

Next slide, please.

So, the SECY paper also provided these provisions for staff assurance the mechanistic source term was acceptable, or that they were considered.

And the first thing is that you should sufficiently well understand the performance of the reactor and fuel under normal and off-normal conditions, so that you can do a mechanistic analysis; and that you should have sufficient data to provide adequate confidence in the mechanistic approach.

Transport of fission products can be adequately modeled for all barriers and pathways to the environs; and that the calculation should be as realistic as possible, so that the values and limitations of any mechanism or barrier are not obscured.

And that events considered in the analysis to develop the set of source terms for each design are selected to balance severe accidents and design-dependent uncertainties.

The SECY went on to say that design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

Next slide, please.

So, to go on now to more recent information, we contracted with the National Labs to address accident source terms for non-light water reactors as part of the NRC's non-light water reactor vision and strategy near-term implementation plans, and to respond to the Nuclear Energy Innovation and Modernization Act, or NEIMA. This resulted in these two recent reports that I'll describe now -- one from Idaho National Lab and the other from Sandia.

They are technology-inclusive and they tell what to do to develop accident source terms, and not specifically on how to do it. There are no specific methods or phenomenological models that are presented as "the way to do things." And they do not provide technology-related source terms or releases. In other words, they don't provide specific values that you would expect to see.

Next slide, please.

The first report is the one from Idaho National Lab. It's titled, "Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities." And it summarizes a risk-informed, performance-based, and technology-inclusive approach to determine source terms. This report was issued in June of 2020, and there is the ADAMS Report Number, Accession Number.

It's a graded process that allows both the non-mechanistic source term calculation methods which adopt conservative approaches and assumptions based on known physical and chemical principles, and more importantly, the mechanistic source term calculation methods, which consider design-specific scenarios and use best-estimate models with uncertainty quantification for a range of licensing basis events to be used for the design and licensing of advanced nuclear technologies.

Accident source terms are developed to address licensing issues to support the application processes of 10 CFR Part 50 for a construction permit and operating license or 10 CFR Part 52 for combined operating license, standard design certification, early site permit, or standard design approval, or manufacturing license. They can also be used for other purposes, including equipment and environmental qualification, control room habitability analysis, of assessments of severe accident risks and Environmental Impact Statements.

Next slide, please.

This is taken from page 6 of the INL report. It's a picture of the mechanistic source term as being developed through a systematic evaluation of transport through barriers to the radiological release to the environment. It's a picture that they took, reproduced from a Sandia report, which is the next report I'll be talking about.

As you can see, it looks at each of the barriers, and once you figure out what the release is from each barrier and the retention in that barrier, it would result in the release to the environment.

Next slide, please.

This is another figure that is taken from the INL report. And it shows all of the different considerations and information that you would use to develop source terms to compare to regulatory criteria. And I know it's a nice picture, and it is described in more detail, or the concepts are described in more detail in the paper.

Next slide, please.

So, the INL report describes some steps in the methodology to develop mechanistic source term. The steps to developing a complete mechanistic source term were developed with the Licensing Modernization Project in mind. So, it includes determination of licensing basis events, as the term is used in NEI 18-04.

The steps allow for flexibility to accommodate refinements in the source term determination, as needed for the specific purpose, including allowing for non-mechanistic or simplified mechanistic approach.

One approach is to use the initial bounding calculations from step 4 to meet requirements. If they meet requirements that are sufficient for your licensing purpose, that should be sufficient.

This is intended for facilities that have a small enough initial inventory of source terms to meet radiological requirements upon a full release of the initial inventory. So, we don't expect it's something that the majority of designs may be able to use.

The second pathway can use the System Hazard Analysis, or SHA, and perform simplified calculations, as noted in step 5, to identify barriers and a maximum fractional release to perform a simplified mechanistic bounding analysis that could meet radiological control requirements -- or radiological consequence requirements. Excuse me.

A third pathway, which is still not a full mechanistic source term approach, is to use the loop of redesign in step 6, after going through step 5, and following through back again to step 4, and then, back to step 5, to continue to refine your analysis. If these pathways are not sufficient, a complete mechanistic source term approach going all the way through to, well, 14 -- you should always document your source term development -- would be included.

So, as you can see from these steps, there is a set of systematic evaluations that you go through and you refine and you make it more detailed, and consider the uncertainty and the design goals, as you go through this assessment.

And I know I went through that really fast, and I don't know that it necessarily made sense. So, do you have any questions?

(No response.)

If not, we'll go to the next slide.

Now I will talk about the lab report that we've gotten, which is a "Simplified Approach for Scoping Assessment of Non-Light Water Reactor Source Terms." And this was issued by Sandia in January of 2020. And that's the ADAMS Accession Number for that.

This approach is intended to identify the characteristics of reactor design concepts, release mitigation strategies that are most important to different classes of accident scenarios. And it uses a scoping methodology to provide an approximate, order-of-magnitude estimate of the radiological release to the environment and associated offsite consequences.

The scoping method is applied to different reactor concepts, considering the performance of barriers to fission product release for these concepts under sample accident scenarios. The accident scenarios and sensitivity evaluations are selected in this report to evaluate the role of different fission product barriers in reducing the source term to the environment and associated offsite consequences.

It did not develop quantitative estimates of radiological release magnitudes and compositions, and I did say it looked at several different concepts. Those concepts were high-temperature gas reactors, sodium fast reactors, and liquid-fueled molten salt reactors. And it is primarily qualitative.

Do you have any questions about this report?

(No response.)

Okay. The next slide. This is slide 118.

And, of course, there's the non-light water reactor PRA standard, which was just issued last year. This standard describes a full-scope PRA, which includes consequence analysis, and there's a mechanistic source term analysis element, or MS, which provides useful information on what to do to develop mechanistic source terms.

MS describes the objectives and the characteristics and attributes of the mechanistic source term analysis for the PRA element. It provides information on what to do to develop mechanistic source term, not specific assumptions, methods, models, or computer codes. And it describes such considerations such as timing, location, amount released, and the radionuclide transport barriers and transport mechanisms and associated uncertainties.

If an applicant is using the PRA standard, I think there's good information in there that they could reuse for their safety analysis beyond their PRA.

Are there any questions on this aspect of the PRA standard?

(No response.)

If not, I would like to hand the presentation off to Bill Reckley to talk about the Licensing Modernization Project.

MEMBER REMPE: So, I didn't realize you weren't going to be the next presenter, Michelle. So, I do have a question.

MS. HART: Yes. Sure.

MEMBER REMPE: Are you planning, is the staff planning to endorse the INL or Sandia reports, or something?

MS. HART: So, we haven't decided if we're going to endorse them or not. Right now, we're putting them out there as information that is useful and could be used by a designer to help develop their source terms.

So, they weren't written in the sense that they would be like a Regulatory Guide on acceptable analyses to do a siting analysis, or anything like that. They weren't written in that manner.

MEMBER REMPE: It just seems like you're putting it out there, but it's not endorsed, or there's no NUREG to say these are different approaches that are acceptable to the staff. But if somebody goes and follows that approach, and says, "Well, it was on the website as something good to do, we thought," and then, it's "No, no, it doesn't work." So, I guess I'm surprised that they're not asking for some sort of additional guidance. But, again, if they're happy, I guess they're happy.

MS. HART: Well, I think the thing with these, both of those reports, is that, much like with the PRA standard, there's a lot to be left to the implementation. It doesn't give specific models. It's doesn't say, you know, "This is the way to do things." It says, "These are the considerations that you should use when you're developing a methodology."

And so, I think it's like a pre-step to you determining how you're going to do that. And then, you could come talk to us and say, "This is what we're planning." And we're saying, okay, yes, if you use this information from these guides, we understand where you're coming from.

MEMBER BROWN: How does that help an applicant? I mean, to me, it would be just confusing. I wouldn't know what would be acceptable and what would not be acceptable.

This is Charlie Brown.

MS. HART: Right. I think --

MEMBER BROWN: You've gone through a whole mess of possibilities, and, I mean, I'm confused. I apologize.

MS. HART: So, they all have -- I'm sorry -- they all have similar features. They all talk about considering the same things. They all talk about, you know, you need to quantify these things -- "these things" being, you know, whatever they are.

I think there are some issues, but there are so many different technologies; there are so many different approaches that they can take to licensing -- I mean, like they can determine to do a full mechanistic source term or they can determine to do something in between. Or they can determine to do a maximum hypothetical accident, which has a non-physical, obviously, bounding analysis, where they wouldn't have to do as much, I guess, justification of the assumptions that they had done, or detailed analysis to supply the justification, is probably a better way to say it.

So, I think we're trying to remain flexible for these different approaches for these different designs and different goals that these applicants would have.

MEMBER BROWN: Well, one of them is qualitative. Some of the other ones are more quantitative.

MS. HART: Uh-hum.

MEMBER BROWN: It just seems --

MEMBER HALNON: So, this is Greg. Let me try to frame it just a little bit.

These small reactors are technologies that are usually -- I'll just say they're probably commercial reactors or commercial facilities that will be out there trying to make a profit. And in that, there's going to be a tremendous economic pressure to have extremely certain design parameters.

And when you get into these situations -- and I'm not sure if there's an answer to it or not -- where you get this sort of "bring me a rock, and we'll let you know if we think it's okay," without that bright line to design to, there's inefficiencies and there's potential economic issues.

So, there's going to be this drive to push back. And I would hope that at some point we would get, after enough technologies have come in, that we could see the similarities in all of these and come up with a process that gives more certainty to the designers. That's the pressure, and there is frustration amongst the designers because we don't have, you know, how do you slice and dice the licensing basis events, you know, get rid of the 10-to-the-minus-12 events and talk to 10-to-the-minus-5 events, and that sort of thing?

So, that's the frustration I think that at least I'm experiencing through this. It came out in droves when we were doing the 50.160 P rule. And I think you're going through the same type of discussions with NuScale.

MS. HART: Right. I think, you know, this transition period we're in -- or I don't even know that that's the right term -- but, yes, I mean, it's hard to be efficient right out of the gate. We're trying to be as flexible as possible and allow for different concepts to come in. So, that does leave some room for future refinements. And if we do see, as we do see commonalities and obvious -- well, I won't even say, "obvious." As we learn lessons as we go through, we may find in the future that there is some guidance that we can give that could help refine things and bring things into a more efficient space.

MEMBER HALNON: Well, I think it's essential that we look for them and we have a project in play that every time we do find one, that we can refine the guidance, so that it gets more certain, we should do that. We shouldn't wait five years, and say, okay, now what did we learn? Each time we find something that's going to be advantageous to the certainty of a designer, we should allow that to be used.

And I think the nature of Topical Reports, and being able to use them across the board, is good, except I keep on hearing, you know, this is very specific to NuScale, or very specific to Kairos, or it's very specific, and it causes a problem in the transposition of these types of methodologies to other reactors.

MR. SEGALA: This is John Segala with the staff in NRR.

I just wanted to add on the question that Joy had on the endorsement, I did want to add that, for the non-light water reactor PRA standard, we're getting very close to putting out a trial use Reg. Guide endorsing that standard. So, yes, we're not endorsing the other two industry guidances, industry documents -- I'm sorry -- the National Lab documents that Michelle just presented on. Those are really just being provided for information to help the developers.

MEMBER REMPE: I get it, but it would be nice if -- again, it's their bailiwick. If they're not asking for it, I guess all you can do is say that and be very careful to say, "These are just out here for information. Use it. It doesn't mean that we're going to approve it." But, anyway, that's what I'm hearing.

MEMBER BROWN: Well, in my mind, this is similar -- one of the concerns I've had with Part 53, it's more generalized requirements as opposed to general design criteria. And if somebody submits that, then, all of a sudden, the NRC decides, well, that's not quite good enough.

And the source term issue, it seems to me, is going in the same direction: "Tell us what you'd like to do and we'll figure out whether it's okay or not." Flexibility is great, but that's really tough to design hardware, plants, anything to get that work done. It's almost like nobody wants to have any specific requirements at all. I'm exaggerating slightly, but that seems to be, if you're listening to these slides, it seems to be where we're going. I'm glad I'm not a designer.

So, I'll pass. I just think this is a concern. I've spoken my piece.

CHAIRMAN PETTI: So, to me, my bigger concern is the diffused nature of what's called, quote, "requirements." They're there, but you've got to go read six, seven, or eight documents -- and it's only the source term, the most important thing we care about in terms of protecting the public.

And that's why the idea of consolidating it, you know, a website, whatever, so that someone doesn't overlook something, (a) I think is good, but (b) it would provide confidence, right, that, yes, the NRC is doing its job and doing it well, and here's what it looks like.

I would think NEI would be interested in trying to put out something. Yes, it may be functional, it may be higher level, because it goes across technologies, but, as it is now, it is somewhat diffuse. And this has to do with the NPUFs versus power reactors. It has to do with the history of how, what's called the "source term tree" has grown, but it seems like it's worth stepping back and asking these sorts of questions at this juncture, given what the future may hold in terms of advanced reactor applications coming in.

Keep going, Michelle.

MS. HART: I'm sorry. Okay.

So, I will be handing off the presentation to Bill Reckley now.

MR. RECKLEY: Okay. Thanks, Michelle.

So, thank you, everyone.

And I only have a few slides, but I'll go back and visit a few of the comments, and not disagreeing with anything anybody has said.

Part of the challenge is, as you provide flexibility and try to make things technology-inclusive, addressing a variety of reactor technologies, it gets harder to be specific and produce anything like a TID document or even NUREG-1465 and say, "Use this source term and it's going to be used as a confirmation of a specific barrier like the primary containment for a BWR or PWR."

But, that said, just to kind of summarize licensing modernization -- and we've talked to the

Committee and this Subcommittee a few times on this -- it provides a risk-informed approach to the selection of licensing basis events and categorizes them into the categories we've talked about -- anticipated operational occurrences, design basis events, beyond design basis events, and then, another category of design basis accidents that are used kind of in a traditional deterministic method.

Those event categories are assessed and compared to acceptance target figures to test specific barriers and to assess the margin to the frequency consequence target and design objectives, such as a lower-dose value to justify an Emergency Planning Zone, as Michelle talked about under the 50.160 rulemaking.

And this also includes an assessment against cumulative risks, using the QHOs and some other measures introduced by the NEI 18-04.

And then, just a summary that there are key roles in the LMP, as we've talked about before, for probabilistic risk assessment and, also, mechanistic source term. So, I don't know.

If we can go to 120?

This is a figure similar to what Michelle showed from the lab reports, and that's not a coincidence. We were all starting from the same fundamental source document, which is a DOE order on assessing radiological releases from any facility. And so, it's simply modeled as an inventory, and then, various barriers, and then, the integrity or the performance of those barriers for particular scenarios as it relates to particular radionuclides, and as it relates to time, as the transients are being modeled.

So, Michelle had the same equation. Obviously, we don't have it down yet to being able to model things using an equation like this. But you can, if you go back to Hossein's presentation and Jason's presentation, you can see that these things are what's being modeled in things like MELCOR, the transient response of particular barriers and the transport of radionuclides against those barriers.

So, one way to look at this, for example, when Michelle talked about the Sandia report, in looking at a first systematic way to look at it, it would be, for a particular design, where am I going to put the emphasis on retaining fission products? So, under the current light water reactor model, a lot of emphasis is put on barrier four, the containment building. Under HTGRs, there's a much larger focus on barrier one, the fuel, and maybe barrier two, the matrix around the particles. In other cases, or for other transient scenarios, since mechanistic source term is done on a scenario basis, it might be the pressure boundary or some other consideration.

And I know it's complicated. And I have said, when it comes to helping the developers, if they don't know it's complicated, they're getting into the wrong business. They have to understand that, when they're given these design choices and flexibility, then they have responsibilities.

So, when you go to how they implement this maze, if you will, let's say they are going to emphasize barrier one in their design. That's where they want to put the emphasis.

Then, Arlon, if you could do me a favor and go back up to 115? Okay.

This is another complicated figure, but what does it mean? It means, where are they going to have to do their research? Where are they going to have to put their emphasis? What does the R&D have to go? Where are they going to try to minimize the uncertainties in any particular barrier? If they're putting the emphasis on the TRISO particle, there's got to be a lot of work by DOE and by the developers to show that they've addressed that qualification of the TRISO fuel. On the other hand, should they choose to say, "I'm going to put my emphasis on another barrier," then, they're going to have to do the R&D on that other barrier.

So, again, it is complicated and it's iterative, how a designer is going to have to walk through this. But, again, given that we're trying to address a lot of different technologies, and we're trying to give flexibility to the designers as to where they put the emphasis in their design and in their analysis, it's just the nature of the beast that it won't be as clear as giving them a spreadsheet with radionuclides and saying, "Here, put this into this volume and model your containment."

And I oversimplified. I don't mean to -- even the current process is not as simple as that, but, anyway --

So, Joy, I see you've got your hand up.

MEMBER REMPE: Yes. You're right, and what you just said would be good to have in a document somewhere.

CHAIRMAN PETTI: Amen, amen, amen.

MEMBER REMPE: The other thing I have to say is --

CHAIRMAN PETTI: I completely agree.

Bill, it was good.

MEMBER REMPE: Let me finish, Dave. Let me finish, Dave. Okay?

CHAIRMAN PETTI: Okay.

MEMBER REMPE: Old farts like me can recognize this equation really comes from NUREG-1150. This is not something new. But perhaps the new design developers are not old farts like myself. Okay?

(Laughter.)

So, yes, and what you're saying is very good, Bill, and as Dave is saying, "Amen," and singing a chorus here. It ought to be written somewhere. Is it written somewhere?

MR. RECKLEY: I think the reports that we commissioned talk about it in this sense, when you look at them. And, for example, the Idaho report, then, goes on to further say things like Michelle was describing. You have an option in there at some point to say, "I'm going to address the uncertainty by being conservative. There's things I don't know in terms of the behavior of the radionuclide, or the behavior of a barrier. I'll just be conservative," and that's an option that's available to the designer as well to get through the licensing process.

So, whether it's as clear as it should be, I'm not sure. As John was saying, one thing we can do -- or even some of the members were saying -- bring this up at a public meeting and ask the developers, who are the ones we're trying to do this for, is it clear; is this understandable? Between the website, the reports, the history, is it clear? I don't know.

MEMBER REMPE: Okay. I think you've got our point. But Dave may want to expand --

CHAIRMAN PETTI: No, no, I agree. That's why I keep thinking of this, sort of an executive summary that pulls all this together. I'm thinking three-, four-, five-page manual, heavily annotated. So, it's kind of a roadmap, but the important things get put there, so that it helps them put all these different documents in context.

Because, you know, it is diffused right now, and I'm just trying to figure out, is there a way to better focus it, just to help them? I agree with you that you still need the flexibility, and it's never going to be as cut and dry as "Do X, and then, do Y, and then, do Z." But if we can make sure that they know what the considerations are in each step, and what's important, that's, I think, the best you can do at this point.

MR. RECKLEY: Okay. Well, I guess -- and

John will pick this up maybe at the end with the website -- that's a step; that's our effort to take a step in that direction. And you guys haven't had a chance to look at it yet, and neither have the external stakeholders. Maybe the next step is to see if that provided additional clarity where we're seeking it.

So, Arlon, if you can go back up to 120?

MEMBER REMPE: While you're going up there --

MR. RECKLEY: Yes?

MEMBER REMPE: -- again, I'm thinking about websites and how they can easily be changed. And it's nice to have the website, but you always have to say, "Last accessed," and what date you did, because --

MR. RECKLEY: Yes.

MEMBER REMPE: -- it can easily be changed. And so, again, I think something that's a little more concrete might be a good idea.

MR. RECKLEY: Okay.

MR. SEGALA: Yes, this is John Segala, Bill.

I just wanted to add, I mean, now that we made the website, the web page on source term, publicly available just earlier this week, you know, we were planning to share that at an upcoming stakeholder meeting, just to make everybody aware that we have this new page. So, just throwing that out there.

MR. RECKLEY: Okay. And, Arlon, if we can go to 121?

It's basically a very similar representation, just coming at it from a different direction. And as I just mentioned, and the bullet emphasizes here, there is flexibility provided on how they want to develop the safety case, both to reflect the design and within the analysis.

And that involves combinations of active, engineered safety features, passive safety features, and increasingly, reliance on inherent properties of materials in the reactor cores. And that's consistent, as it says here in the second bullet, with the Advanced Reactor Policy Statement encouraging passive and inherent features.

But sometimes, those also bring in uncertainties and things you need to address in both the design and in the analysis.

So, that's just a quick summary, again, of licensing modernization; the key role that modeling the mechanistic source term has in that.

So, with that, if there's no other questions or suggestions, we can go on to fuel qualifications and Tim.

MEMBER KIRCHNER: Dave, may I make an observation? This is Walt Kirchner.

CHAIRMAN PETTI: Sure.

MEMBER KIRCHNER: You know, Dave, and going back to this morning, we have this DG -- let me make sure I get the number right -- DG-1199. Is that an appropriate place to at least lay out this generic approach to source term analysis, identification and analysis?

CHAIRMAN PETTI: I think that's RWR-based.

MEMBER KIRCHNER: I know it is, but, conceptually, it seems to me, at least what I do is, each time we see a new document for non-LWRs, my first proof of concept or test of the document is, would it work for an advanced LWR? And I think they will, most of them.

And the advanced LWR, of course, is where we've got the most experience. We've got very mature PRAs. We've got equipment reliability to feed into the PRAs. We've got a good understanding of the technology and vulnerabilities, et cetera, et cetera.

So, you know, in my mind, when we look at these new approaches, I'm testing them on an advanced LWR, and I think they will work. So, conceptually, the charts we were just looking at on mechanistic source term, I mean, those could be used for an advanced LWR as well.

I thought what Michelle was covering, that synopsis of the methodology from the INL report was useful because you started off trying to bound the problem, and then, you worked further down. If you've got the data, if you've got the experimental data, if you've got the experience with the technology, you can probably go down further. As Bill was saying, this is complicated, especially for newer technologies where you don't have the kind of databases, and such, and experience.

It just seems to me that this DG-1199 might be a good place to put the conceptual approach in, in a technology-inclusive manner, and then, go on to --

CHAIRMAN PETTI: My view is, if there was a place that it needed to be put, it would be in an appendix to the NUREG on fuel qualification that Tim's going to talk about. Sorry, Tim. To me, that was where it made most sense. But I would leave that up to the staff to decide where to put it. But there's a lot of --

MEMBER KIRCHNER: But that's just fuel. To do the source term --

CHAIRMAN PETTI: No, the source term is --

MEMBER KIRCHNER: -- you've got to have the barriers.

CHAIRMAN PETTI: It's talked about in there. You'll hear it. You'll hear it from him. That stuff is in there. But, again, I'd leave those details sort of to the staff, but --

MEMBER KIRCHNER: That was just a comment. No need -- no answer required.

MR. DRZEWIECKI: Okay. So, this is Tim Drzewiecki from the staff.

Sorry, can you guys hear me?

CHAIRMAN PETTI: Yes, go ahead.

MR. DRZEWIECKI: Okay. Wonderful. Okay. Yes. So, yes.

So, I've got a few slides to just kind of give an idea of what's in this document. This is the fuel qualification guidance that we came out with and we talked about at the end of last year.

And what it kind of lays out is a top-down method to kind of highlight a list of goals or criteria the staff could evaluate to say that this fuel is qualified and it does have certain aspects that I'm going to touch on source term. So, I'll kind of walk through how it relates to the source term in the next couple of slides.

So, may I have the next slide, please?

So, how it starts out is we're trying to make a finding that the fuel is qualified for use. And there are two kind of conceptual things that we want to make findings on for that. One, that you have the manufacturing specification to control the key fabrication parameters, and the other puts that the safety criteria can be specified. Obviously, the safety criteria are not well-defined. So, we break that down more on the next slide.

So, the next slide, please.

And in here, specific criteria are coming from the NRC regulations. The box on the left, that's talking about margin on the safety limits under normal operation and the AOOs. And the one in the middle, which is the focus of this presentation, is showing that you can maintain the margin of a radionuclide released when it's under accident conditions.

Now the point is, for this report, we were focused on the fuels role, but it does talk about other barriers, too. And this goes into the role that fuel plays in the protection against the release of radionuclides.

And the box on the right, that just goes into, basically, showing that you can maintain a safe shutdown condition.

Next slide, please.

Okay. And so, that box that was in the middle, it's broken down into four other criteria. I want to say that the boxes that are gray are gray because they're considered these base goals, in the sense that we would expect to be able to have enough evidence to make a finding on that item and not have to break it down any more.

So, in terms of the box on the left, we want to make that the fuel performance envelope is defined. In other words, for this goal, it's knowing what kind of accidents that the fuel is going to be credited to perform under. So, whether it's under-cooling events, overpower events, things like that.

The box second to the left is, basically, specifying what's the radionuclide retention requirements of the fuel under accident conditions. This is going into the role of fuel as it plays in the safety case, in the sense that, if you're not crediting your fuel, if you have other barriers, that's going to have an impact on the kind of data that you would need to support that.

These other two boxes on the right, these, I think, line up pretty well with the source term, a slide that was shown on -- both Bill and Michelle had shown it. So, these are associated with, one, the barriers to the release of radionuclides, showing that you're conservative on criteria there. So, this should be things like propellant/clad/mechanical interaction, your limits, or it would be other things, too, like whether it's, say, like melting, and things like that.

And then, the box on the right, that's talking about the release of radionuclides or the migration of radionuclides within the fuel matrix.

So, I was going to break down the box, that whole G2.2.2 on the next slide. Yes.

Essentially, this has two boxes. One is that you can show the criteria are conservative, which we think is something that can be shown, as long as you have data. This assumes that you're comparing it against experimental data.

Then, the box on the right is just making a statement that experimental data is appropriate.

And that box is blue because there's a whole separate framework that's going to break that down. And that goes into things like data independence. Is that data applicable to the scenario, things like that?

And so, what I have on this next slide -- may I have the next slide, please? -- so, this is not really meant to be read, but it kind of shows you what this framework is. It takes a high-level goal and it breaks it down into, effectively, like a checklist, a bunch of things that we want to make findings on. And if we can make findings on all of those gray boxes, then we can say that the fuel is qualified.

And there is some overlap here. And what I mean by that is the main framework is the one on the left, and that relies on things like having evaluation models in a couple of sections and experimental data is used to support multiple goals as well.

So, that's all that I had for NUREG-2246, unless there's any questions.

(No response.)

Arlon, can you hear me? Okay.

MR. COSTA: Yes, we can hear you.

MS. HART: I guess if there are no questions, we can go to the next slide.

So, I wanted to talk a little bit about this web page that we keep referring to, the non-light water reactor accident source term web page information. At the top, there's the link to it. It's on the public website and it's underneath the advanced Reactor-related documents.

And it's a one-stop shop for existing information, and it's discussion of accident source terms. There's a link list of documents relevant to development of non-light water reactor accident source terms for licensing. It includes some of the light water reactor information. We're not trying to be comprehensive. Right now, we are putting information out there and we will keep it up-to-date and keep adding to it, as we continue to go through this process.

Arlon, if you could open the website, so they can take a look at it?

So, we have a little bit of a preamble that talks about what a source term is and a little picture of the barriers and kind of the barriers assessment.

And next, we go through the history and evolution of the light water reactors source term. So, you have some of that history.

And then, it kind of follows along the lines of what we've been talking about today. We do have the links in the text here to TID-14844 and Part 100, Reg. Guide 1.183.

We talk a little bit about the analytical tools and past studies. So, we have references to RADTRAD and MELCOR.

And then, we talk about information on SMRs and non-light water reactors. So, we do have a discussion to our mechanistic source term and how it interacts with the regulatory analysis and the siting analysis that we talked about in SECY paper 16-0012.

And then, we talk about the SECY-93-092, which was kind of the base for the information on what a mechanistic source term is and the considerations for what the staff would look at for a good mechanistic source term.

And then, further down on the page, we have a section where we would talk about accident consequence-related regulation activities. We don't have links to those right now at the moment, but it says that there are things that we're going through now.

And then, we have the guidance and information for developing advanced reactor source term with reference to the LMP, is the first thing, the Regulatory Guide that endorses the LMP.

And then, we go through document links. And so, you can see that we have a link to TID-14844, NUREG-1465, Reg. Guide 1.183. And then, Hossein's discussion about the history of source terms.

We have a link to our vision and strategy code development plans for severe accident analyses.

We have some links to those demonstration projects that Hossein and Jason discussed earlier today. So, we do have the slides and a video recording of those presentations from the workshops available on this site.

A little bit further down, another link to SECY-94-302, which is about source term-related, you know, the information on source terms for evolutionary and passive light water reactors. So, it's not non-light water reactors, but it's that kind of information.

And we have the wrong SRP here. It should be 15.0.3 for new reactors. We can fix that.

We do have a link to the approved NuScale Topical Report for Accident Source Term Methodologies. So, if somebody wanted to see an example of an approved methodology for light water reactors SMRs, and we do have the SECY, our staff approach to evaluating the source terms for NuScale.

And then, here, we have some SECYs and some NUREGs and other information on the source term approach for non-light water reactors, including SECY-93-092 and it's SMR; some pre-application Safety Evaluation Reports for several of the designs, and then, an assessment of a white paper from NGNP on fuel qualification and mechanistic source terms. So, that's our staff assessment.

In the next section, we have some guidance on developing advanced reactor source term. And so, for that, we have the endorsement Reg. Guide for NEI 18-04.

We do have the Reg. Guide for the performance-based emergency preparedness for small modular reactors and other new technologies.

And then, we have the lab reports that I described today and a link to the ANS standard, which, of course, we are working on an endorsement Reg. Guide for that as well.

And then, the next section has several reports from labs and from other places about source term information on designs that we've seen in the past.

And then, the next section is presentations on non-light water reactor source term, and I expect that we will probably be putting this presentation and transcript, a link to that here as well.

Like I said, this is what we put so far. And we do intend on keeping that up-to-date. And I do like the idea that Joy was talking about, that maybe we need to have a "last updated" note on there.

CHAIRMAN PETTI: Yes, the other thing you could do is, if something's new, sometimes like websites put a little tag, "New" next to it.

MS. HART: Uh-hum.

CHAIRMAN PETTI: So, it draws the eye of people visiting it.

MS. HART: Yes.

CHAIRMAN PETTI: I'm just wondering, nothing is there about NUREG-1537 for the NPUFs, and that might be useful, to consider bringing in some of those documents, too.

MS. HART: Right. And I think that's a good piece of information. I think there's also been some discussion about maybe some information that they had used when they developed the source terms for like SHINE, or whatever. We can look into what information we can include on that.

CHAIRMAN PETTI: Right, right. I think that would be good.

And, I mean, I like that introduction stuff. That's what I was kind of thinking of as needed. So, I think it's a good flow of all that.

MS. HART: Yes. And it's a very high level right at the moment. As we go through learning things again, you know, and as we get some comments from folks as they look at it, not that we're requesting comments officially, or anything like that, but if there's missing pieces of information or information that doesn't make sense, certainly, we'll take that comment into consideration.

MEMBER REMPE: The intro is good, but some of the things that Bill mentioned about where you emphasize and the barrier diagram might be good to consider --

MS. HART: Uh-hum.

MEMBER REMPE: -- in the actual formation, the introduction. But, again, that's just one member's comment.

MS. HART: Sure.

MEMBER REMPE: And we're getting into designing your web page, which I don't think ACRS members should be doing. But it's just something to think about.

MS. HART: Right. No. No, I understand. You know, there could be more pictures and more discussion. Or they could be more callouts to other documents, as necessary.

So, as we go through this, and if we hear from industry or anybody about what they think is useful, we can take that into consideration. I can't make any promises necessarily, because we're not asking for guidance on how to do our website necessarily, but we thought that this would be a good way to gather this information. And it may be a little bit more flexible than trying to write like a series of white papers or SECY papers, or something, which takes a lot of time and effort to try to get all the language correct.

MEMBER REMPE: And I do want to say, I were a developer, having all the documents in an accessible place would help big time, and sharing your knowledge this way is good. It's just it's very fluid, I guess. I'm still thinking that some sort of paper document might be good to be thinking about.

MS. HART: Yes, we can also think about this as we go through things. I will say, you know, a certain amount of what is on this web page is constrained by the way that the NRC website is generally configured. So, there may be some of these good ideas that you have that maybe aren't implementable or generally haven't been implemented on our website. So, I'll just add that as a thought to throw out there.

MEMBER REMPE: Good point.

MS. HART: Are there any other questions?

DR. BLEY: Michelle, yes, this is Dennis Bley.

After you introduced it, I went over and found it and have been poking around in it. And I think it's a great start. And it's the kind of thing I think the Committee has been looking for for a while, and I think it will be very helpful to people. Thanks.

MS. HART: Right. And somebody did tell me that there is a timestamp on the page near the bottom that says the last time it was updated. So, it's not in big print or anything like that, but it does have it.

MEMBER REMPE: It is on there. But I guess I was thinking, if a person came in and said, "Well, I used document X that was available on January

15th, 2021," or 2022, and in the ensuing two years before they came in, that document has been removed because the NRC has decided that wasn't a good idea to post it, they're kind of stuck. If you say, I used Reg. Guide 1.-whatever that was Rev. 3, the fact you've updated it, well, it was there at that time and there's more certainty with the document.

MS. HART: Right. I think, right now, we only have things that already existed. We haven't added new, different information, other than this introduction/preamble stuff. So, if you're using a link that's on the page, you should be able to find that again later, if you've taken down the information. Most of them are ADAMS documents; not all of them are, though.

And, of course, there are some pluses and minuses to that as well. I'm thinking of like specifically Reg. Guide 1.183. Right now, we have a link to Rev. 0. What happens when we have Rev. 1? We have to make sure that that's up-to-date.

MEMBER REMPE: And again, these are just comments, and you've emphasized ACRS members shouldn't be designing a website. But it's just, you know, we can't help ourselves sometimes.

MS. HART: No, I get it.

Are there any other questions or concerns?

(No response.)

All right. If not --

DR. BLEY: Well, yes. Now that you mention that there are constraints on NRC websites, I'm reminded of the briefings we've had on NRR's Venture Studios, and maybe they can be of help to you guys. And maybe you've already used them to make this more flexible and easier to use, and flexible in areas where you're looking for more flexibility.

MS. HART: All right. Thank you.

So, if there are no further questions, I guess we can move on to the next topic, and that will go to Bill Reckley.

MR. RECKLEY: Okay. Thanks, Michelle.

And I think the only reason I'm doing this is because I was gone in January. So, I stepped forward or everyone stepped backwards, or however that works.

So, just kind of summarizing things -- and I'll ask Michelle or John to jump in whenever it's appropriate -- but if we want to go to 131, Arlon?

Just kind of, as a summary and a general approach, using that same kind of complicated figure that Michelle used, but it does emphasize the level of effort that needs to go into this.

And we understand the general observation that reactor developers, and especially those that may be somewhat newer to this kind of interaction with the NRC, it's complicated and maybe it could be made simpler. On the other hand, this is really what it comes down to, right? I mean, this is, as I think Mark said in the very beginning, this is why the NRC is here, and that's to address the potential hazards of nuclear reactors and radioactive materials, and ensure that they aren't released to the environment.

And so, as you look at how that evolved for light water reactors since the 1950s or the TID in 1962 to present day, that reflected the way that it was done, the importance of the containment, all the way back to the genesis of that, as part of the siting decisions of the Atomic Energy Commission. And it's worked over that period of time.

And so, as the first bullet says, if a developer wants to use that kind of an approach, if they're a non-light water reactor, they're going to have to show a conservative source term, and it may not be the same as NUREG-1465 because of the technology differences. But if they want to put an emphasis on a barrier like a containment and use that same approach, that remains available to them.

The other thing that the second bullet starts to go into is they can, also, decide to use a more mechanistic source term and a more LMP-type approach, where you are assessing the performance of barriers and the behavior of radionuclides for particular scenarios and assessing the system overall. That's the functional containment concept. You're, basically, looking at all of the barriers and their performance for various scenarios, as the LMP would allow you to do.

In either case, the third bullet, when you're dealing with a variety of technologies and designs, the actual implementation is going to be specific to that technology and that design. The source term for a molten salt reactor, and what radionuclides stay in the salt and what gets released, and what barrier they're going to rely on, is going to be different than a gas reactor or a sodium fast reactor. And that will go to my next slide to some degree.

The last bullet is the NRC is not currently planning to do the equivalent of a TID assessment or a 1465, where we give source terms to applicants and say, "If you use this, it's acceptable."

As Jason and Hossein mentioned, the work we do, we share, but that's somewhat different than giving them something in a Regulatory Guide and, basically, saying, "Use this as an input." And the reason we're not planning to do that is for a couple of reasons. One, the variety of technologies, and two, a general trend where that kind of scientific work, the burden of doing that is put over onto the applicants.

And if you want to go to 132, Arlon?

Obviously, the Department of Energy and the National Laboratories play a role in this. And so, as you know, the National Labs are organized into various campaigns for different technologies. All the labs associated with taking the lead for a particular technology are looking at the behavior of radionuclides and barriers and materials, and how that plays into source term.

I just threw up a couple of reports. The HTGR, that dates back from the NGNP, Next Generation Nuclear Plant, project from 10 years ago. Argonne and Oak Ridge are doing work on molten salt. Argonne is doing work to develop models for sodium fast reactors.

So, if we go to 133.

The NRC's activities have been more focused on the methodology and how to use that scientific data, or how an applicant might use that scientific data, that they do by their own experiments or through reference to DOE activities, to support their design and to support their safety case, again, going back to the previous discussion of where they're going to put their emphasis. And both of these reports kind of echo the flexibility that designers have and where they might want to put their emphasis.

Joy?

MEMBER REMPE: So, I'm hearing a little bit different from you in the prior slide discussion, which maybe you're right, but I also think that the design developers need to understand that, too. I still think I like the discussion about, if you rely on that barrier, you'd better have more research. That's technology-independent, and I think that to be vetting guidance.

But you said, I believe -- and I'm paraphrasing -- that, because we've got so many technologies coming in and they're not using the standard LWR stuff, the burden is on the new design developer to come up with a source term. That's just the way it's going to be. If you're going to pick a new technology, you've got to realize you've got to have enough money to get the data and the approach for your source term. And maybe that ought to be in some guidance. Or is it communicated well to them, so they understand that?

MR. RECKLEY: Well, yes.

MEMBER REMPE: Am I misquoting you, by the way? Because I am kind of paraphrasing what I'm hearing.

MR. RECKLEY: No, no. I think it's an accurate paraphrase. The only thing -- Arlon, if you can go back to 132? -- is, and I think you guys are aware of this, all of these are complex relationships, right?

The Department of Energy is not an innocent bystander. They are a key player. And they do this through their own work. They do it through working with developers. They solicit input from the developers on where to do the research through things like Project GAIN. They work with individual developers, and some of the Project GAIN grants have related directly to developing source terms.

And we're not just standing by, either. We're interacting with DOE on where to do some of the research. And so, you do have all parties kind of trying to stay cognizant of what each other is doing to make sure this works out. I don't want to make it sound like we're just stepping back. We're talking to the developers, as Michelle and others have said, in this pre-application discussions. Those same developers are talking to DOE. We're talking to DOE. And so, the hope is that, although it is complicated, that it is kind of an organized approach to this.

So, I'll leave it there, unless John or Michelle want to weigh-in on the interactions we have with the developers and the laboratories and DOE. Okay?

MEMBER REMPE: It helps. Again, it's just something I'm thinking about. I'm reacting to things I see in the popular press.

MR. RECKLEY: Right.

MEMBER REMPE: And I'm thinking that some points are worth emphasizing.

MR. RECKLEY: No, and I don't think any of us really have much different thoughts. I mean, in terms of the development of a technology, it's like a 3-dimensional chess game here -- working with the developers, the DOE, the NRC, and other entities. As Hossein was mentioning, you can bring in the international elements. It's a complex set of activities to support any of these technologies.

MEMBER REMPE: Go ahead. Thank you.

MR. RECKLEY: Sure.

So, Arlon, if we want to go to 134?

You know, this is just kind of reinforcing that complexity. This is maybe the most famous simple, well, single figure representing a source term, and it goes back to the NGNP and HTGR discussions.

But, if you look at those DOE reports, you will see similar representations for molten salt reactors, similar representations for sodium-cooled fast reactors. And you can do this for any of the designs and the technologies. And it's just another way to try to represent sort of what was going into the barrier diagram, and just another way to try to represent mechanistic source term; and, again, the complexities of trying to, for individual scenarios, say which of these is going to come into play; which of these are going to contribute? In this particular case, when are the contaminants going to come off of the primary circuit and get released into the reactor building? When might they be retained, and so forth?

So, I didn't really have much of a message, other than just using this somewhat familiar capsule diagram to kind of reinforce the number of things going on and the complexities of developing a source term.

So, go to 135.

This is just, again, summarizing what Jason and Hossein were saying. Our activities in regards to this, developing the models, which we share, for at least the consideration of the developers, whether they're using MELCOR or some other system analysis code to support the design and licensing of their plant.

So, 136.

Again, this is, to some degree, just summarizing the previous discussions, as Michelle went through. We have had both applications and pre-applications with NuScale, along with Kairos, and then, pending discussions with Westinghouse, TerraPower, X-energy. And we'll consider those in terms of a step forward; what is useful to share.

As another developer comes in, we can talk to them about at least the public versions of the interactions with these other vendors, to say, you know, this is one approach that was taken. And if it's an approved Topical, then so much the better; that this approach was found to be acceptable.

Slide 137.

So, moving forward, we, the staff, will continue to follow the scientific work being done by the National Labs and the developers. We continue to engage with those developers on the approaches they're taking in both the design and for presenting a safety case.

We'll continue, as Hossein and Jason mentioned, to develop, refine what we've done to date, and to develop models for the last two technologies we plan to do.

And in terms of the MELCOR workshops, we will engage stakeholders, including the meeting here today, and consider additional guidance, if we get feedback like the last bullet: the web page maybe was useful. But if stakeholders are commenting that it needs to either be revised, we'll take that into account, or we need to do something more, then we'll consider that, based on the feedback we get from our engagement with stakeholders, individual developers, and so forth.

So, I think that's the last slide.

Questions?

MEMBER HALNON: Bill, this is Greg.

Given that last bullet there, maybe an

inclusion on the web page would be a link that emails a certain person that could take those kind of comments. I don't know if that's possible in the public world, but it seems it would be an easy way to get some feedback on it, rather than just wait for public meetings or someone to get so frustrated that they call you. It might be a good way to get some of that feedback.

MR. RECKLEY: All right. We'll put Arlon's name right on there.

MEMBER HALNON: That's what I was thinking.

MR. COSTA: Appreciate that.

DR. BLEY: Hey, Bill, it's Dennis Bley.

I haven't looked up what's on the public web about Part 53 lately, but have you thought about doing anything similar to support Part 53?

MR. RECKLEY: We have quite a bit on Part 53. I think this one might flow a little better than our current Part 53. So, we'll take a look at it. But we do have a lot of information on Part 53 currently on the public website.

DR. BLEY: And I was just thinking about the roadmap, and we've talked about that before.

MR. RECKLEY: Yes.

DR. BLEY: But that might be a place to do that.

MR. RECKLEY: Okay. Thank you.

Hey, John, I'm not sure if you had any additional kind of --

MR. SEGALA: Yes, this is John Segala.

I mean, I appreciate all the discussions and the feedback, and I appreciate the opportunity that we had today to present on all the activities that we've been working on, and the new web page, everything.

I think I'll let you guys conclude and how you want to move forward. But I guess I'm just kind of hearing that some of the things that we're thinking about as sharing the new web page at our upcoming stakeholder meeting in March, you know, reaching out to stakeholders at that meeting and see if there's a need for something else in terms of source term.

It seems like, from what I've heard, that there seems to be a general flavor that there's a lot of good information out there. It's just, you know, there's a lot of information scattered. So, it's hard for a new developer to look at in more of a consolidated manner.

Our attempt with the website was an

attempt to try to pull it all together and make it easy for people to find. And as Michelle and Bill said, we're going to continue to refine that web page and enhance it, and add to it as new information comes out.

So, anyway, again, I appreciate the discussion and even Joy saying that ACRS providing feedback on our websites -- we're open; any sort of feedback for things that we can do better to communicate externally, you know, that's something that we're interested in.

With that, I think that sort of concludes the staff remarks.

MEMBER REMPE: Again, it was just individual member comments. This is a work-in-progress. John, do you think it's appropriate, I mean, knowing that this is a work-in-progress, do you think an ACRS letter is really going to help the staff? You've got our individual comments.

But I heard you say at the beginning of this meeting, "We hear there's going to be a letter." You guys were not asking for a letter. What's your thought on this? I mean, we're doing individual, fly-by-night comments here. Do you want us to be more measured and come up with a letter? Or do you think you've got enough?

MR. SEGALA: Well, I think, from our perspective, we feel like we have enough for developers. We feel that the early-moving developers that were engaging with or developing source term, we haven't really heard any strong interest from the developers through our interactions with them in pre-app, or even in the public meetings, that there's a need for additional guidance.

And so, we kind of feel like we don't really need a letter. We tried to listen to the feedback that you all provided to us at earlier meetings on EP and other things. And we've tried to do all sorts of activities, like develop guidance and develop the MELCOR-SCALE demonstration projects, and make the videos and information available, and the contractor reports.

So, I think what we decided sort of at the end was, you know, that that's where we kind of ended up with a web page. We thought, well, maybe we do need to pull all this information together and make it easy access. And so, that was sort of what we thought was kind of the piece that was needed to kind of pull it all together. And it is a work-in-progress, and we're going to continue to engage with these developers and applicants, as they submit applications and do the reviews.

You know, we've been developing guidance, as you can see from this whole presentation, we've been developing guidance for LWRs since the early '60s, and we continue to develop guidance for LWRs. And I imagine that we will continue to do that for non-light water reactors, as well as SMRs and other new technologies that come forward.

So, I think we're trying to remain flexible. We're trying to remain agile. We're trying to optimize and enhance our guidance, as we move on and as we learn new information.

So, I'm not sure if that answered your question, but that's kind of our view.

MEMBER REMPE: It helps to understand. Again, we do this now with the research folks; that we're having more frequent meetings, which takes more staff time to come and talk to us, and individual members provide comments. And we hear from your personnel that's more helpful to them than having formal letters coming out. And I just wondered where you guys were on it.

Personally -- and again, I'm sure Dave will have us go around -- but I think you guys have done a great job. You've made good progress. But, you know, interacting, we have some ideas. I think it will be interesting to see what the public comments are at the end of this meeting, as well as what you learn in the middle of March, when you go back to them and talk to them.

Anyway, that's just my thoughts.

CHAIRMAN PETTI: Other members?

MEMBER DIMITRIJEVIC: Yes, Dave. Hi. This is Vesna.

I wanted to make one general comment in the end about something which is very important for me, sort of my hangup. And that was not addressed so much in the source data or it was stressed in the beginning of looking. And that's uncertainty associated with results.

So, we saw some of your results where there was a play with the parameters in Monte Carlo and produced some uncertainties. But is just a thought for an ice break. There is some much uncertainties in these results which are associated with models, you know, the radionuclides, barriers, the facilities, the methodology which is used. So, uncertainties are so much bigger than these parametric uncertainties. And maybe there we see results presented with uncertainties, the brackets.

So, I would like, actually, you know, and this is also my personal opinion, but I think that everybody agrees the uncertainties is a necessary part of the game and something starting to be used in regulation. You know, in my opinion, my personal opinion, the uncertainties from the PRA Level 1, I don't know if they necessarily are handled well, but they are much better and astute, and they always address and we talk about mean values and 95 and median and point estimates.

However, as we progress through the Level II and Level III, in my opinion, uncertainties triple. They become a much bigger and much larger, and we don't really see that yet, because these uncertainties are not as well and astute, and they're not looking to in the level of details necessary.

So, I understand that this is just in the beginning and maybe it's too early to want a lot of attention to uncertainty, but I definitely think that that's something we should be very seriously looking to.

All right. That was my comment. You know, I don't think we really understand the uncertainties associated with this and we need to devote much more attention to understand these methods, models, and methodology associated with it.

CHAIRMAN PETTI: Other members?

Oh, go ahead, John. No, John, go ahead, if you want to respond.

MR. SEGALA: Yes, I was just going to say that I think we agree that uncertainties are important. I think modeling and understanding the uncertainties and accounting for the uncertainties are something that's acknowledged in the Licensing Modernization Project, as well as the non-light water reactor PRA standard. I think that that is something that is going to have to be accounted for in the design margins, and whatnot, as we move forward.

So, thank you for the comment.

CHAIRMAN PETTI: Anyone else?

MEMBER KIRCHNER: Dave, this is Walt.

CHAIRMAN PETTI: Yes?

MEMBER KIRCHNER: I misquoted the number of the Draft Reg. Guide earlier. I should have been saying Draft Guide 1389. Is that the topic of your Subcommittee's meeting in March? Are we going to review that document?

CHAIRMAN PETTI: Yes, I think so. That's -- we gave up on 1.138 one -- what it will evolve to, I believe.

MEMBER KIRCHNER: Yes. Well, just a thought and observation. I want to go away and study it more closely. But it seems that, echoing an earlier observation, that the methodology used to develop an alternate source term for an LWR, where we have a lot more data, we have a lot more experience, should, in general, and conceptually, apply for advanced reactors.

And I'll take a look at the Reg. Guide Draft, but it seems to me the LWR specifics could be in appendices and the methodology or the recommended approach, much like was covered earlier from the INL report, could find its way into at least an executive summary or an introduction to the Reg. Guide. I'll go and take a look, and I'll give you some comments perhaps that could be used for that interaction.

CHAIRMAN PETTI: Okay.

MEMBER KIRCHNER: And the other thing, Vesna, I agree with you. Having had some experience in this modeling and simulation business, once you lose the fuel geometry, then the uncertainties in the models are much greater than the parametric uncertainties. You get into stochastic and random kind of results.

So, once you're in that territory, so to speak, once the uncertainties expand, and bounding analyses are probably a wise choice to try and get some sense of the consequences -- this is true of even thermal hydraulics for existing plants. Once you go into two-phase flow and the phases separate, then the uncertainty in the model results increases substantially. And that has nothing to do with parametric uncertainties. It's just the uncertainty in terms of our ability to model what is happening physically. Fuel fragmentation is a good example of that.

MEMBER DIMITRIJEVIC: Yes. You know, Walt, I was thinking, how can this be addressed? I was thinking that, maybe since this is international, there are other programs which do similar maybe computing results, or in the case of Fukushima, we have real data to compare with the results of the models.

I don't know how to estimate that. It's not an easy thing to do. I mean, even I don't think the bounding things always is a way to address something, especially as these become a more important part in the regulation. I mean, how are we going to do the risk-informed applications based on, you know, source term, and the dose and dispersions, and things like that?

So, I'm not sure how to address that, but it's exactly what I thought, what you said. Thanks.

MEMBER KIRCHNER: And, Dave, I would just like to thank the staff for this great presentation today. There's a lot of information here.

CHAIRMAN PETTI: Yes.

MEMBER KIRCHNER: So, I look forward to browsing or grazing on their website.

CHAIRMAN PETTI: Yes.

MEMBER KIRCHNER: Thank you.

CHAIRMAN PETTI: You took my comment. I really want to reiterate what Walt just said. This presentation, you know, the depth and breadth is fairly unique from what we've heard, I think, in other presentations. It required a lot of different folks to come together, and I really appreciate it, given my background, of course, in this area. I've thought about these things as well, and I really do appreciate the slides.

That said, we should discuss, among ourselves I guess, the need for a letter. My letter was going to be --

MEMBER REMPE: Can we do public comments

first, though?

CHAIRMAN PETTI: You want comments first? Okay. Sure.

MEMBER REMPE: Yes. Because I think that I'd like to hear what the public has to say.

CHAIRMAN PETTI: Okay.

MEMBER REMPE: Is that okay?

MR. SNODDERLY: Yes, that's the way we had -- we had it set up that way, before member discussion.

CHAIRMAN PETTI: Okay. So, for members of the public, *6 to unmute yourself. Give us your name and your comment.

MR. SNODDERLY: I'm sorry. Dave, we did something different today. So, the members of the public are on. So, you just have to unmute yourself.

CHAIRMAN PETTI: Oh.

MR. SNODDERLY: Yes. Yes.

CHAIRMAN PETTI: Okay.

MR. SNODDERLY: I don't know if Ms. Fields is still there, but I saw some members of the public in.

MS. FIELDS: Hi. Yes, this is Ms. Fields. Can you hear me?

CHAIRMAN PETTI: Yes, we can.

MS. FIELDS: Oh, okay. Because of another appointment, I didn't hear the whole presentation, but what I did hear and see was very, very informative. So, I just have a few comments.

As you know, there's still no permanent repository for irradiated nuclear fuel. And to me, it does not make sense for the United States to keep making used or spent nuclear fuel when there is no permanent repository and none anticipated in the near future, and I believe none in my lifetime.

Also, the NRC should not use the term "advanced" in these various rulemakings and documents. "Advanced" is a public relations term. It doesn't have a regulatory or a statutory or a technical basis.

The NRC has already dropped the term "advanced" in the Part 53 rulemaking. The rulemaking is not for advanced nuclear reactors; it's just for commercial nuclear reactors. And the NRC has taken the definition of "advanced" out of that rulemaking.

And there are some various reasons for that -- in part, I imagine, because the NEIMA definition of "advanced" just does not make sense. There's just no real basis for that.

Also, I'm still concerned that the NRC has not been able to accession properly the documents related to these pre-application and application submittals to the NRC. Many of the documents have been accessioned to ADAMS, but they have no docket number, or the docket number should be on an application document for an application docket, but it's still on the pre-application docket. The NRC has been paying attention to this and made some changes, but there's still a big issue.

I also think that the NRC and the ACRS should pay more attention to the needs of the public. Obviously, this whole process and the discussion today was really to satisfy the needs of the industry, as they move forward with these new reactor designs and proposals.

But there's little information available to the public that would give them an idea of what exactly they should look for, if there is an application to site one of these new reactor designs in their community. There's just really not much there that would guide the public, as they review these applications and they consider whether this is appropriate for their community.

So, I definitely think that the NRC should think -- well, maybe even have a public meeting for public input on what exactly the public, information the public needs, not so that the industry can site something in their community, but so the community has the tools it has to actually evaluate what's going on, like the amount of water that might be needed. Or is it really appropriate to reduce the EPZ and not have it at the traditional 10 miles?

So, it's just something for you all to think about in the future. Thank you.

CHAIRMAN PETTI: Thank you.

I believe Kati Austgen is next.

MS. AUSTGEN: Yes, thank you. Kati Austgen with the Nuclear Energy Institute.

First, I just wanted to thank the ACRS and the NRC staff for trying something new and letting us participate in the Teams webcast this time around. It was much easier to follow along on the slides and understand which members and which staff were making which comments. So, we very much appreciate that.

And then, I agree with the ACRS that this was a very informative presentation, and I look forward to sharing the information about the new website with our members, and continuing our conversations with the staff on guidance in this area and how we can all make sure we're on the same page about what's required.

Thank you.

CHAIRMAN PETTI: Thank you.

Jan Boudart?

MS. BOUDART: Jan Boudart, and I'm a Board member of the Nuclear Energy Information Service in Chicago.

And apropos to what Sarah Fields said, there is being considered an experimental reactor at the University of Illinois in Champaign-Urbana. And the question that NEIS is asking is, how do people feel about sending their kids to that University if there's going to be a fission project on the University campus? And how is the NRC, or whatever appropriate agency -- maybe it's the DOE -- how are they going to present this to the public and to the people who are considering sending their kids there? And is there going to be a public hearing? And will the ramifications of having this on the campus be explored, et cetera, et cetera?

But I think this is very appropriate to what Sarah said, because the people who are going to be affected by these small modular nuclear reactors and the experimental reactors, like the one that's being considered with the TVA in Tennessee, the people in the neighborhood and the people who are going to be in the area need to be consulted as to whether they think it's appropriate to have these things near them.

That's the end of my comment.

CHAIRMAN PETTI: Thank you.

Any other comments from members of the public?

MEMBER REMPE: Dave, am I -- oh, I'm sorry, I interrupted. Yes.

CHAIRMAN PETTI: Please go ahead.

MR. STEIN: This is Adam Stein from the Breakthrough Institute. I have a comment and a clarifying question, if you are willing to entertain the question.

To what extent does the staff expect to require a MELCOR model to be developed for specific advanced reactors? Because, as was presented today, the models that have been created so far are not direct replicas of the various developers; they are based on other similar technologies, or similar designs, I should say. But, as the staff mentioned today, a significant amount of time and effort was put into developing those models. So, to what extent is the staff expecting to require a MELCOR model that is similar to an applicant's design to be developed? And if not, how closely resembling results are they expecting between the existing MELCOR models or the ones that are being developed and an applicant's submission of results?

CHAIRMAN PETTI: Thank you. We don't respond to comments from the public in our meeting. But you can send an email to Mike Snodderly, the Designated Federal Official. Mike can tell you his email, and take it from there.

MR. SNODDERLY: Yes, Adam, I think if you want to send me a written comment, then we can add that to the record. But the other option is you just turn your question into a comment, and that also is being transcribed.

So, what I heard could be your comment is that MELCOR models, as they've been developed, are valuable and should be required or part of any review of a confirmatory analysis of a review of advanced reactors.

But you just need to form it in that kind of a phrasing. And if you want more time to do that, you can do that, and then, send me an email to michael.snodderly -- S-N-O-D-D-E-R-L-Y -- @nrc.gov.

MR. STEIN: Thank you.

I would also like to comment that it is important to consider that, when you are looking at the differences between a MELCOR model and a, potentially, applicant-developed model, that it is important to consider and disposition why and how they may differ in results. Because, as was discussed today, there are still various considerations or physical parameters that are not in both models identically, and there will still be dispositions.

Thank you for your time.

CHAIRMAN PETTI: Any other public comments?

I see a hand. Jan?

MS. BOUDART: It's me again.

I wanted to ascertain whether Mr. Snodderly's email, does it say @acrs.nrc.gov or is it just nrc.gov?

MR. SNODDERLY: Yes, Jan, it's @nrc.gov.

MS. BOUDART: Oh, good. Oh, thank you. That's what I thought. Thanks.

MR. SNODDERLY: Thanks.

CHAIRMAN PETTI: Okay.

MEMBER REMPE: Mike, is your hand up?

MR. SNODDERLY: Yes. If you don't mind, Dave, I just wanted to remind the Committee, as you start your member discussion, I went back to your presentation on October 8th to the Commission. And in that, just to remind everybody, in your slides that you presented -- let's see -- the second-to-last slide that you presented, you said that, "Numerous recent and upcoming source-term-related activities. A roadmap showing how all the pieces fit together would be worthwhile"; and that "Many different pieces are coming together, and that the ACRS plans an integrated review later this year."

So, I just wanted to remind the Committee that, well, you didn't make a commitment, but you told them that you were going to have this interaction. You've had it. It doesn't require a letter, of course. But I think the staff has done something to provide somewhat of a roadmap, and you may want to comment on how good or bad it is. But you, of course, can say nothing. But I just thought that might be a good kickoff point, and now, I'll be quiet.

MEMBER REMPE: Mike, also, while you're on the line, just so I understand the new process, everyone can just -- does anyone need to press *6 now? Because there are a lot of phone lines out there.

MR. SNODDERLY: No, they're on just like you and I, and it's mute and unmute.

MEMBER REMPE: Okay.

MR. SNODDERLY: Now I can mute folks, any participate, but --

MEMBER REMPE: Right, but the phone line folks --

MR. SNODDERLY: Yes.

MEMBER REMPE: I understand the folks that are linked in. Like Sarah has a real link.

MR. SNODDERLY: Yes.

MEMBER REMPE: But the phone line folks can it themselves? Okay. Just wanted to make sure we weren't leaving anyone out. Thank you.

MR. SNODDERLY: That's my understanding, but Tom Daschle (phonetic) -- you know, correct me if I'm wrong -- but I think that --

MR. BURKHART: This is Larry.

One of the commenters, our last commenter was on the phone. So, I think it's working.

MR. SNODDERLY: Yes. Thank you, Larry.

MEMBER REMPE: Great. Thank you.

CHAIRMAN PETTI: Okay. So, Members, let's talk a little bit about the need for a letter; what your thoughts are.

I have, as I always have, put together what I call the guts of a letter. It's just a bunch of points and starting points.

While I don't think the staff needs a

letter, my opinion is that, in light of everything that's going on right now in the area with non-LWRs, that a pointer letter, which is what the letter would be, pointing out some of the important things that they've done, might be quite helpful. And I'm certainly willing to write the letter, given the importance of the topic, as we heard from (audio interference) for the NRC.

So, comments, Members? And consultants? I'd be interested in hearing their perspectives, too.

MEMBER BROWN: It's Charlie.

Based on my listening, this was kind of a potpourri-type presentation. I head a lot of comments from us, but unless we have some specific things that should go on in this general, multiple opportunities for people, I don't see where a letter would really add a whole lot, other than just saying, "Continue." My opinion is I wouldn't write a letter based on this meeting.

MEMBER KIRCHNER: Dave, this is Walt.

I guess I would concur with Charlie. I would wait until you heard, we hear about DG-1389 and its status. Because what I'm thinking is, although this set of presentations today primarily is addressing, quote-unquote, "advanced" reactors and applicants coming in, it's likely that they're going to go in the near term, while 10 CFR 53 rulemaking is in progress, that they are going to come in under 10 CFR 50 or 52. And if they do so, then they can either do exceptions to guidance and the regulations or they'll follow the spirit of the guidance that's out there. And that's why I'm very interested in what DG-1389 contains and whether it's a broader template for other advanced reactor applications.

Thank you.

CHAIRMAN PETTI: Go ahead, Ron.

MEMBER BALLINGER: This is Ron.

I think that, unless we want to reinforce something that we've already communicated on at least two occasions -- the first with the Commission presentation and the second, at least the second today -- I think that we should probably -- I think the staff has got the message. And so, I think I'm in agreement with Walt -- waiting until we have something which we can actually review in detail.

Thank you.

CHAIRMAN PETTI: My only concern, having read that Draft Guide, is that it's extremely light-water-reactor-specific and we're going to focus on that. So, I don't see us getting into non-LWRs in that review because that's not where that is.

MEMBER BALLINGER: That would not keep us from making a comment, does it?

MEMBER KIRCHNER: That's what I was thinking, Dave. You can build on that to say whatever we feel -- more, less, or keep on going, or whatever -- for the non-LWR reactors or something new, completely new and different is needed. That would be a good juncture to make that kind of observation.

MEMBER HALNON: Dave, this is Greg.

CHAIRMAN PETTI: Dennis? Oh, go ahead, Greg.

MEMBER HALNON: No, well, Dennis had his hand up. I couldn't see that.

Go ahead, Dennis. I'll come after you.

DR. BLEY: Okay. Yes, Dave, well, first, I'd say I found today very useful, and I've played around on that website already this afternoon. I think that's going to be especially nice.

Now we sometimes get tricked in things we read, as Joy said, in the popular press and in papers people send to us to look at. I'm not sure how much of what I read is politics and how much is real, where people are technically.

There's a lot of things passing around

that kind of say, "Gee, the NRC is stuck in the Dark Ages and they ought to just say, almost, you don't need to do anything with these new reactors. Just let them operate."

I think what we saw today ties a lot of very important information together. And if I had a new design and it had extremely low consequences and likelihood of damage, I think I could do something not terribly elaborate to prove that point within those constructs.

So, I think what they've put together today is really useful, integrating all these pieces. I would have agreed with Walt and Ron that we ought to wait for the meeting next month, and there's when it would be most useful for you folks to write a letter. But I didn't read the draft and you have. And if they aren't closely coupled, then, maybe it makes sense separately.

I mean, this isn't at a point where this rates a 10-page letter. A one-page letter might be appropriate to acknowledge what's being done and its value. So, I think we shouldn't let this pass without -- I don't think you should let it pass without some comment.

CHAIRMAN PETTI: So, yes, my talking

points are all of 80 lines. You know, it's not a big letter. But we're viewed as sort of an independent look at things, and there's these questions out there by all sorts of folks about NRC, as you say, being in the Dark Ages and they don't really understand these non-LWRs. And I think nothing could be further from the truth. They've done an awful lot of work to get themselves ready to accept an application. And to recognize that, I thought, personally, was of value.

MEMBER HALNON: This is Greg.

I look at it from a perspective of the Commissioner sitting in his or her office, and somebody walking in wanting a drop-in meeting and have discussion about some iterations they're going through with reviews, and whatnot.

And I guess there's a couple of nuggets that came out today. And notwithstanding the one about the burden of developing the methodologies is on the developer and expect it to be complicated, and those sorts of things, there's a couple of messages that I think the Commissioners need to hear, so that they're armed with some information.

And I'm not sure if it gets to them or not, but the fact that there is an integrated website and it's in process; it's in really pretty good shape, but we're taking comments on it; the fact that there's a couple of nuggets in there that we heard today that we're either expecting or anticipating or have learned -- I think those type of things, going to the transcripts, might be useful.

And I'm not talking about a long letter, but a couple of sentences here and there that acknowledge that we're in an iterative-plus process that's going to be getting better every time we have interaction with a new technology and we all learn more about it. I think those types of things are pretty important.

So, I'm kind of in with you and Dennis, Dave, that there could be a very short letter that has some of the things that we heard, some of the things that we anticipate, and wouldn't negate having another letter down the road for a specific Reg. Guide, but, certainly, from the topic itself, could use some, I guess, collation into a concise letter.

CHAIRMAN PETTI: Steve? I see your hand up.

DR. SCHULTZ: Yes. For the Committee, I do appreciate the discussion here about a letter, but I don't think it's letter or no letter. Because I think the Committee has made these comments to the

Commission in their last meeting; that this is a topic that is extremely important to the advancement of the new reactors; and also, to the work that is being done associated with their development and application.

And I think a followup to just those comments to the Commission would be very useful, especially in light of the excellent presentations that were made today. A very well-integrated presentation and very useful to the Committee and, also, in the public forum.

My other comment would be that not only is this of interest in the United States, but the international community. The regulatory agencies in those areas are also working on the same thing. And for the ACRS to not come out with a continuing statement about what is ongoing and what is appreciated and associated with this work -- I think that's important for the Committee to do. Whether it's one letter or two letters, it's up to the Committee. I think that an individual letter on this topic would be not the last letter the Committee writes, but very important to continue to endorse.

CHAIRMAN PETTI: Joy?

MEMBER REMPE: So, I'm torn. Clearly, we need to decide because the month is coming up and the staff needs to know if they're going to be coming back to us.

But, listening today to the public comments, it seemed to me, and what the staff said, that they're going to have an upcoming stakeholder meeting in March, where they'll maybe learn more about whether more guidance is needed -- I mean, clearly, they've made a lot of good progress and they've got a good -- you know, I really do like the reference plant evaluations. I'd like to see a little more coming out of it, which I think the staff was receptive to.

I think that some of our comments about the need for guidance is important, but I'm not sure whether we should be telling the staff to do that, when I really appreciate that there's so many different designs out there, that maybe the staff is at the right -- that what they're doing is correct. Because why waste resources on doing guidance for so many different designs and flexibilities?

And one of the public comments, to me, indicated that the person didn't understand the staff's broad perspective. I don't think the staff is trying to say, by any means, that a MELCOR model is needed in what we've heard today. So, I think communication with the stakeholders is very important.

But that was my impression of it and all, but I don't know if the timing is right. I definitely would put "Interim Letter on Staff Progress," or something like that, as the title of whatever is issued. But I'm open to whatever the Committee decides, but I don't know if this is the right time or not.

I'm not sure that's a very helpful comment.

CHAIRMAN PETTI: My biggest concern is, you know, there's probably never a good time. And we know the bow wave of what's coming in front of us, starting in April. There just seemed to be a window of opportunity here, given what's on the plate for March.

MEMBER REMPE: But what will we say other than, "Good job."? And we can do that in the minutes of the meeting summary, when you come back and do something like what Jose often does. I don't know; I just am wondering if we're making them come back to present to us and things like that. So, anyway, I'm just kind of thinking about it.

CHAIRMAN PETTI: Anybody else? Matt? Jose?

MEMBER MARCH-LEUBA: Yes, Jose. Jose has

his hand raised because Joy used my name in vain, and she took my argument out of it.

We have an intermediate method of dealing with this. First, let me say that I'm ambivalent on the letter. On the one side, the person that knows and cares more about this is Dave, and he thinks we should have a letter. And that makes me want to support it. On the other side, I see all the other arguments from the people that maybe we should wait for a real Reg Guide.

But we have a middle ground, which is writing a couple of paragraphs to be included in the summary of this meeting. And they get presented to P&P, and they become the property of ACRS. It's not a subcommittee any longer. I mean, once P&P approves those two or three paragraphs, it's half a letter, and those paragraphs can be submitted by email to whomever.

So, if we are voting, I will vote for the halfway and go for the summary.

MEMBER REMPE: And that avoids the staff having to come back and do another presentation.

CHAIRMAN PETTI: Matt, you've been silent.

MEMBER SUNSERI: I've been listening, Dave. You know, I think it was a very important topic. We've been asking for how these various source term activities integrate each other towards the topic. And so, I'd be in favor of a letter.

MEMBER BIER: This is Vicki.

It's kind of out of my area. I feel sort of similar, in between Charlie and Dennis; that, you know, on the one hand, there was nothing burning today that inspired me that we have to write a letter, but I could certainly go with a short letter that just said, "We support the progress being made," et cetera.

CHAIRMAN PETTI: Okay.

MEMBER DIMITRIJEVIC: Dave?

CHAIRMAN PETTI: Yes?

MEMBER DIMITRIJEVIC: You know, I believe in Jose's philosophy. You know most about that and you put the letter together, and you think it should be written. So, I support that.

MEMBER SUNSERI: Dave, this is Matt. One more thing.

I think that we have precedents, also, for not having to have staff come back and make another presentation to the Committee. I think maybe almost everybody from the Committee is here, and that you could make some kind of presentation, a short one, at the full Committee meeting, and we could write the letter from that. I think we've done that before on a lot of the license reviews that we did for design certifications.

MEMBER REMPE: We did it for MELLA+, actually, and that would make it easier for the staff. I don't know.

CHAIRMAN PETTI: Yes. I certainly don't -- I mean, we would need a very, very, 100,000-foot level presentation, because this here, I'm thinking I would have thought no more than five slides sort of thing. You know, what were the things you touched on? But if we can even get rid of that, maybe, then, I could put a draft together, and then, when we see it, that might even inform us better. To bring it to the full Committee, and then, people make a decision? Is that allowable?

MR. SNODDERLY: Yes, I believe that's consistent with the Bylaws. The Subcommittee needs to make a recommendation to the full Committee on whether to write a letter or not. This would also give some more time to digest it and think about it.

But, Dave, you would, when we come to this item on the agenda, you would make, as a member, a recommendation to the Committee, and then, the Committee would vote whether to take it up further and write the letter or not, or to do some hybrid thing.

And it's already been discussed or the precedent has been set. The staff could be there or not be there. But, of course, we would like them there to support letter writing as usual, anyway. But I don't even know what, if any, presentation would be needed, unless you requested it, what you think you need to support the letter.

CHAIRMAN PETTI: All right. I don't think we need them to be there to make a presentation.

But let me try to write --

MEMBER REMPE: Let me interrupt for a minute, and I think Larry is going to have to weigh-in because the agenda is already published in The Federal Register.

If we were to do this in P&P, that puts this on Friday morning, just to be thinking processwise here and the mechanics of it. I don't think we can have -- or maybe we can; I'm asking Larry. Is it acceptable for Dave to give a presentation during that allocated time for this topic, Larry, instead of the staff, and go ahead with, if the Committee at that point decides to do a letter, is it acceptable to go ahead and go right into letter writing?

MR. BURKHART: I think it's perfectly acceptable for the Chair of the Subcommittee to orient the full Committee on the topic, and then, move into report writing, if that is decided.

MEMBER REMPE: Okay. I just wanted to make sure, because this is a little different than what happened with the MELLA+ thing. But that's great. I just wanted to make sure we weren't going to go forward with something that wasn't allowed. Thank you.

CHAIRMAN PETTI: The other thing is what Mike raised. You know, we did talk about this with the Commissioners, and it would be nice to kind of close the loop on that. Again, a shorter letter, that both sides recognize this issue; thought it was timely, and we did.

So, it's on the agenda. I will, basically, I guess, during that slot, talk about the major points in the letter. And we can have a discussion there, and then, make a decision, as the full Committee, whether we want a document for the record or, in a summary sense, in the meeting summary. Is that reasonable?

MEMBER SUNSERI: Sounds good to me.

MEMBER DIMITRIJEVIC: Absolutely.

Absolutely.

CHAIRMAN PETTI: Okay. I will work on this next week, and I'll actually try to get some of the stuff to folks before the meeting, so they're not seeing for the first time at full Committee.

Okay. Then, with that, I guess if there are no other comments, we're ready to adjourn the meeting.

And I thank everyone and, again, thank the staff. This was a Herculean effort. We really do appreciate it.

So, everybody have a good evening. Thank you.

(Whereupon, at 4:28 p.m., the meeting was adjourned.)

Advisory Committee on Reactor Safeguards
Future Plant Designs Subcommittee Meeting on
Integration of Source Term Activities in Support of Advanced
Reactor Initiatives
February 17, 2022

AGENDA

- Opening Remarks
- Staff Introduction
- History and Evolution of LWR Source Term
- NRC analytical tools and past studies
- SCALE/MELCOR non-LWR reference plant analysis

Break

- Agenda Item IV Continued
- NuScale EPZ Sizing Methodology Topical Report, Rev. 2
- Light water SMR design certification source term approach
- Source term approach for early non-LWR movers

Lunch

- Accident-consequence-related regulation activities

Break

- Guidance and information for developing advanced reactor source term
- Guidance for developing advanced reactor source term (long-term)
- Opportunity for Public Comment
- Member Discussion

Adjourn

Integration of Source Term Activities in Support of Advance Reactor Initiatives

John Segala
NRR/DANU

February 17, 2022

Staff Introduction

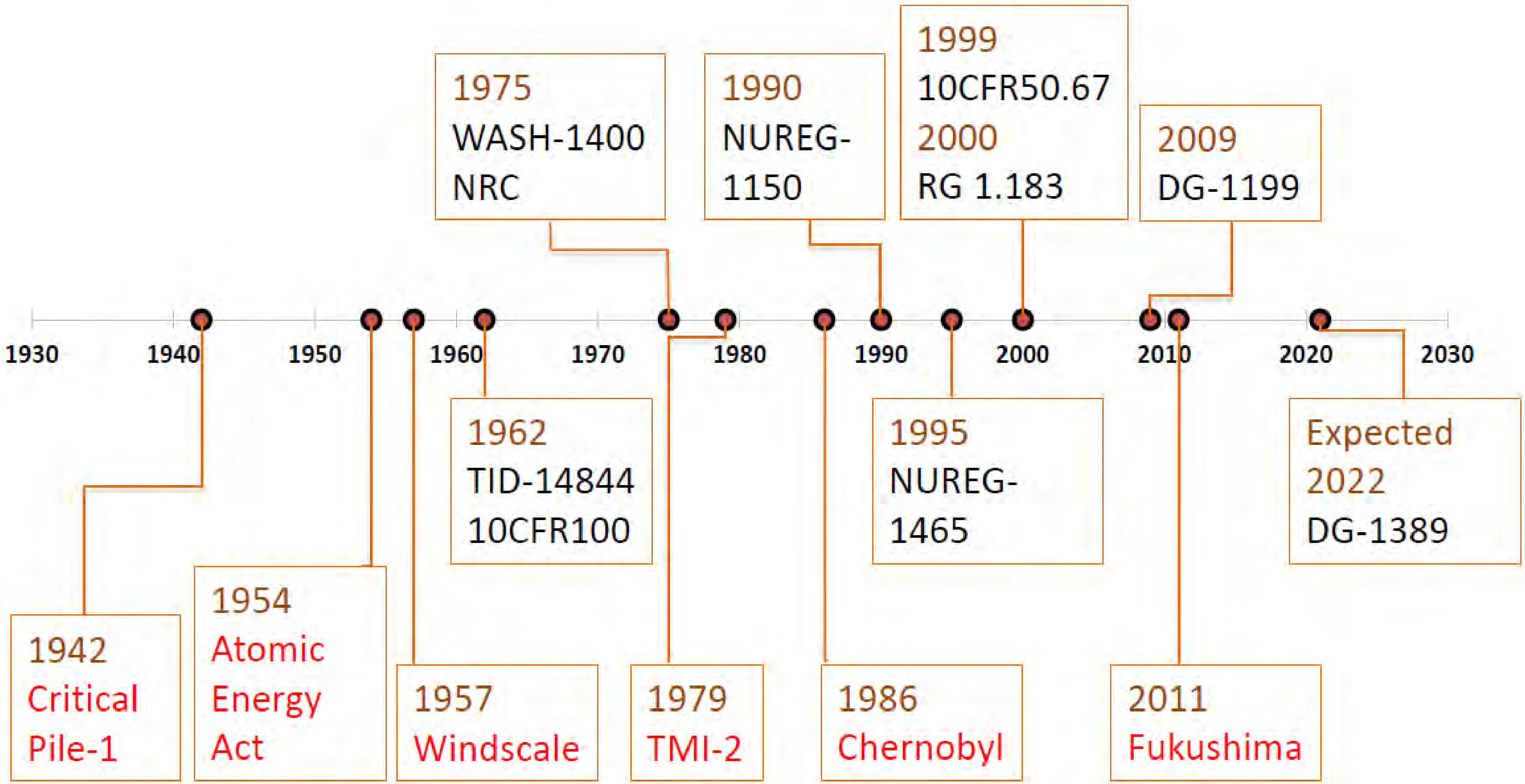
- Determining source terms is a critical component in the NRC's licensing process
- NRC team presenting today:
  - Mark Blumberg – NRR/DRA
  - Michelle Hart – NRR/DANU
  - Jason Schaperow – NRR/DANU
  - Bill Reckley – NRR/DANU
  - Tim Drzewiecki – NRR/DANU
  - Hossein Esmaili – RES/DSA

History and Evolution of LWR Source Terms

Mark Blumberg

Radiation Protection and Consequence Branch
Division of Risk Assessment
Office of Nuclear Reactor Regulation

LWR Source Term Timeline



History – Regulatory Use of Source Terms

- Siting critical issue
  – Safety & Cost
- Principle hazard – Public Exposure
  – Siting key element in protecting public health
- Earliest reactors used containments
- Atomic Energy Commission proposed siting on population densities
- Ultimately decided siting would be based upon dose calculations

10 CFR 100.11

- Footnote to 10 CFR 100.11(a) is a performance-based rule to evaluate the defense-in-depth provided by the containment
- Nearly all current reactors were licensed originally to the Technical Information Document (TID) -14844 which provides guidance on the containment source term for the Loss of Coolant Accidents (LOCAs) involving fuel melt
  - Based on heating fuel 'chips' in a furnace
  - 100% noble gases (Xe, Kr)
  - 50% iodine (half deposits instantaneously)
  - 1% of other radionuclides as particles
- Iodine Chemical Form
  - 91% as I_2(g) (elemental); 5% particles; 4% CH_3I (organic)
- All instantly available from start of accident in the containment
- Source terms for Non-LOCA events are provided in RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light Water Reactors"

NUREG-1465 Source Term

- Radionuclide behavior observed during the Three Mile Island Unit 2 accident in 1979 did not appear at all to be like the Technical Information Document (TID)-14844 source term
- NRC initiated research effects in the area of severe accidents which culminate in publication of NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants.” (1990)
- Source term depends on the nature of the accident
- The NUREG-1465, “Accident Sources Terms for Light-Water Nuclear Power Plants” (1995) source term was derived from the risk significant sequences in NUREG-1150

10 CFR 50.67, RG 1.183

- NRC staff developed RG 1.183 Rev. 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." (July 2000) to support implementation of 10 CFR 50.67, "Accident Source Term"
  - Applicable to nuclear power reactor applicants and licensees who voluntarily adopt 10 CFR 50.67
  - Provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST
  - Used the NUREG-1465 early in-vessel fuel melt source term for LOCAs
  - RG 1.183 also provides Non-LOCA release fractions
  - Identified the significant attributes of an acceptable AST

TID-14844 vs. NUREG-1465

BWR Source Term

	NUREG 1465		TID 14844
	Gap	Early In-vessel	
Duration (hours)	.5	1.3	Instant.
Noble Gases (%)	5	95	100
Halogens (%)	5	25	50
	Elemental I2– 4.85 Aerosol (CsI) – 95 Organic 0.15		Elem. – 91 Aerosol –5 Organic 4

Source Term Updates Proposed in DG-1199

- In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183
- Addressed fuel utilization at the time for Non-LOCA accidents
- The NRC staff has elected not to finalize DG-1199 and is issuing DG-1389 as a replacement

Source Term Updates Proposed in DG-1389

- Staff plans to include changes proposed in DG-1199 as modified by public comments
- Provides guidance to address the review of near-term accident tolerant fuel (ATF) designs with burnups up to 68 GWd/MTU peak rod-average) and U-235 enrichments up to 8.0 weight percent.
- Considered impact of fuel fragmentation, relocation and dispersal[1]
- On going research efforts is underway to update the SAND2011-0128 accident source term to accommodate higher burnup and increased enrichments for LOCA releases.

1 NRC Memorandum, "Letter Report on Evaluation of the Impact of Fuel Fragmentation, Relocation, and Dispersal for the Radiological Design Basis Accidents in Regulatory Guide 1.183 (ADAMS Accession No. ML21197A067)"

Source Term Updates Proposed in DG-1389 (cont.)

- A future RG 1.183 update is expected to accommodate higher burnups and enrichments
- An acceptable analytical procedure for predicting plant-specific non-LOCA radionuclide release fractions has been included and provides flexibility and margin recovery
- Separate BWR and PWR non-LOCA steady state release fractions

Key Messages

- One of the ways the NRC staff and licensees determine what measures and barriers are needed to protect the health and safety of the public is to perform design basis accident dose analyses.
- A key component of these analyses is the determination of the release source term.
- The NRC has developed regulations, source terms and regulatory guidance to provide licensees and the staff with an efficient method of performing these dose analyses.
- Ongoing efforts by the NRC continue to revise these source terms and methods to address modern fuel utilization and the use of accident tolerant fuel.

NRC Analytical Tools and Past Studies-Severe Accident Progression and Source Term

Hossein Esmaili, RES/DSA
Jason Schaperow, NRR/DANU

Key Messages

- Decades of NRC and international investments in the state-of-practice SCALE and MELCOR modeling including development, assessment and application
- Importance of analytical capabilities in a system level code and being ready to resolve regulatory issues and help decision making
- Leverage international collaboration through severe accident research and code sharing programs
- Application to a wide variety of nuclear technologies

Outline

- Introduction
- MELCOR Code Overview
- International Collaboration (Severe Accidents & MELCOR)
- Applications to Regulatory Decision-making
  - Examples: Design Certification, SOARCA, Post-Fukushima activities
- Application to New and Advanced Reactors
  - SCALE/MELCOR demonstration calculations

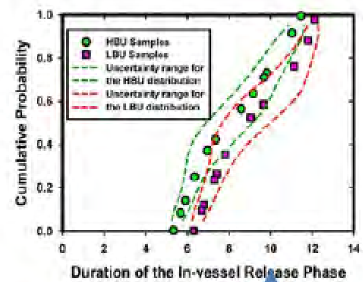
Introduction

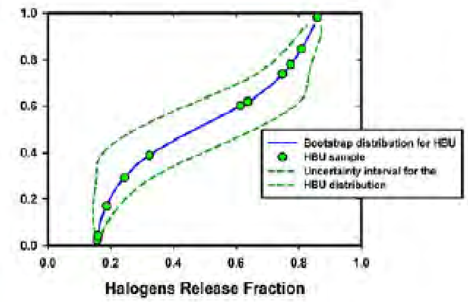
Source Term Development Process

Experimental Basis ← PIRT process

- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

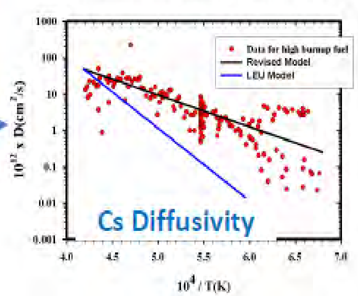
e.g., NUREG/CR-7283 (ATF/HBU)





MELCOR

Cesium release from high burnup fuel
Comparison to results of VERCORS Test



Accident Analysis

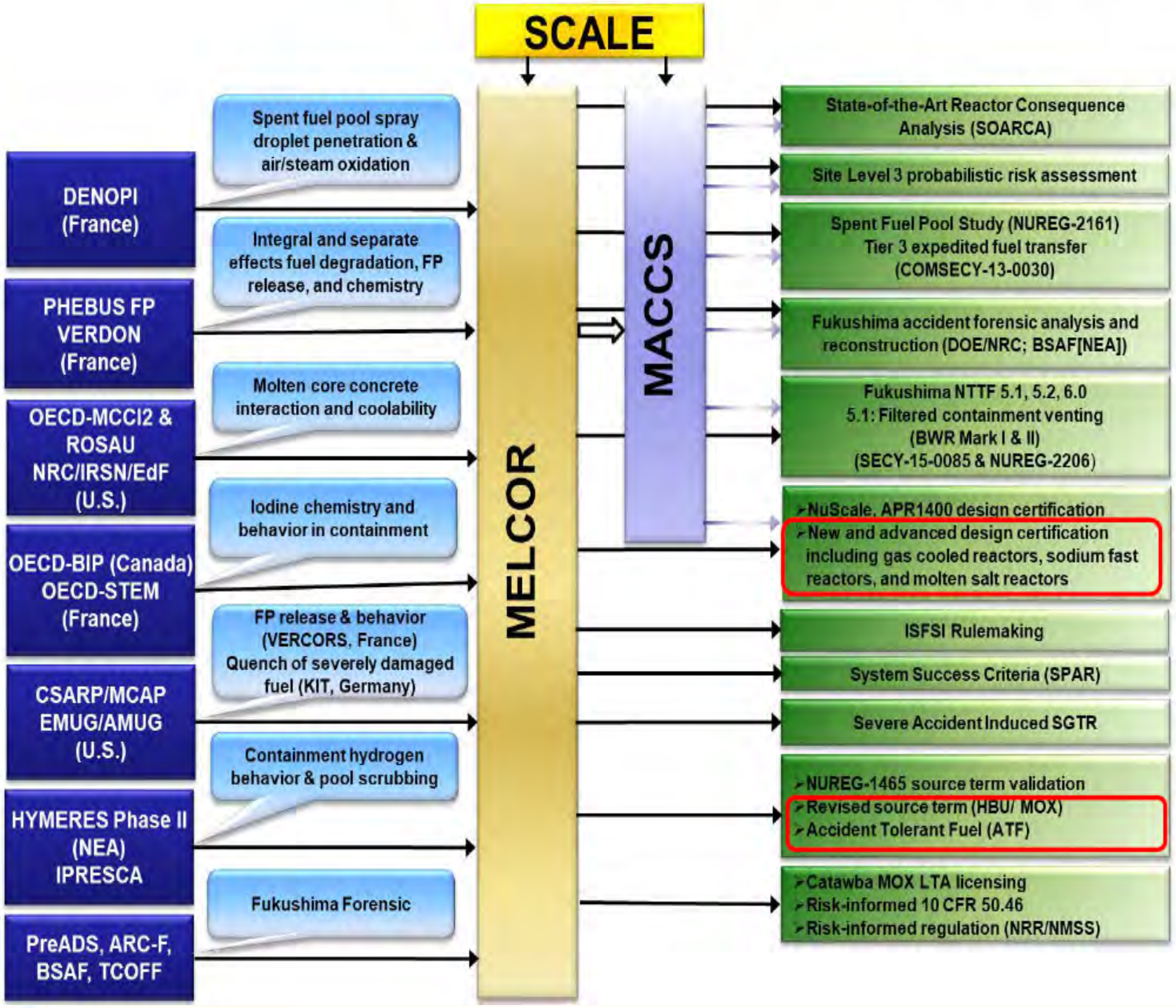
Scenario # 1	Scenario # 2
…………………	…………………
Scenario # n-1	Scenario # n

Synthesize timings and release fractions

Design Basis Source Term

Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	[illegible]
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0

Code Development & Regulatory Applications



What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

Who Uses It?
MELCOR is used by domestic universities and national laboratories, and international organizations in around 30 countries. It is distributed as part of NRC's Cooperative Severe Accident Research Program (CSARP).

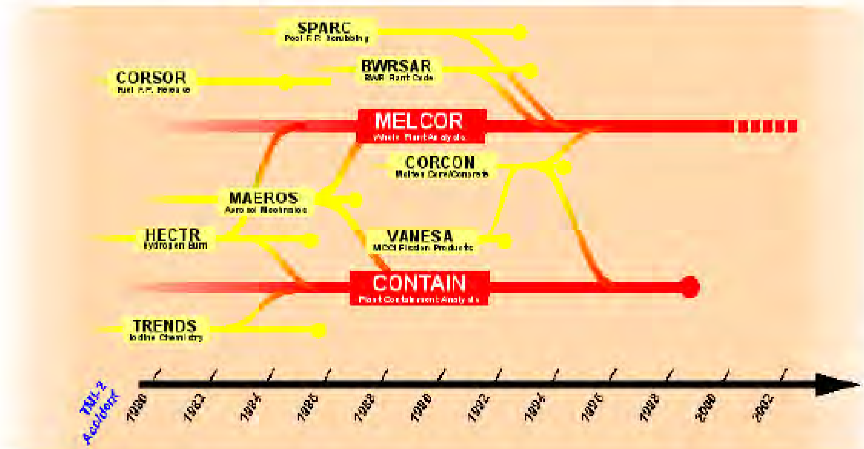
How Is It Used?
MELCOR is used to support severe accident and source term activities at NRC, including the development of regulatory source terms for LWRs, analysis of success criteria for probabilistic risk assessment models, site risk studies, and forensic analysis of the Fukushima accident.

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

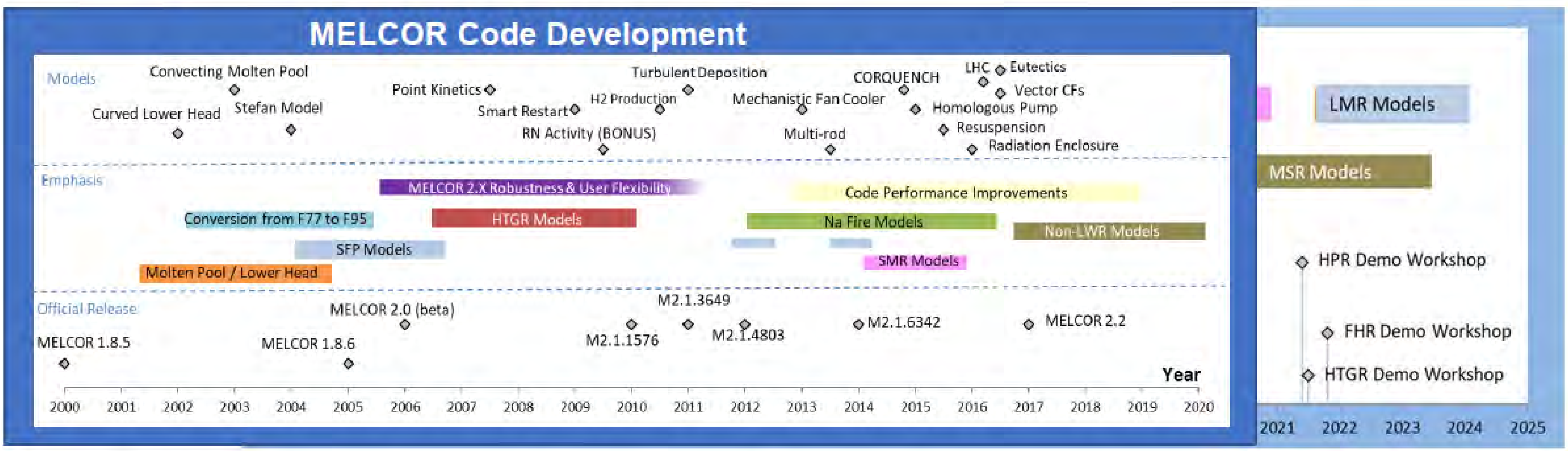
MELCOR Overview

MELCOR History

MELCOR developed at Sandia National Laboratories for NRC since 1982



Version	Date
2.2.21440	December 2021
2.2.18180	December 2020
2.2.14959	October 2019
2.1.11932	November 2018
2.1.9541	February 2017



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MELCOR Development

Fully integrated, engineering-level code

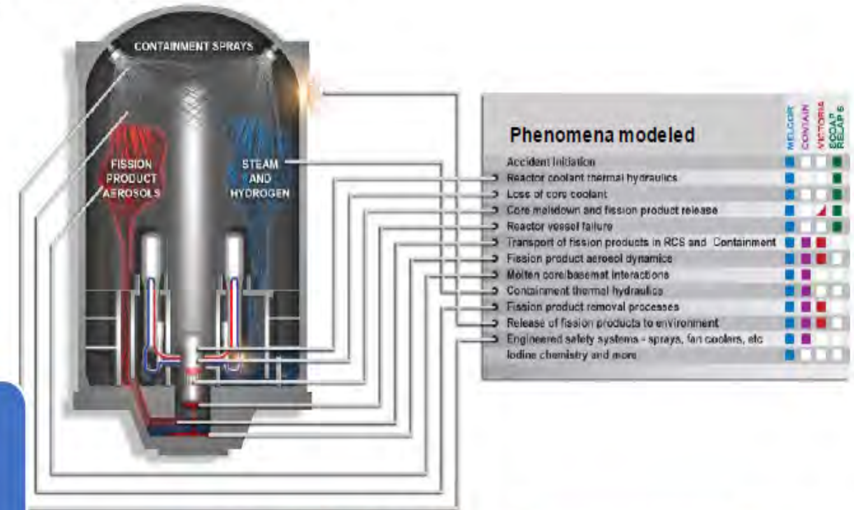
- Thermal-hydraulic response of reactor coolant system, reactor cavity, rector enclosures, and auxiliary buildings
- Core heat-up, degradation and relocation
- Core-concrete interaction
- Flammable gas production, transport and combustion
- Fission product release and transport behavior

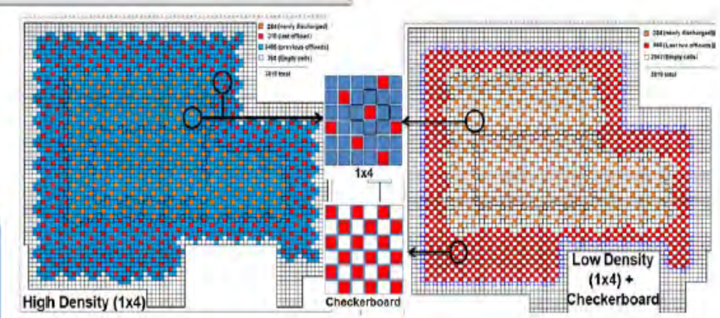
Level of physics modeling consistent with

- State-of-knowledge
- Necessity to capture global plant response
- Reduced-order and correlation-based modeling often most valuable to link plant physical conditions to evolution of severe accident and fission product release/transport

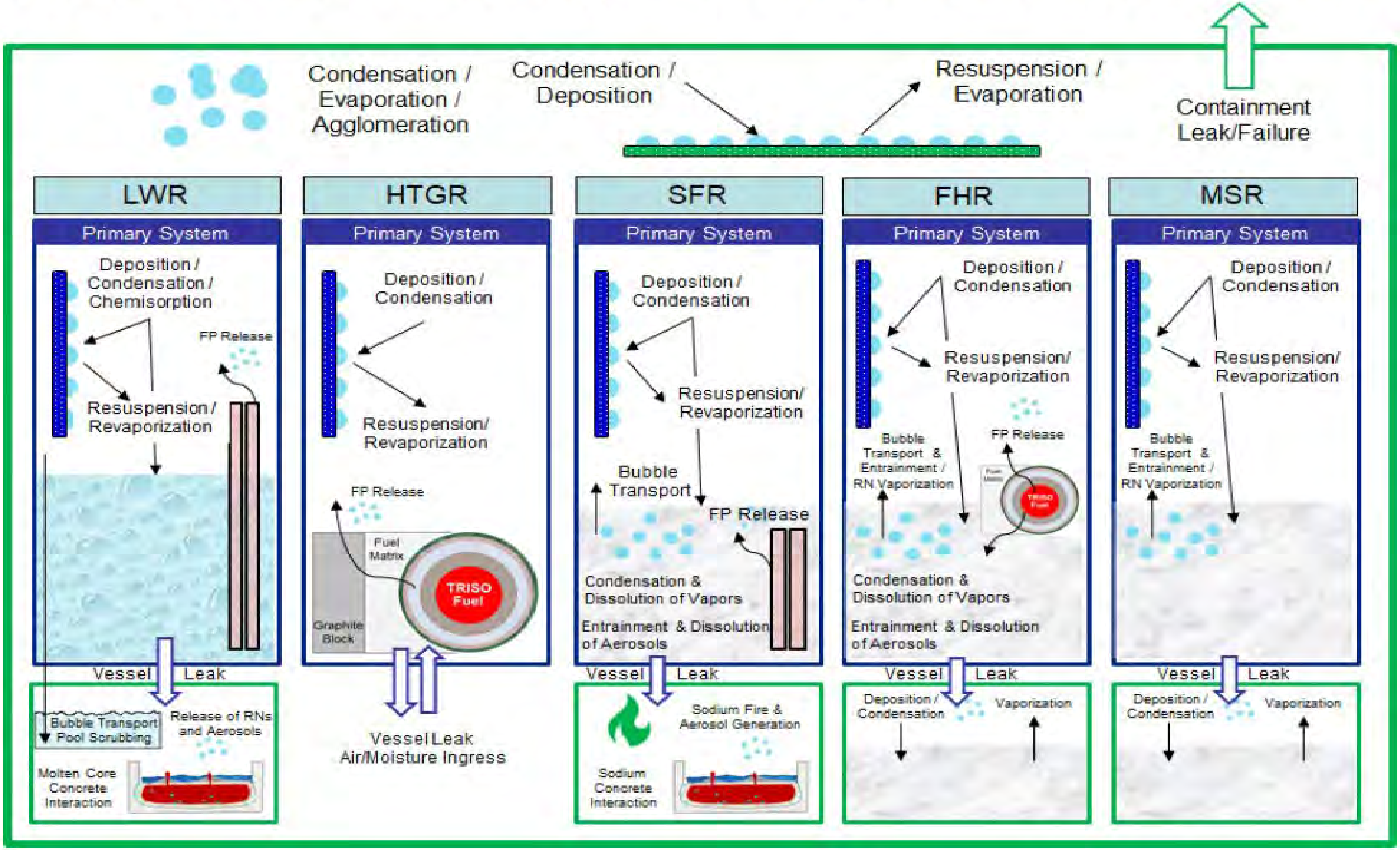
Traditional application

- Models constructed by user from basic components (control volumes, flow paths and heat structures)
- Demonstrated adaptability to new reactor designs – HPR, HTGR, SMR, MSR





MELCOR Flexibility - Common Phenomena



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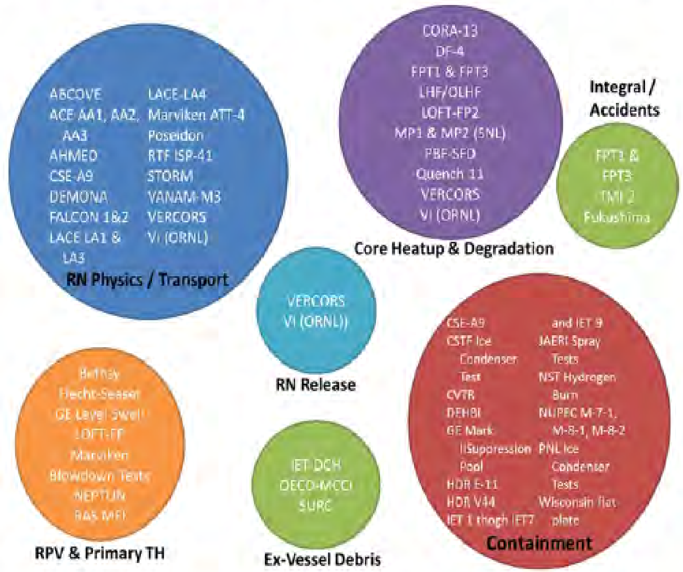
MELCOR Verification & Validation Basis

Primer & User Guide
Reference Manual
Assessment Problems

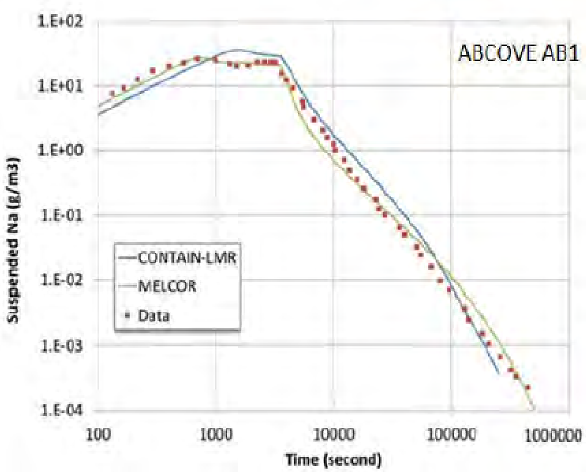


TRISO Diffusion Release

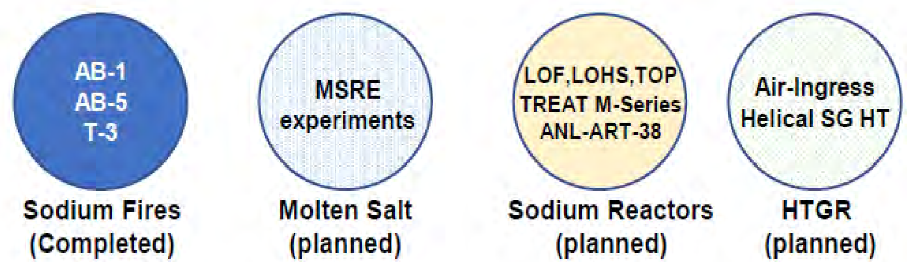
IAEA CRP-6 Benchmark
Fractional Release

Case	1a	1b	2a	2b	3a	3b
US/INL	0.467	1.0	0.026	0.996	1.32E-4	0.208
US/GA	0.453	0.97	0.006	0.968	7.33E-3	1.00
US/SNL	**0.465**	**1.0**	**0.026**	**0.995**	**1.00E-4**	**0.208**
US/NRC	0.463	1.0	0.026	0.989	1.25E-4	0.207
France	0.472	1.0	0.028	0.995	6.59E-5	0.207
Korea	0.473	1.0	0.029	0.995	4.72E-4	0.210
Germany	0.456	1.0	0.026	0.991	1.15E-3	0.218

(1a): Bare kernel (1200 °C for 200 hours)
(1b): Bare kernel (1600 °C for 200 hours)
(2a): kernel+buffer+iPyC (1200 °C for 200 hours)
(2b): kernel+buffer+iPyC (1600 °C for 200 hours)
(3a): Intact (1600 °C for 200 hours)
(3b): Intact (1800 °C for 200 hours)



Specific to non-LWR application



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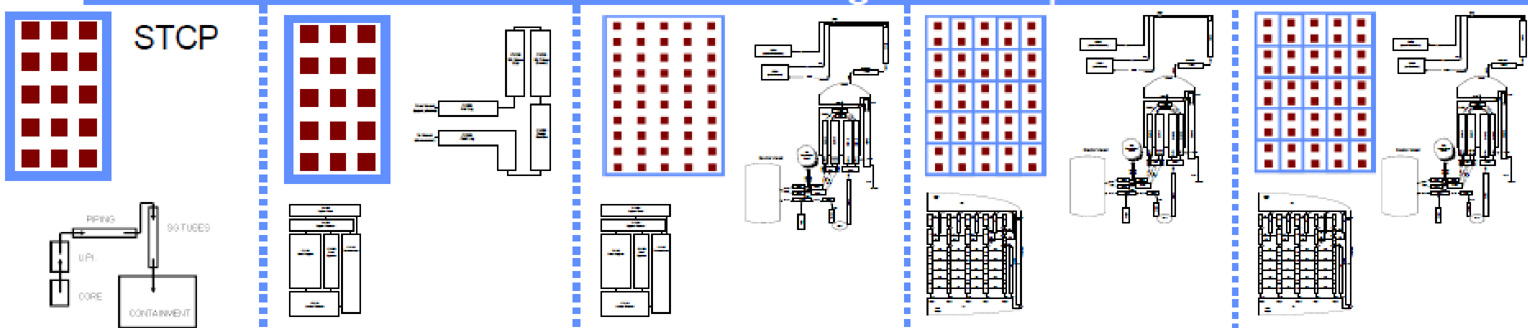
MELCOR State-of-the-Practice Modeling

Timeline for Evolution of MELCOR Modeling Practices

Circa 1985	Circa 1990	Circa 1995	Circa 1998	Present	Future

Modeling Techniques

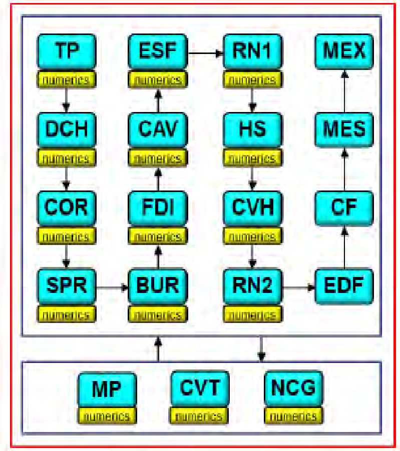


Example Regulatory Applications

Circa 1985	Circa 1990	Circa 1995	Circa 1998	Present	Future
• NUREG-1150 • Basis for NUREG-1465 revised source term	• AP-600 design certification • ESBWR design certification • AP-1000 design certification	• Begin SGTR	• Finish SGTR	• MOX source terms • High burn-up source terms	• Emerging user needs

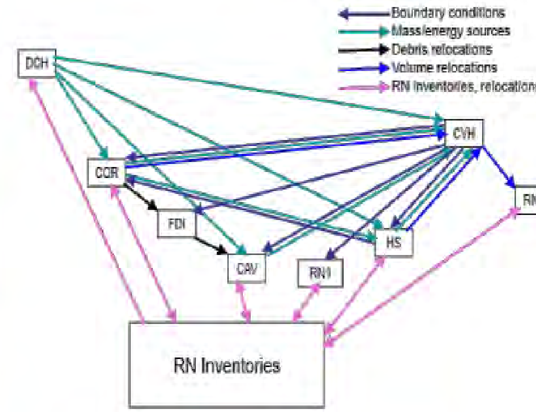
MELCOR Modernization

- Generalized numerical solution engine
- Hydrodynamics
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport

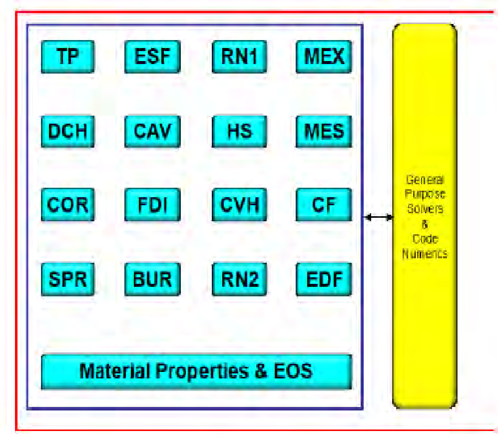


TP = Transfer Process
DCH = Decay Heat
COR = Core
SPR = Containment Spray
BUR = Gas Combustion
FDI = Fuel Dispersal Interaction
CAV = Cavity (MCCI)
ESF = Engineered Safety Features
MP = Material Properties

RN = Radionuclide
HS = Heat Structure
CVH = CV Hydrodynamics
EDF = External Data File
CF = Control Function
MES = Special Messages
MEX = Executive
CVT = CV Thermodynamics
NCG = Non Condensible Gas

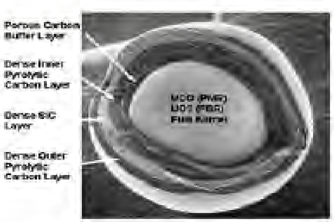


Separate **Physics** & **Numbers**



MELCOR Data Requirements

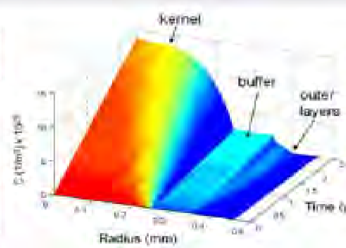
Input Data	HTGR	SFR	MSR	FHR
FP Inventory	SCALE	SCALE	SCALE	SCALE
FP diffusion coefficients (D) and release	Experiments (e.g., AGR) and analysis (e.g., DOE tools)	Experiments		Experiments (e.g., AGR) and analysis (e.g., DOE tools)
Core power shape	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)	Radial/Axial profiles (e.g., SCALE)
Fuel failure	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)		Experiments/other codes (e.g., DOE tools)
Dust generation & FP transport	Experiments, historical data and other code (e.g., DOE tools)			
FP release under air/water ingress & interaction w/ graphite	Experiments			
Kinetics parameters and reactivity feedback coefficients	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)	Experiments/other codes (e.g., SCALE)
Equilibrium constants for release from pool and vapor pressure data		Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)	Experiments/other codes (e.g., DOE tools)



$$\frac{\partial C}{\partial t}=\frac{1}{r^m}\frac{\partial}{\partial r}\left(r^m \boxed{D}\frac{\partial C}{\partial r}\right)-\lambda C+\boxed{S}$$

SCALE

Experiments/Analysis



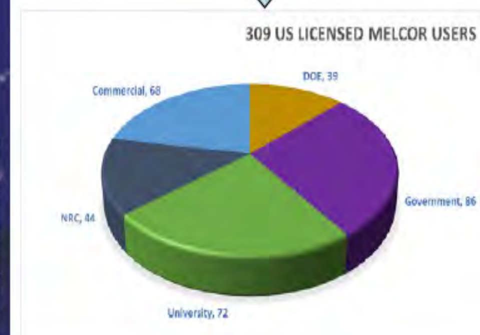
Distribution of Cs-137 in different layers as a function of time

International Collaboration

User Groups & Technical Meetings

Cooperative Severe Accident Research Program (CSARP) – June/U.S.A
MELCOR Code Assessment Program (MCAP) – June/U.S.A
European MELCOR User Group (EMUG) Meeting – Spring/Europe
Asian MELCOR User Group (AMUG) Meeting – Fall/Asia

~1000 Code Users Worldwide

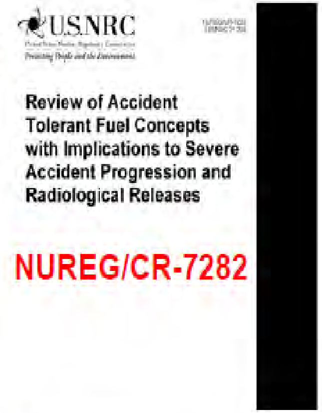


International Severe Accident Projects

- **1988-2010** — IRSN: Phébus-Fission Products Program
- **2005-2010** — IRSN: Phébus-International Source Term Program
- **2006-2019** — NEA: Behavior of Iodine Project (BIP)
- **2005-2016** — IRSN: Experimental Program for Iodine Chemistry Under Radiation (EPICUR)
- **2011-2019** — NEA: Source Term Evaluation and Mitigation (STEM) Project
- **2013–2018** — NEA: Benchmark Study of the Accident at Fukushima (BSAF) Project
- **2017–2021** — NEA: Preparatory Study on Analysis of Fuel Debris (PreADES) Project
- **2019–2021** — NEA: Analysis of Information from Reactor Buildings and Containment Vessels of Fukushima (ARC-F) Project
- **2019–2023** — EU: Management and Uncertainties of Severe Accidents (MUSA)
- **2020–2024** — NEA: Experiments on Source Term for delayed Releases (ESTER)
- **2020–2024** — NEA: Reduction of Severe Accident Uncertainties (ROSAU)

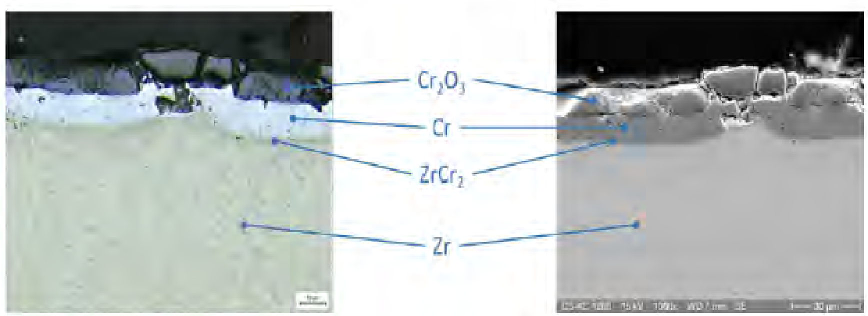
Advanced Fuel Technologies

- Panel of international severe accident experts Phenomena Identification and Ranking Tables (PIRT) that addressed significant phenomenological issues to improve MELCOR
- Source term calculations for HBU/HALEU fuel
- **QUENCH-ATF:** Experiments for ATF cladding materials in the QUENCH facility at Karlsruhe Institute of Technology (KIT) – Near term chromium-coated cladding under design basis accident (DBA) and beyond DBA



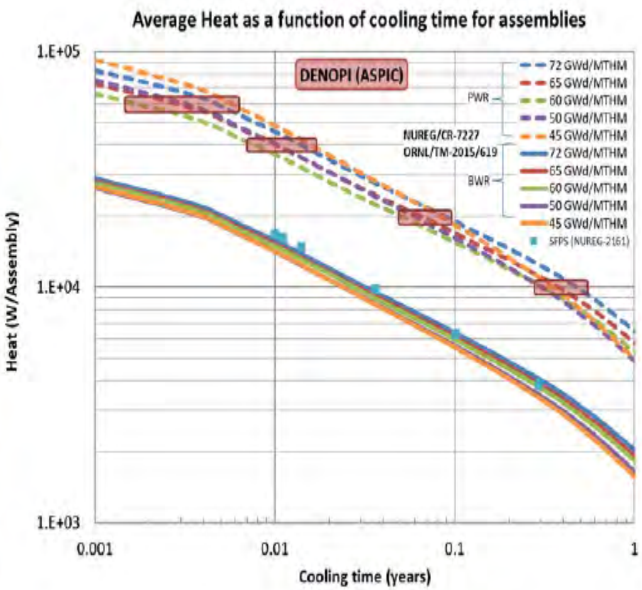
NUREG/CR-7282

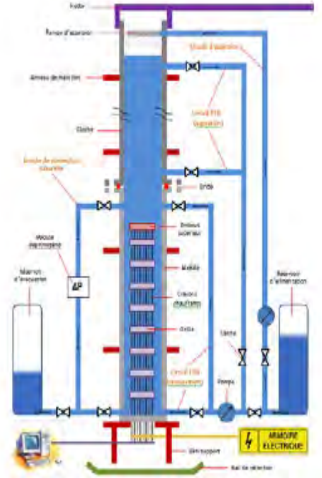
NUREG/CR-7283



MELCOR SFP Modeling

- **SECY-16-0100**: "National Academy of Sciences Study of the Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Power Plants"
- **DENOPI (NRC-IRSN/France):** Provide experimental data to validate spray efficacy on cooling spent fuel bundles, and cladding oxidation under a mixture of steam and air environment.
- Enhance MELCOR SFP capabilities

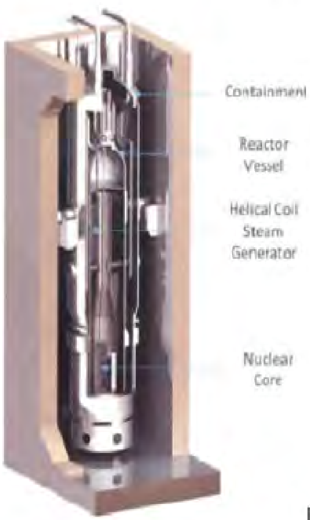




MELCOR Applications

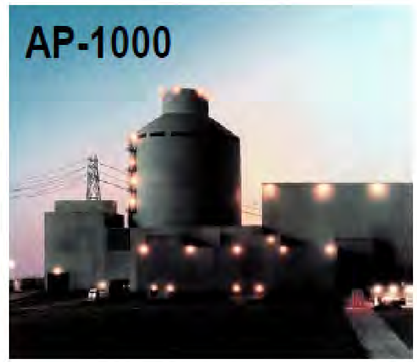
Design Certification

- Severe accident response and source term
- Containment response to design basis accident

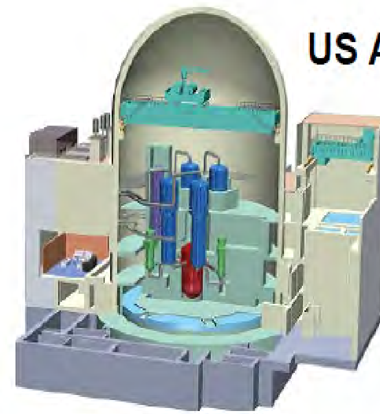


NuScale

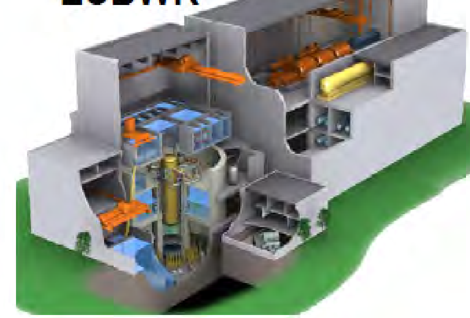
APR-1400



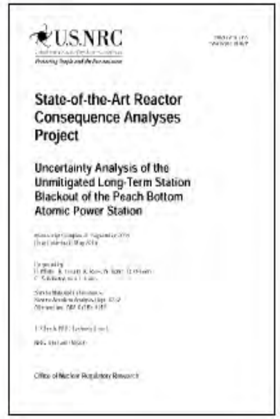
US APWR

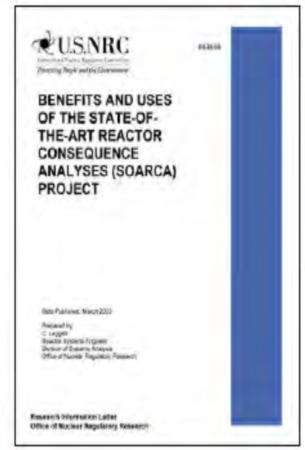


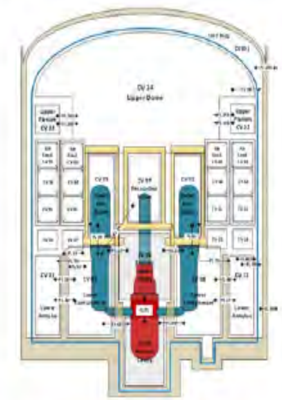
ESBWR

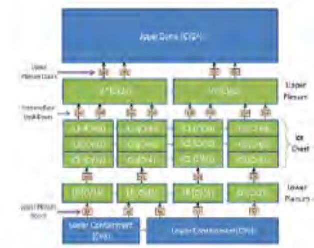


State-of-the-Art Reactor Consequence Analysis (SOARCA)









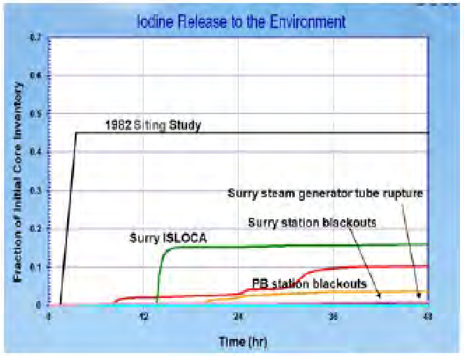
Project Start

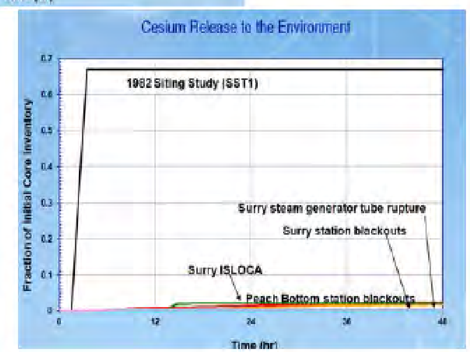
SECY-05-0233

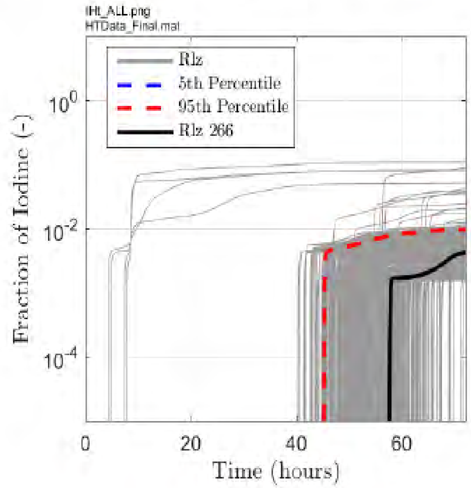
Uncertainty Analysis

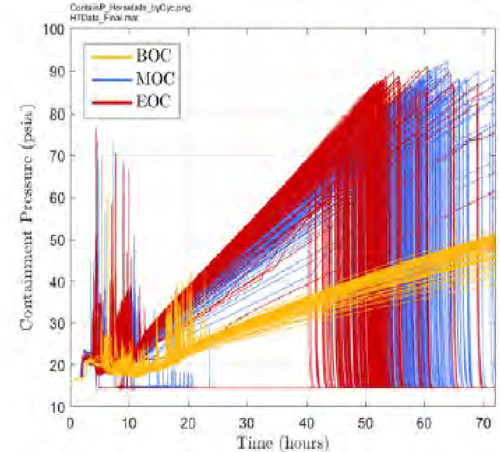
2006 2012 2016 2019 2020

State-of-the-Art Reactor Consequence Analysis (SOARCA)





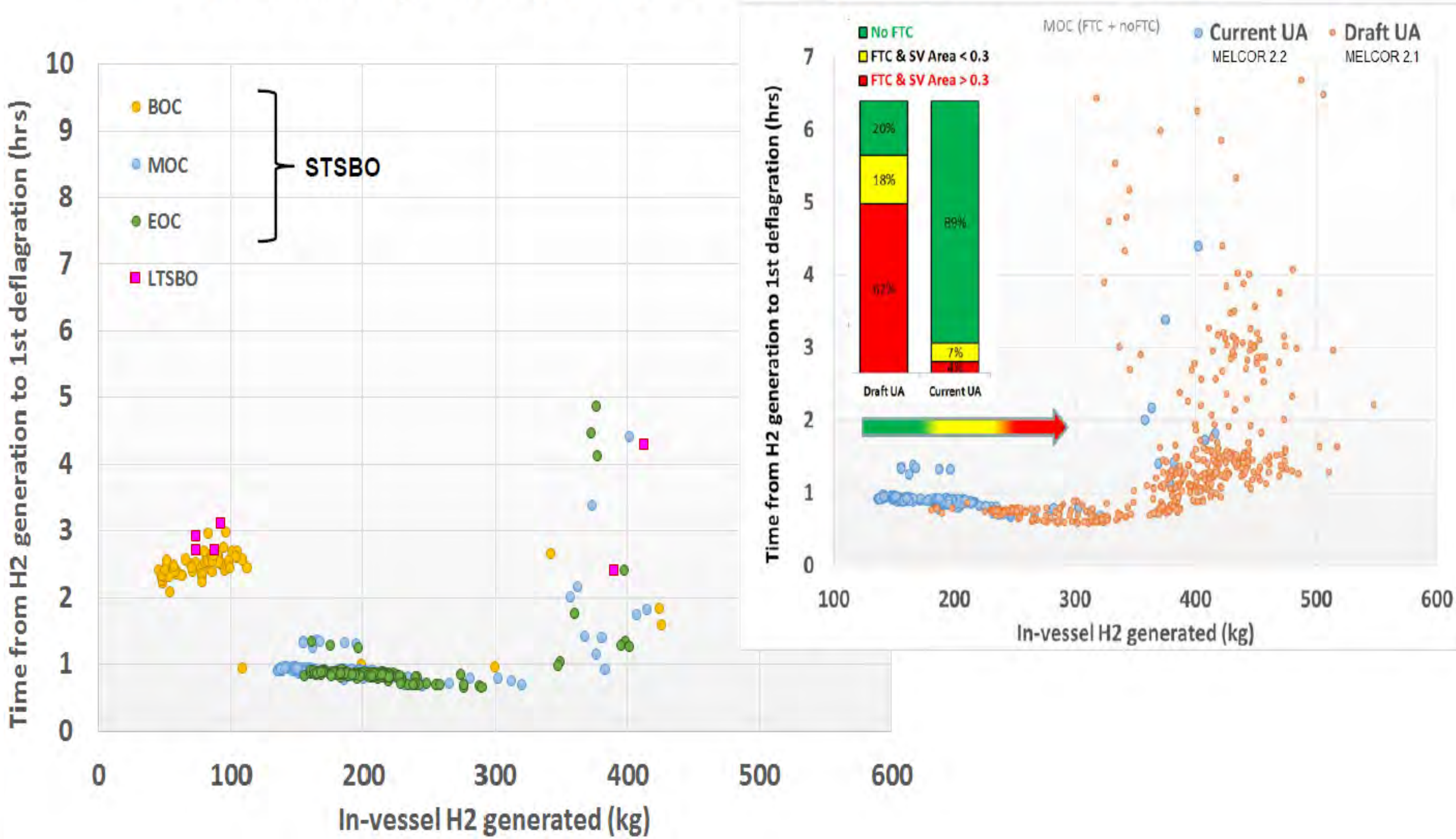




Uncertainty Analysis

2006 2012 2016 2019 2020

Uncertainty Analysis (SOARCA) NUREG/CR-7245



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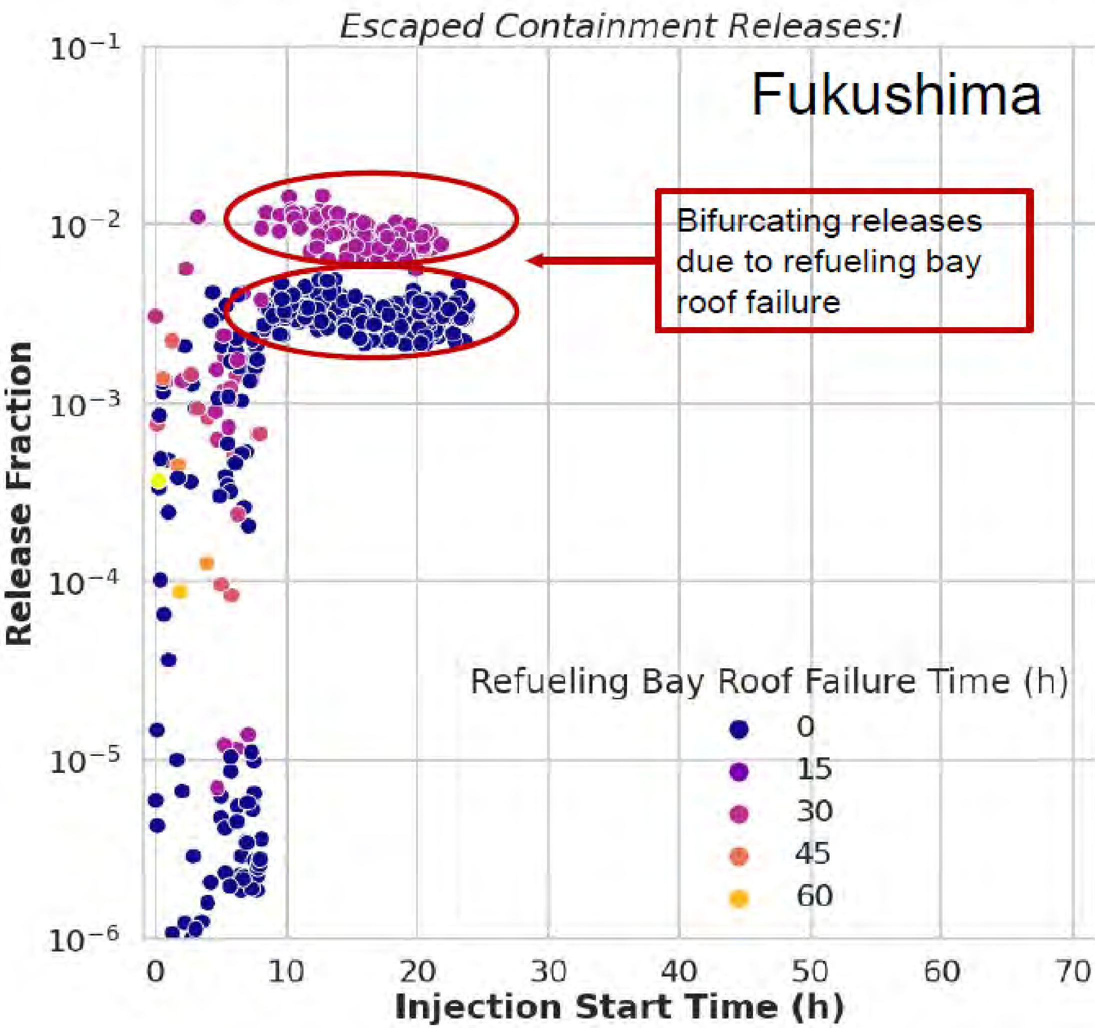
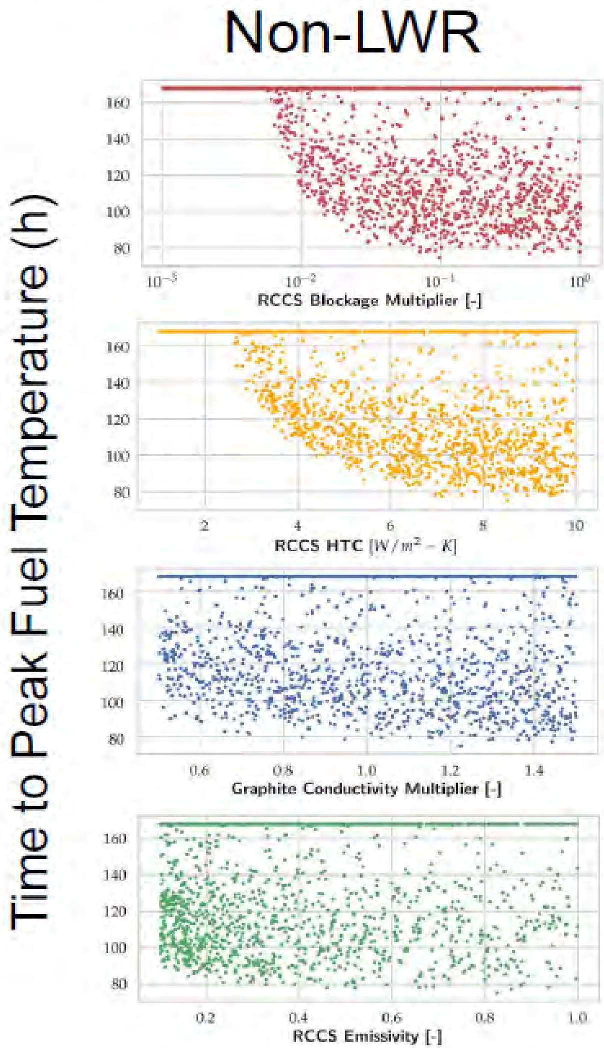
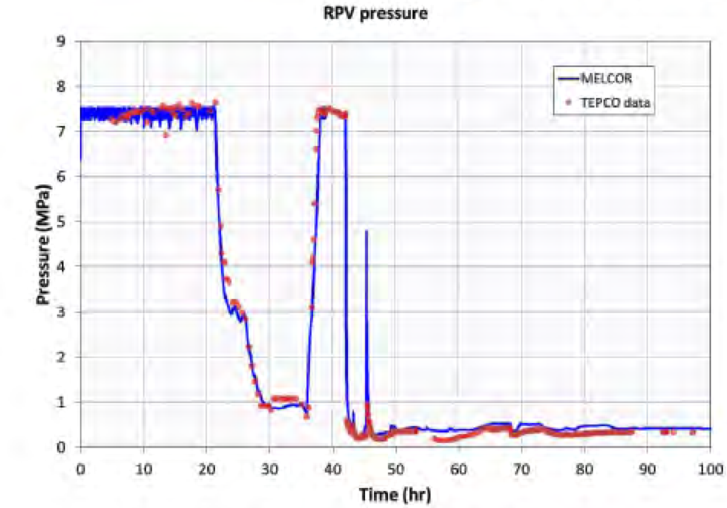
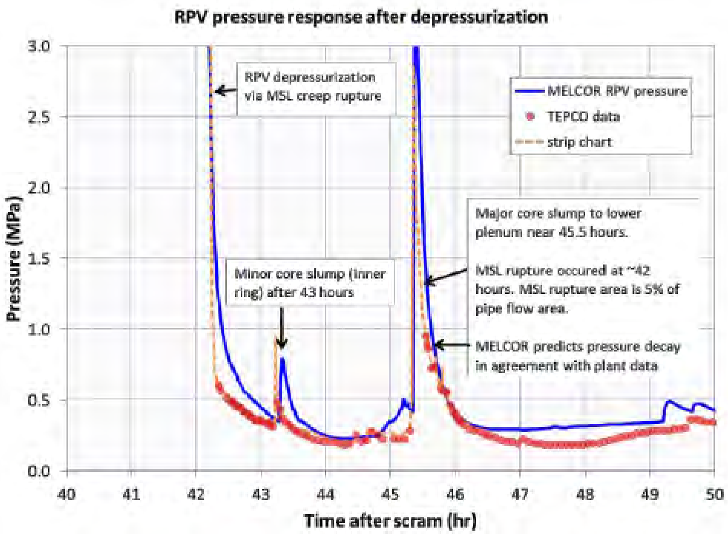
Uncertainty Analysis Applications

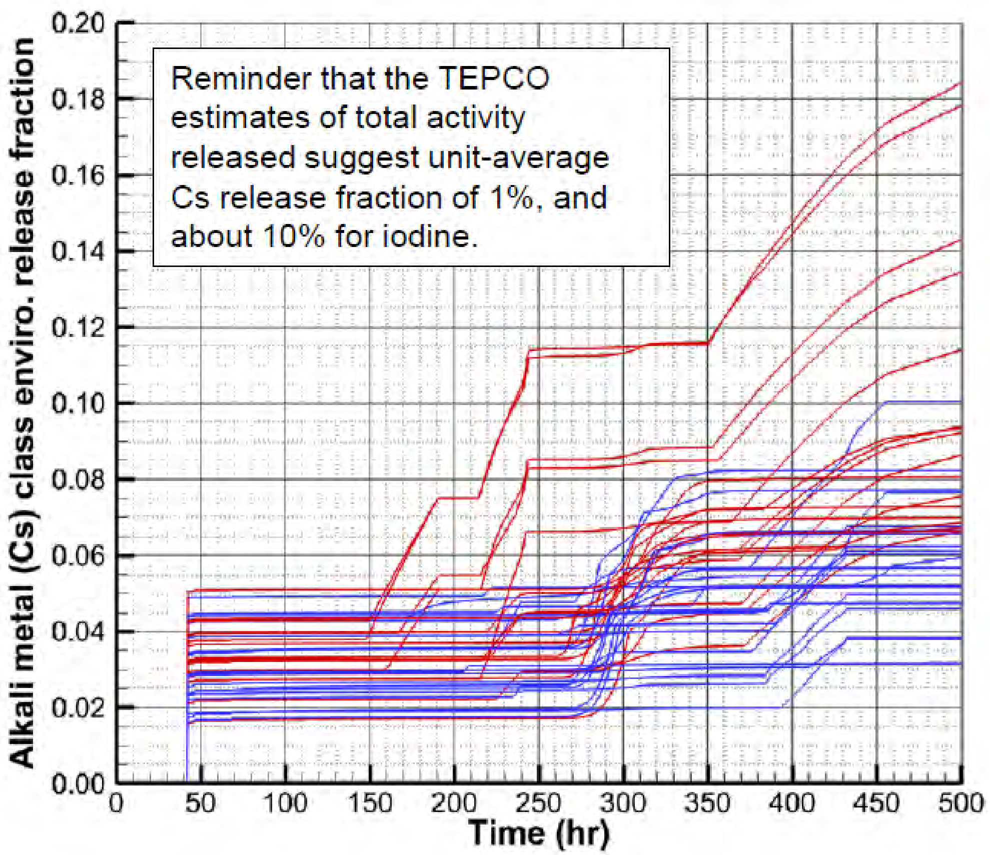




Fukushima Forensics (Unit 3)







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MELCOR Spent Fuel Pool Modeling

Convective Heat Transfer Surfaces:

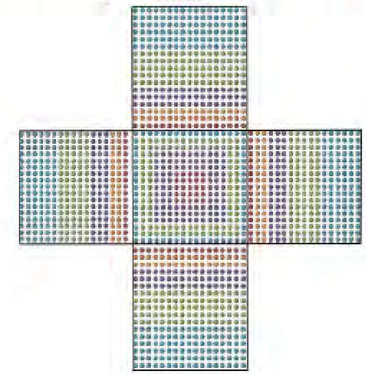
Ring 1 — Clad, Canister / Water Rods, Rack | Ring 2 — Clad, Canister / Water Rods, Racks

Radiative Heat Transfer Flow Path:

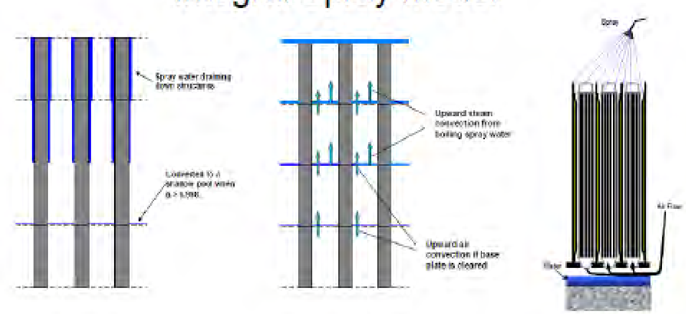
Ring 1: Fuel → Clad → Canister / Water Rods → Rack → | → Canister / Water Rods → Clad → Fuel :Ring 2

Canister / Water Rods → Racks

Multiple fuel rod components in the center assembly (Ring 1) and four peripheral assemblies (Ring 2)



Integral Spray Model



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MELCOR SFP Modeling Basis

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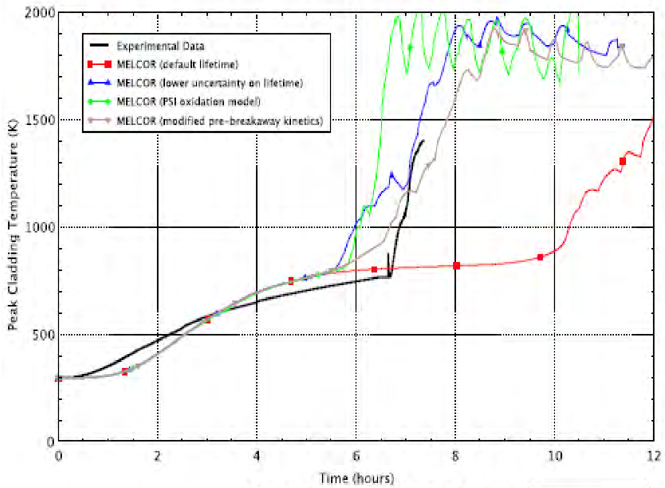
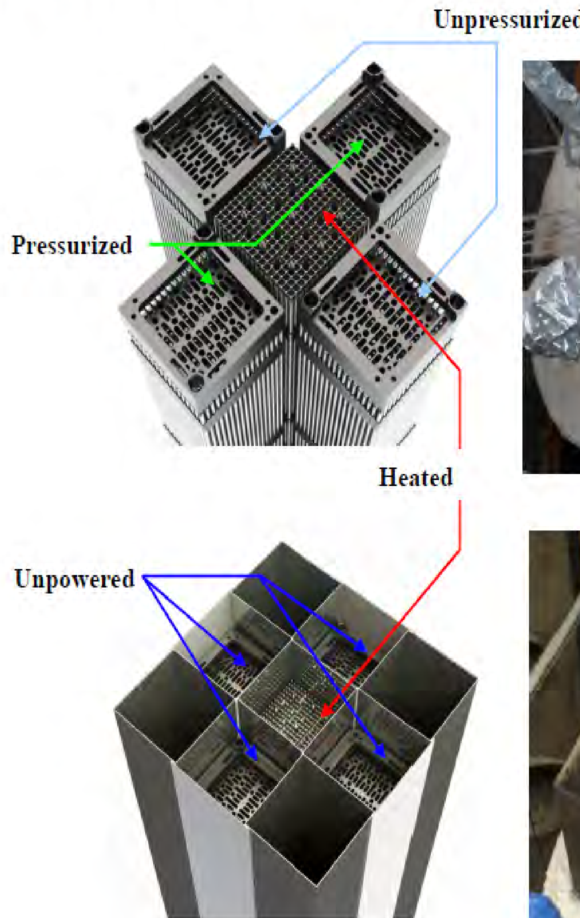
NUREG/CR-7216

Spent Fuel Pool Project Phase II:
Pre-Ignition and Ignition Testing of a 1x4 Commercial 17x17 Pressurized Water Reactor Spent Fuel Assemblies under Complete Loss of Coolant Accident Conditions



Office of Nuclear Regulatory Research



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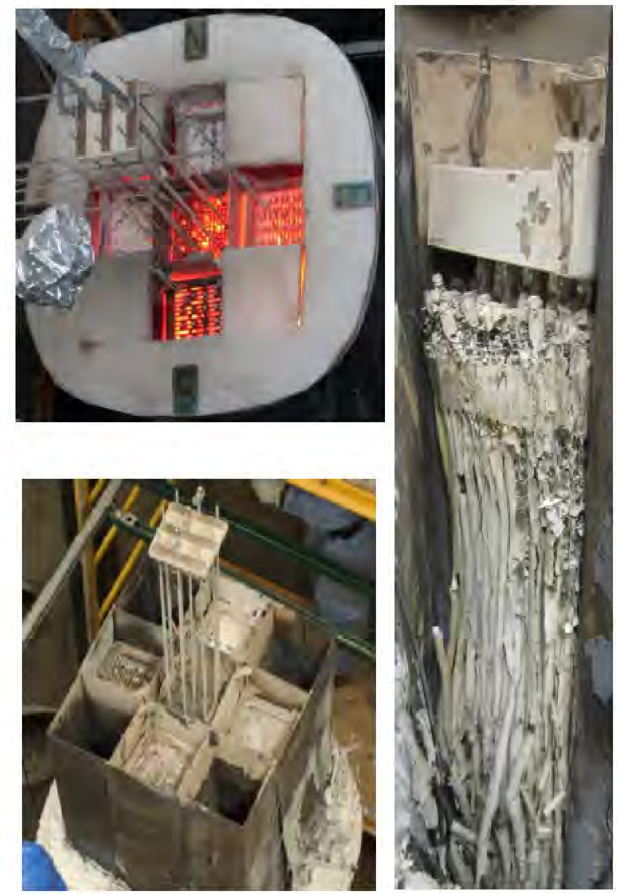

After

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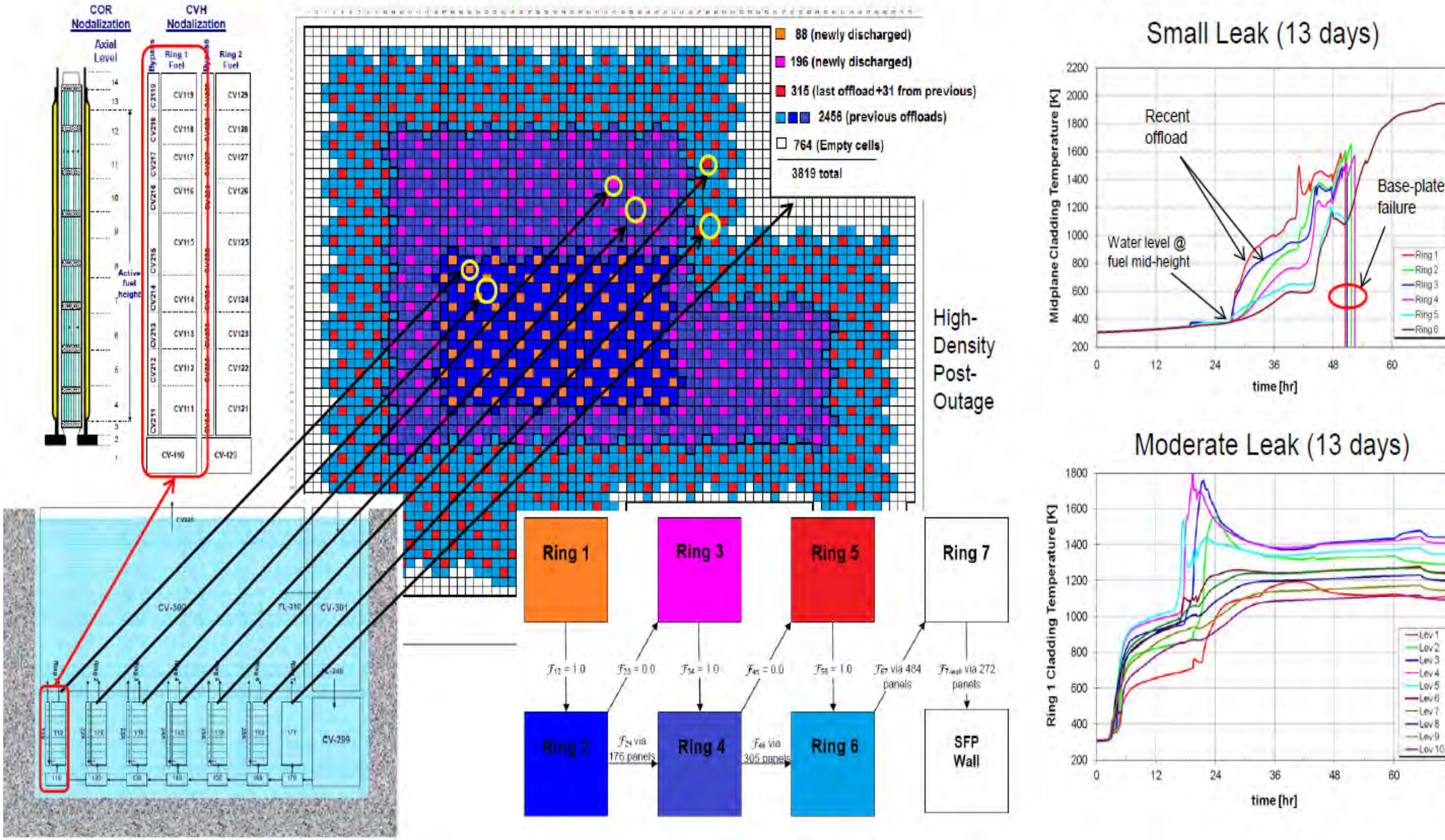
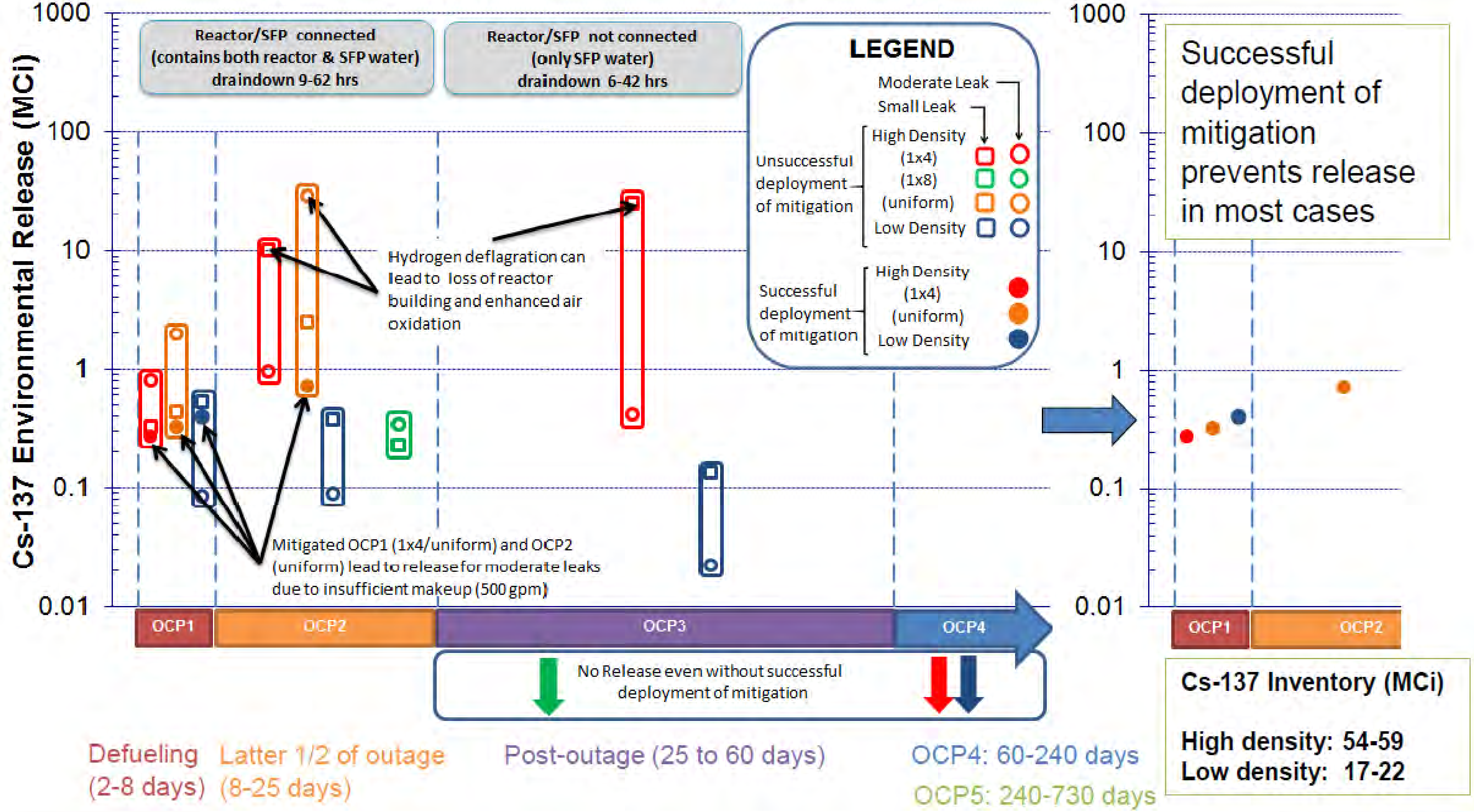
MELCOR SFP Model (NUREG-2161)



BREAK

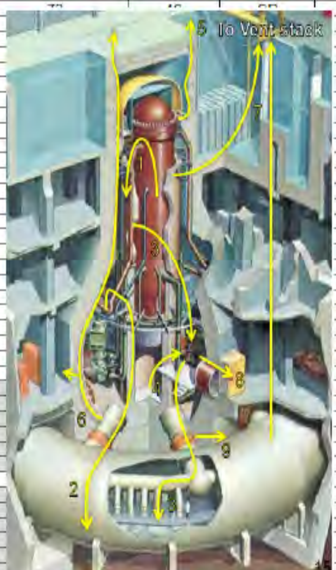
MELCOR Results (NUREG-2161)

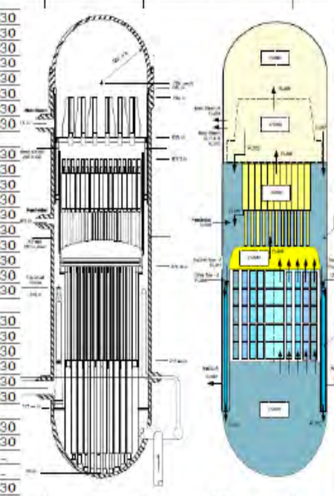
Cases that lead to release (OCP1/2/3)

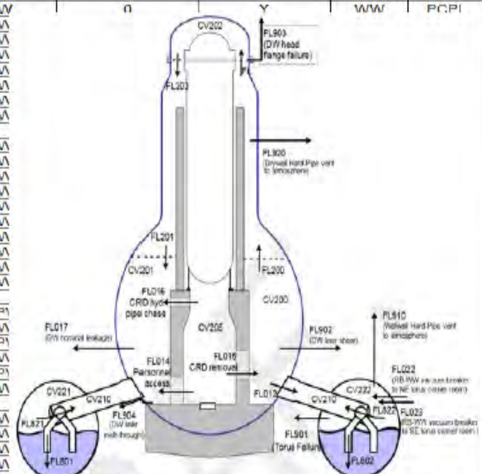


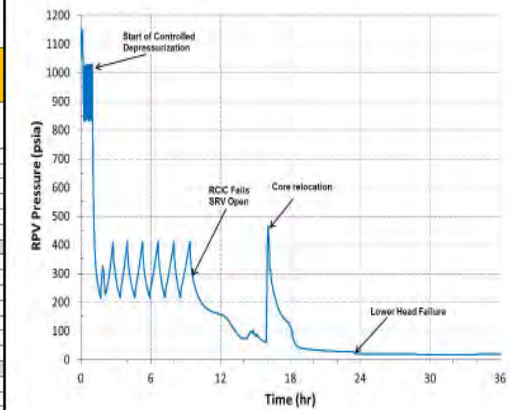
Containment Protection and Release Reduction (NUREG-2206)

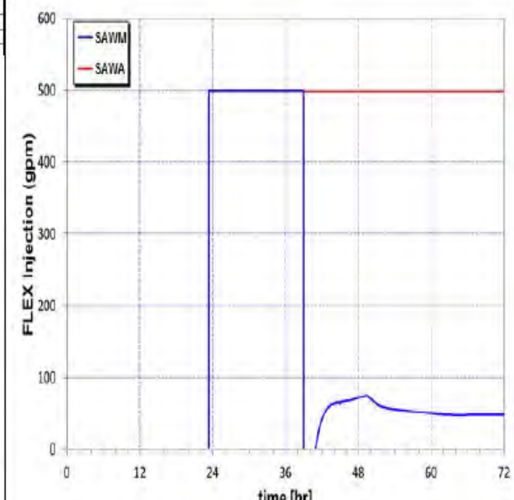
		Pre Core Damage						Post Core Damage				
		RPV Pressure control	RCIC Operation				Anticipatory Venting	Flex Operation		SRV Operation	Venting	
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	WW Level Control Injection @ 21' (gpm)	Allow SRV stuck open failure?	Location	Setpoint (psig)
Option	Case											
1/2A	1	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	1S1	72	16	SP	230	N	5	-	-	Y	WW	PCPL
1/2A	2	72	16	SP	230	N	15	-	-	Y	WW	PCPL
1/2A	3	4	4	SP	230	N	15	-	-	N	WW	PCPL
1/2A	4	72	16	SP	240	N	15	-	-	Y	WW	PCPL
1/2A	5	72	16	CST	230	N	15	-	-	Y	WW	PCPL
1/2A	6	72	16	SP	230	N	15	-	-	Y	WW	PSP
3A	7	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
	7dw	72	16	SP	230	N	15	RPV	0	Y	DW	PCPL
3A	10	72	16	SP	230	N	15	RPV	500	Y	WW/DW	PCPL
3A	11	72	16	SP	230	Y	15	RPV	500	Y	WW/DW	PCPL
4Aii(1)	8	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	9	72	16	SP	230	Y	15	RPV	throttle	Y	WW	PCPL
4Aii(1)	12	72	16	SP	230	N	15	RPV	throttle	Y	WW	PSP
4Aii(1)	13	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	14	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL
4Aiii(1)	15				230	N	15	RPV	throttle	Y	WW	PCPL
4Ai(1)	18				230	N	15	RPV	500	Y	WW/DW	PCPL
4Ai(1)	16				230	N	15	RPV	500	Y	WW/DW	PCPL
3B	21				230			DW	0	Y	WW	PCPL
3B	24				230			DW				
	24dw				230			DW				
4Aii(2)	22				230			DW				
	22dw				230			DW				
4Aii(2)	23				230			DW				
4Aii(2)	25				230			DW				
3B	26				230			DW				
4Ai(2)	27				230			DW				
4Aiii(2)	28				230			DW				
	28dw				230			DW				
4Ai(2)	32				230			DW				
4Ai(2)	30				230			DW				
	30dw				230			DW				
4Aiii(2)	29				230			DW				
	29dw				230			DW				
4Ai(2)	31				230			DW				
	31dw				230			DW				
3A	41				230			RPV				
3B	43				230			DW				
3A	42				230			RPV				
3B	44				230			DW				
4Aii(1)	47				230			RPV				
4Aii(2)	48				230			DW				
3B	45				230			DW				
3B	46				230			DW				
3B	49				-			DW				
4Ai(2)	50				-			DW				
3B	51				230			DW				
3B	52				230	-	15	DW	500	N	DW	15
3B	53	-	16	SP	230	-	15	DW	500	Y	DW	15



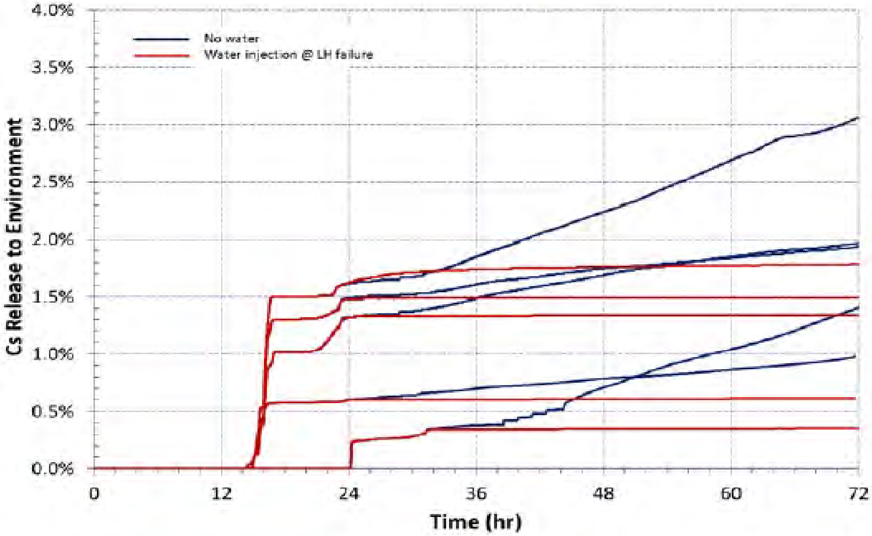






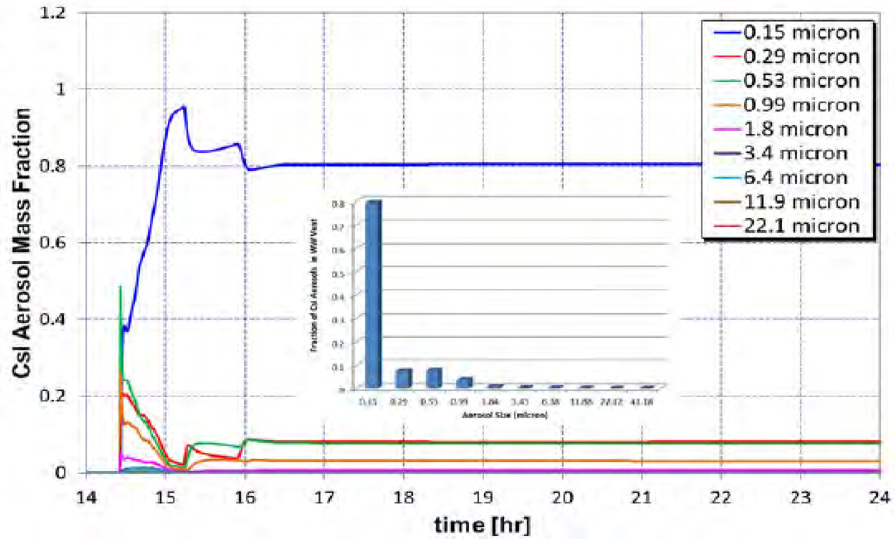


Containment Protection and Release Reduction (NUREG-2206) - Mark I Results



Water addition at lower head failure has the benefit of mitigating further release, but does not affect the release at the time of venting

Particle size distribution dominated by very small aerosols at the time of venting



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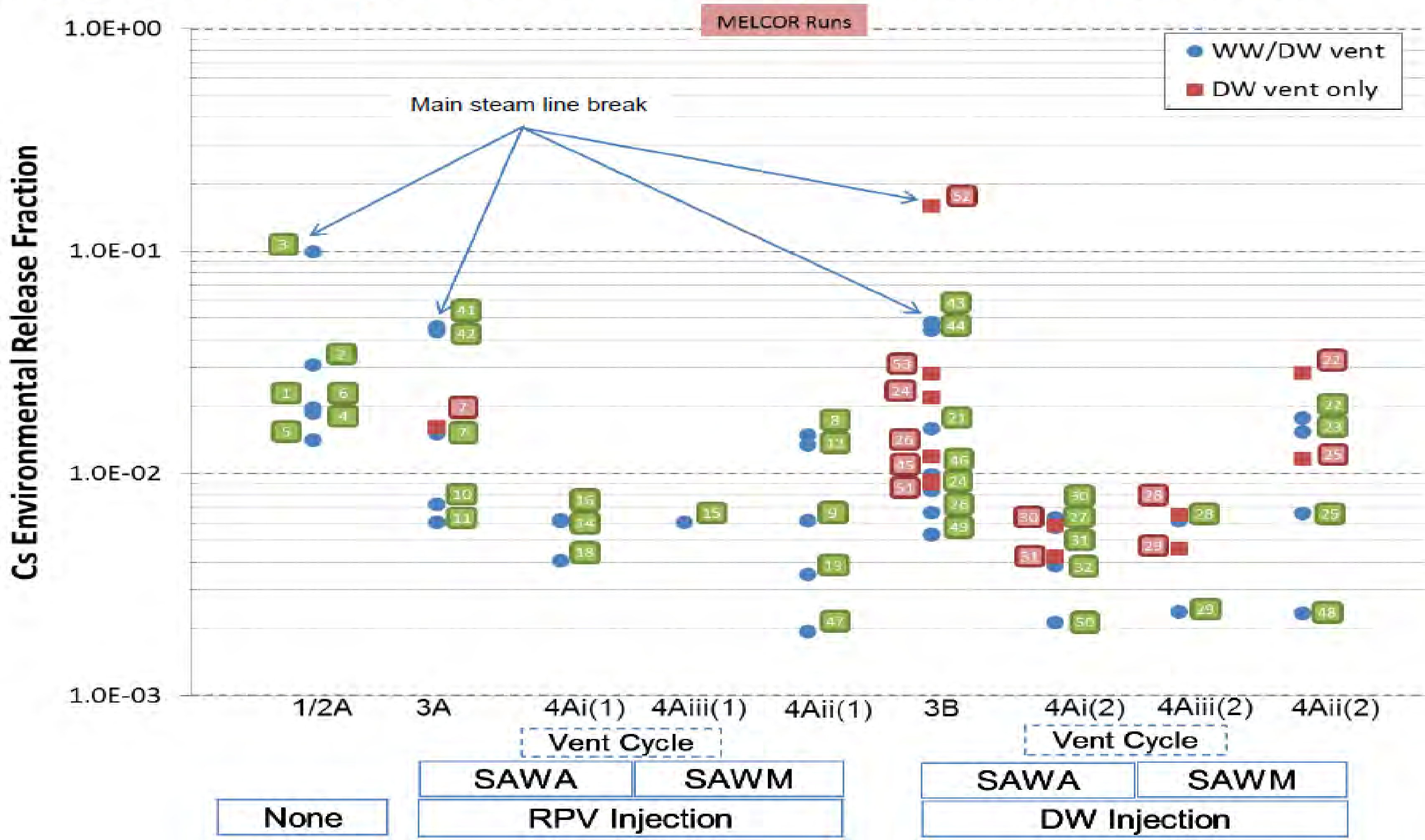
Containment Protection and Release Reduction (NUREG-2206) - Mark I Results



Summary

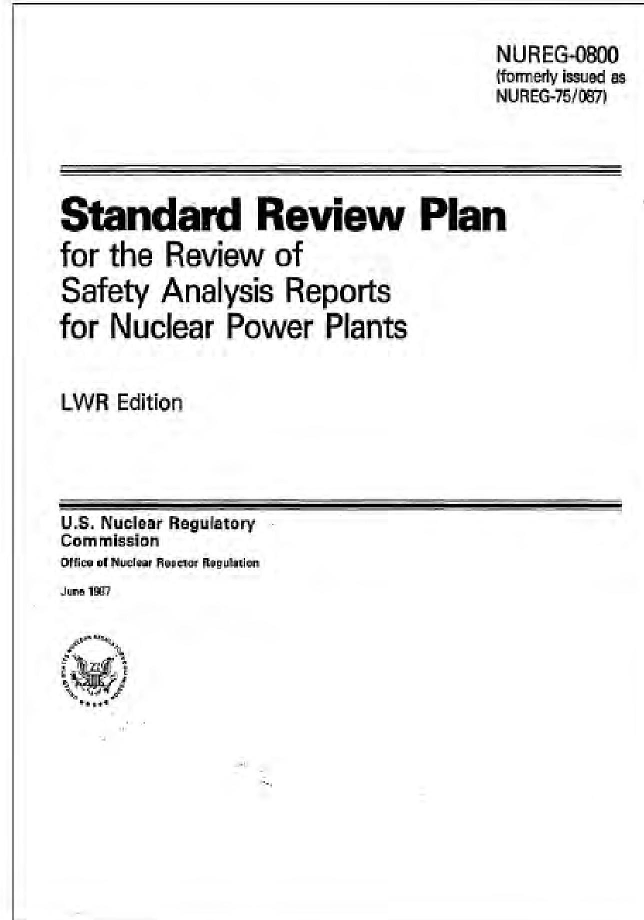
- Decades of experimental and analytical research in severe accident progression and source term
- Validated state-of-practice MELCOR code ready for application to a wide variety of nuclear technologies including advanced designs
- MELCOR has been an essential tool for resolving safety issues and informing regulatory decision making

MELCOR application to new reactors

Standard Review Plan

Staff independent analysis

- Independent assessment of plant response and source term
- Scenarios from the PRA
- Engage with the applicant to resolve differences with the applicant's analysis


NUREG-0800
(formerly issued as
NUREG-75/087)

Standard Review Plan
for the Review of
Safety Analysis Reports
for Nuclear Power Plants

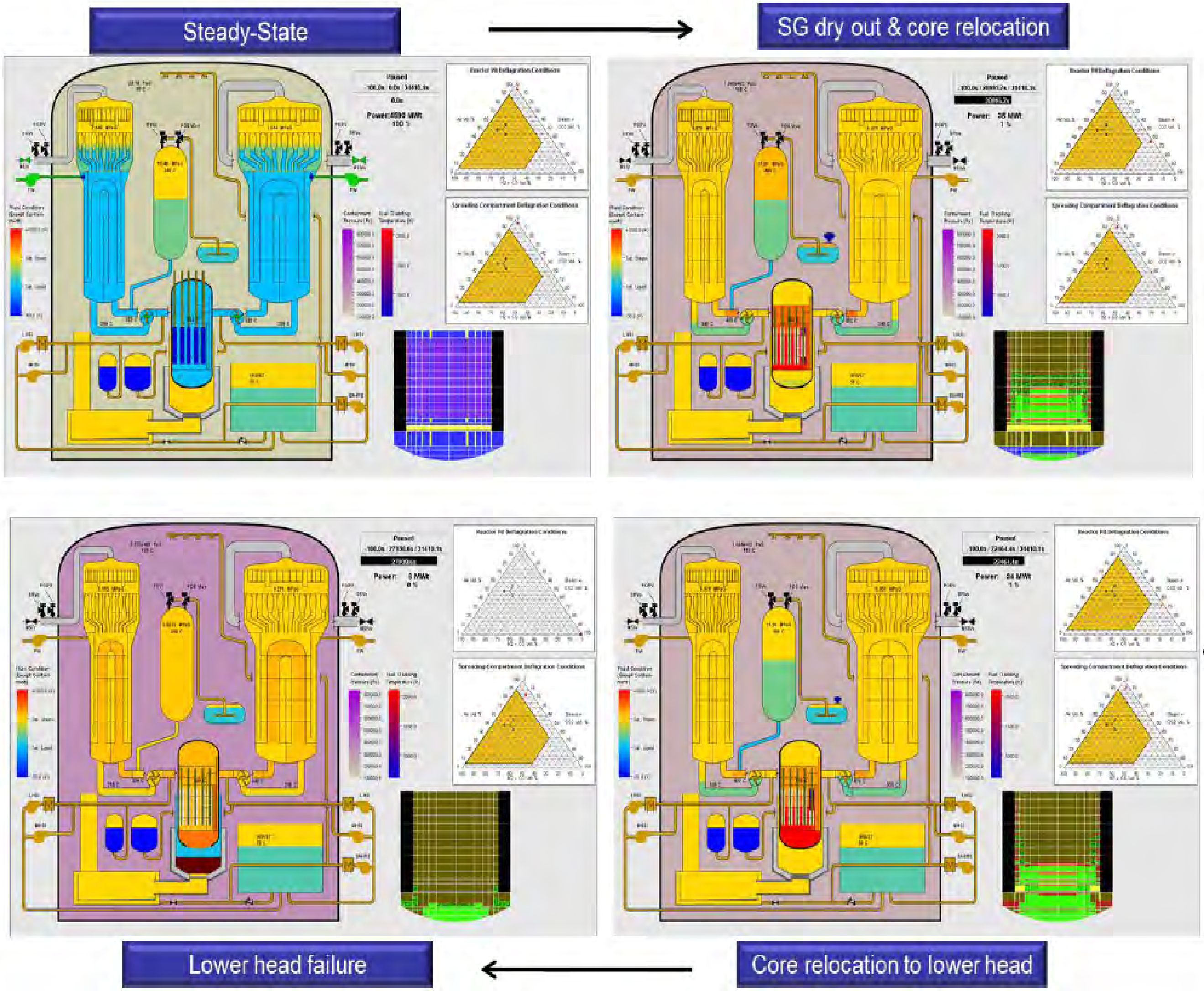
LWR Edition

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

June 1987

Large LWRs

- ABWR
- AP-600
- System 80+
- AP-1000
- EPR
- APWR
- ESBWR
- APR-1400



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SMRs

- NuScale
- mPower
- Westinghouse SMR
- BWRX-300

RES/FSCB 2019-01

Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor

S. Campbell
H. Esmaili
J. Schaperow[1]

April 2019

Fuel and Source Term Code Development Branch
Division of Systems Analysis
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

[1] PRA and Severe Accidents Branch, Division of Safety Systems and Risk Assessment, Office of New Reactors, U.S. NRC

NuScale

- Applicant-developed source term for demonstrating EAB/LPZ dose criteria met
  - Replaced RG 1.183 source term
  - MELCOR, STARNAUA
- NRC independent analysis
  - MELCOR, RADTRAD

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Accident Source Term Methodology Topical Report Staff Review

NuScale Design Certification Application Review

Presentation to the ACRS Subcommittee

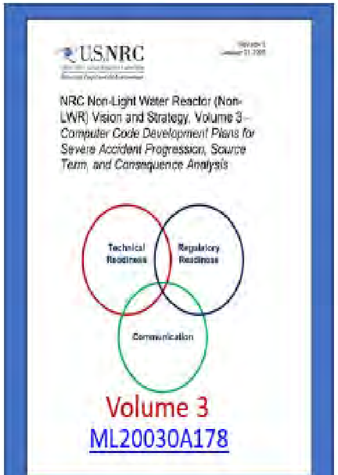
November 20, 2019

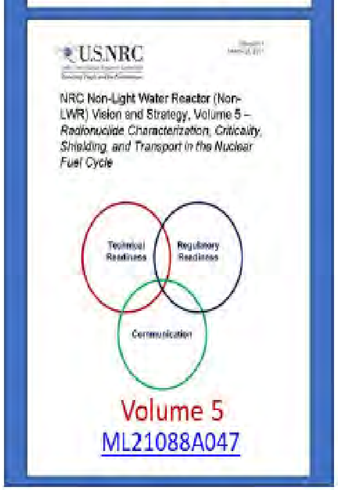
SCALE/MELCOR non-LWR source term demonstration project

Outline

- NRC strategy for non-LWR source term analysis
- Project objectives
- Public workshops
- Sample results
  - Heat pipe reactor (HPR)
  - High-temperature gas-cooled reactor (HTGR)
  - Pebble-bed molten-salt-cooled reactor (FHR)
- Summary

NRC strategy for severe accident analysis

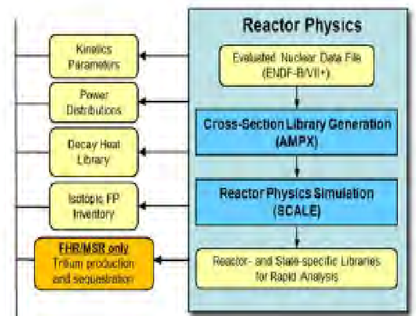


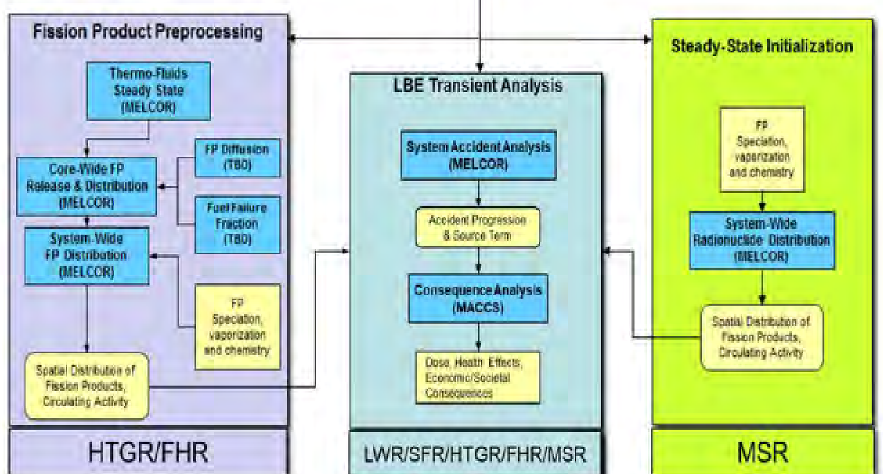


Evaluation Model and Suite of Codes

Code strategy for source term

"NRC Non-Light Water Reactor Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, January 2020, ML20030A178





Project objectives

- Understand severe accident behavior and provide insights for regulatory guidance
- Facilitate dialogue on staff's approach for accident progression and source term
- Demonstrate use of SCALE and MELCOR
  - Identify accident characteristics and uncertainties
  - Develop publicly available input models for representative designs

Scope

Full-plant models for representative non-LWRs

- Heat pipe reactor – INL Design A
- High-temperature gas-cooled reactor – PBMR-400
- Pebble-bed molten-salt-cooled – UCB Mark 1
- Molten-salt-fueled reactor – MSRE
- Sodium-cooled fast reactor – ABTR

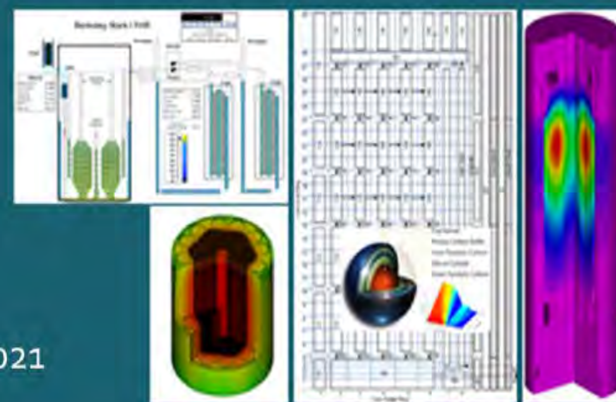
Approach

1. Use SCALE to calculate core decay heat, radionuclide inventory, reactivity feedback
2. Build MELCOR full-plant input model
3. Select accident scenarios
4. Perform MELCOR simulations for the selected scenarios
5. Public workshops to discuss the modeling and sample results

Public Workshops

Public Workshop: SCALE/ MELCOR Non LWR Source Term Demonstration Project

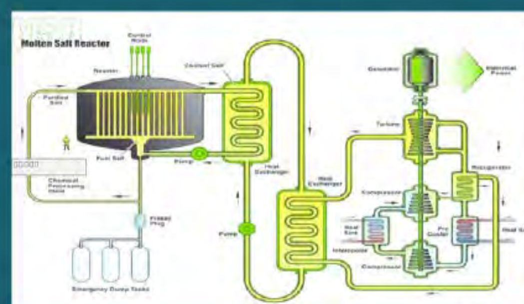
Heat pipe reactor – June 29, 2021
Gas cooled reactor – July 20, 2021
Pebble bed molten-salt-cooled reactor – Sept 14, 2021

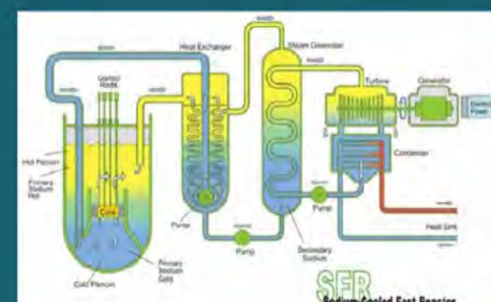


For More Information

Coming in 2022

Molten-salt fueled reactor
Sodium-cooled fast reactor



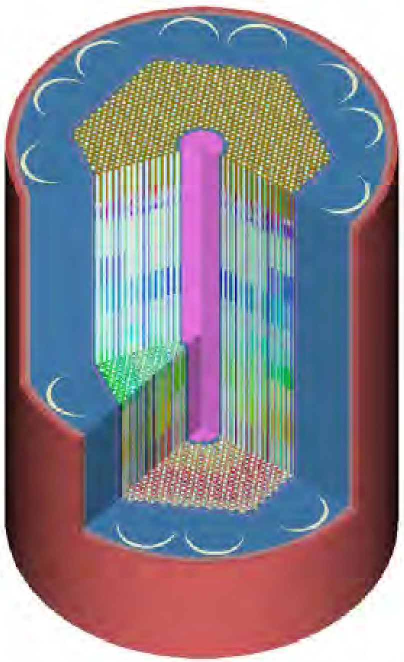


https://www.nrc.gov/reactors/new-reactors/advanced/details.html#non-lwr-ana-code-dev

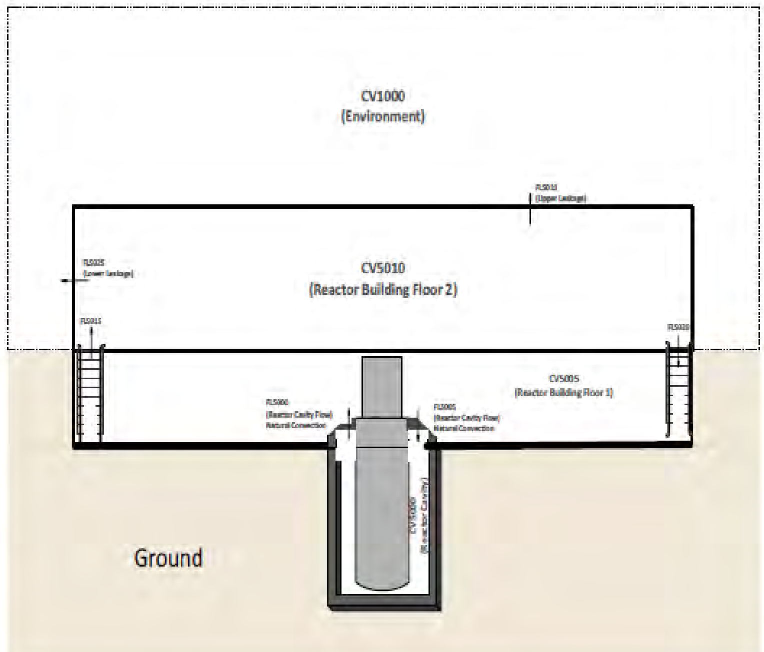
Sample Results

calculations by ORNL and SNL

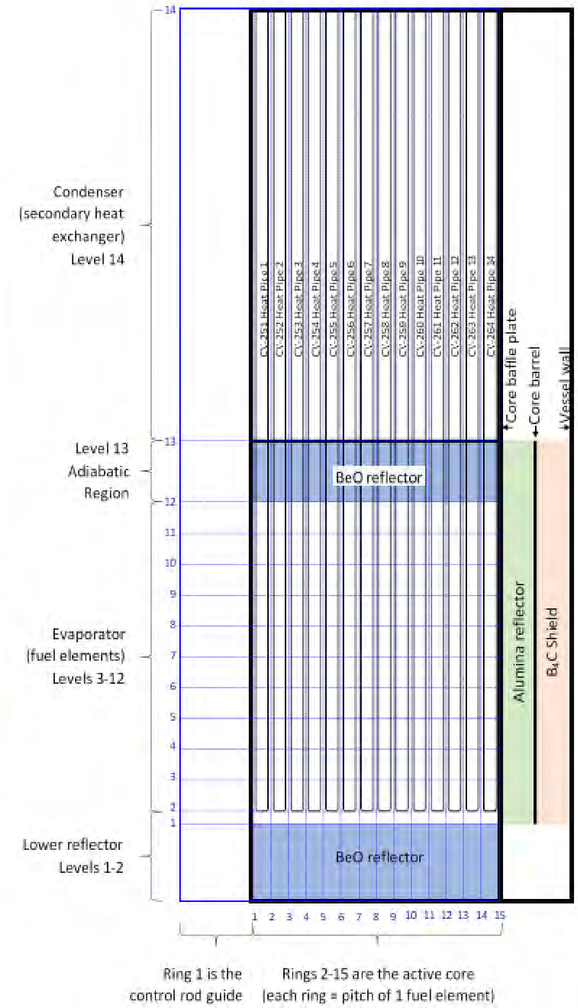
HPR model (INL Design A)



SCALE Model



MELCOR Model



HPR – reactivity addition accident with delayed scram

The control drums start rotating at t=0 sec, which leads to an increase in the core power over 0.9 hr

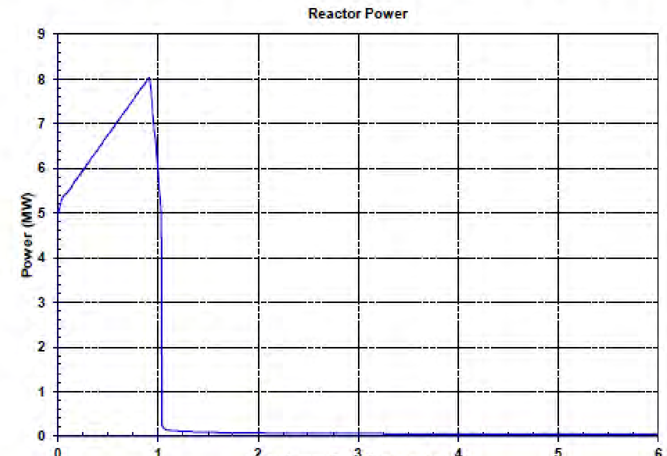
- Negative fuel temperature reactivity feedback limits the rate of power increase

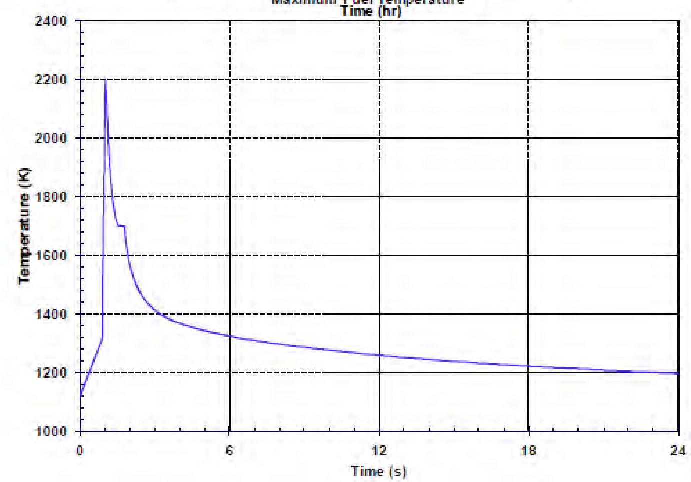
The core steadily heats until the maximum heat flux location reaches the boiling limit

- The heat transfer rate is limited above the boiling limit, which leads to a rapid heatup rate
- The SS cladding is assumed to fail at 1650 K (just below its melting point), which starts the fission product releases into the reactor
- The reactor is assumed to trip at 2200 K

Radial heat dissipation and heat loss to the reactor cavity passively cools the core

- No active heat removal (secondary system trips and isolates)





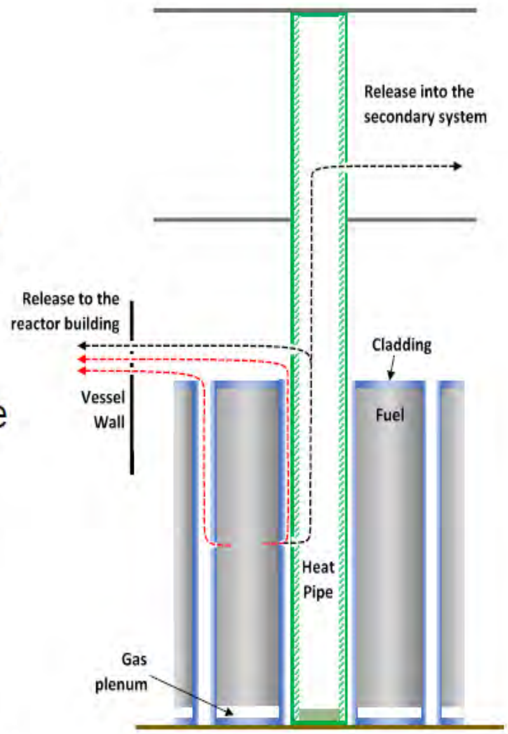
HPR – reactivity addition accident with delayed scram

Cladding failure at 1650 K resulting in fission product release

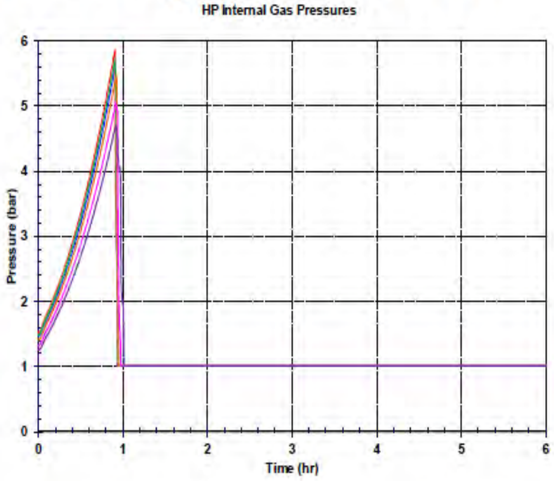
- Heat Pipes (HP) that exceeded the boiling limit rapidly heat to cladding failure (1650 K)
- ~20% of the 1134 HPs and fuel elements failed
- HP depressurization on failure drive release from the vessel

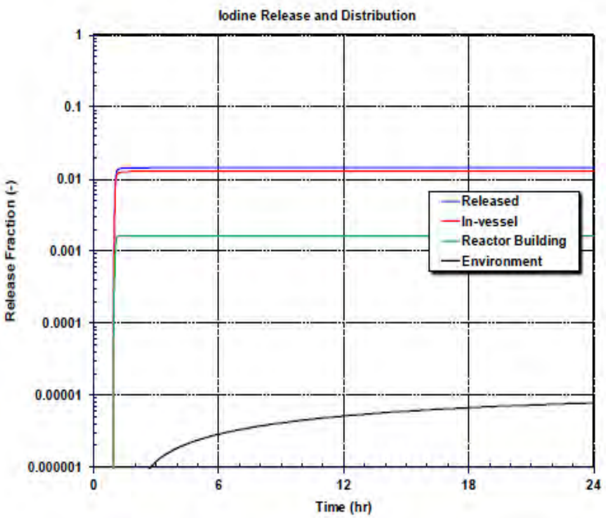
Iodine releases also depend on time at temperature

- Fuel release – 1.4% of core inventory
- Environmental release – 0.0008% of core inventory



- Vessel leakage is 1.6 in^2
- Building leakage is 1.8 in^2





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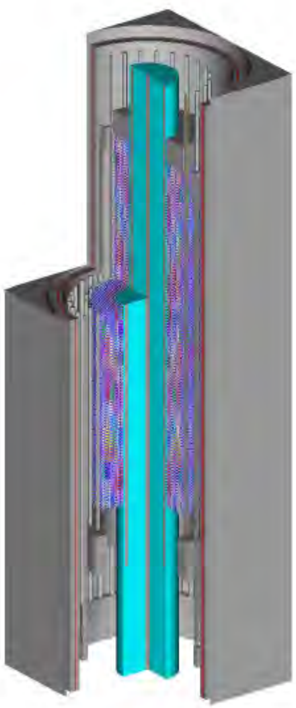
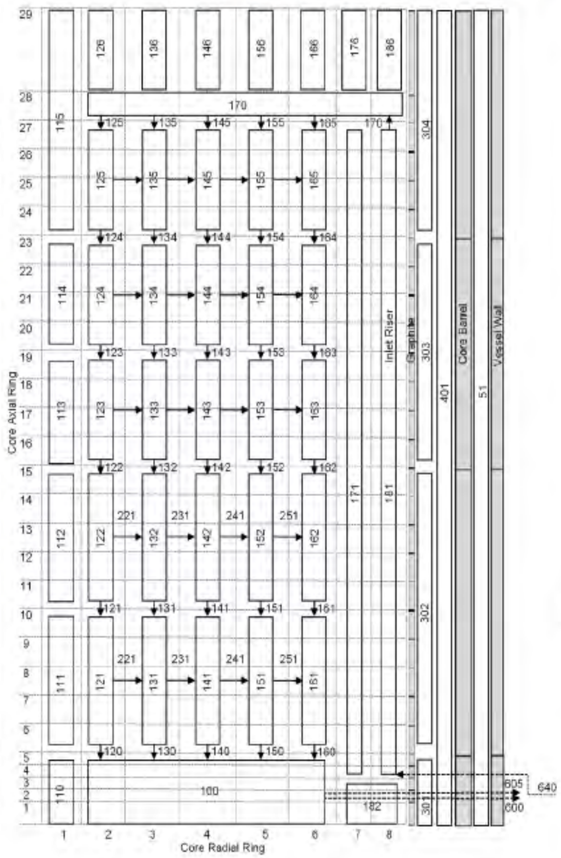
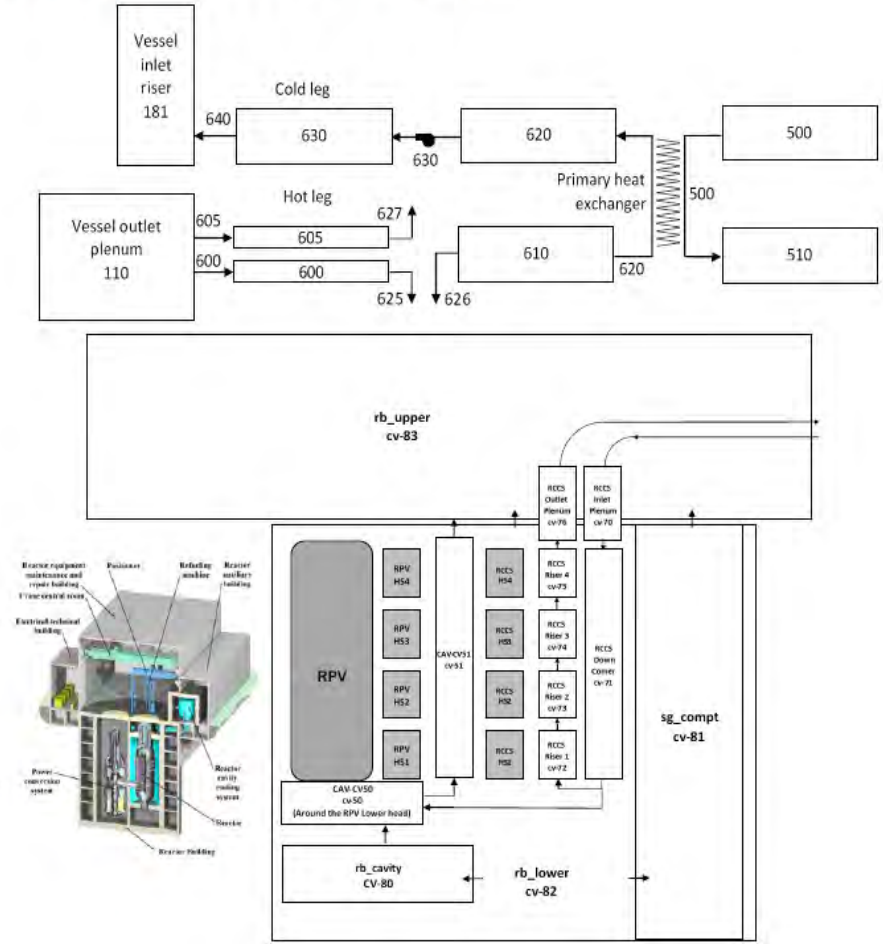
HTGR model (PBMR-400)



SCALE Model



MELCOR Model



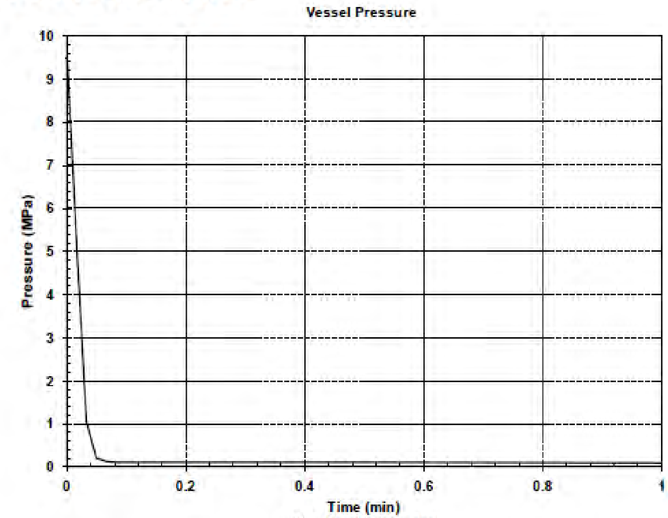
HTGR – loss of coolant accident

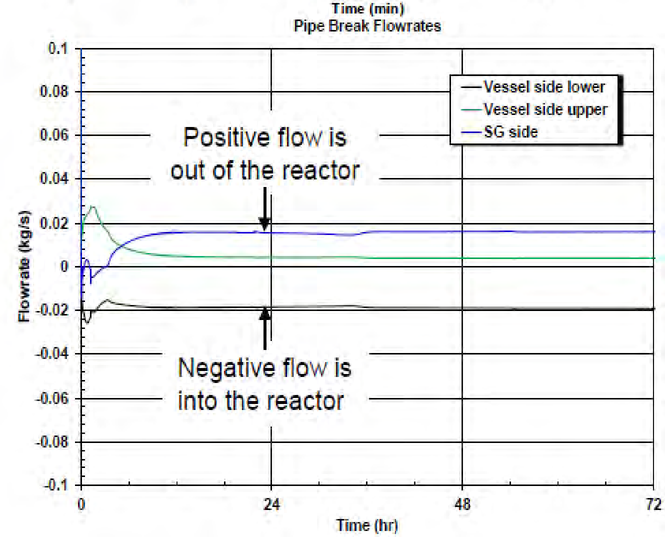
Following pipe break

- Control rods insert to terminate fission
- The vessel depressurizes in seconds as the high-pressure helium escapes out both sides of the broken pipe
- Peak velocity in the pebble bed is 45 m/s (normal flow rate is 11-18 m/s)

Counter-current flow established on the vessel side of the pipe break

- Hot gases from the exit plenum escape on the top side of the broken hot leg pipe and cooler gases enter along the bottom of the pipe





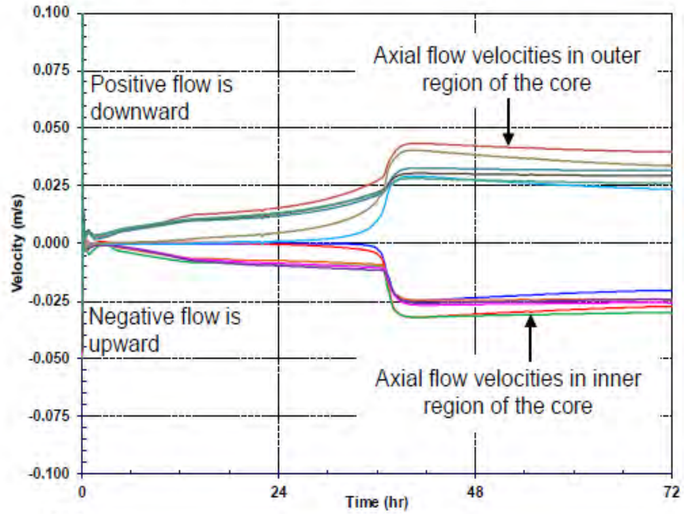
HTGR – loss of coolant accident

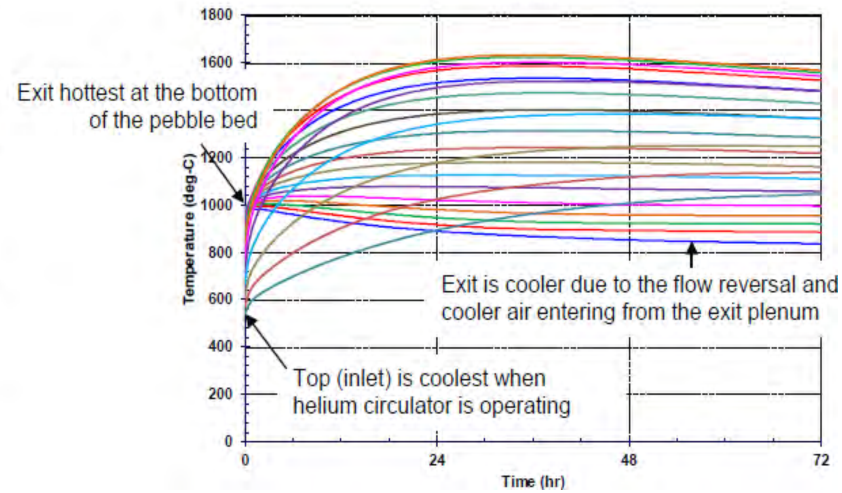
In-vessel natural circulation flow after blowdown

- Upward flow in the inner region of the core where the fuel temperatures and decay power heating are higher
- Downward flow in the outer region of the core where the fuel temperatures and decay power heating are lower
- Flow increases when the fuel starts to cool

The fuel temperatures in the inner region of the pebble bed shift from cooler at inlet and hot at the outlet due to the flow reversal

- The axial fuel temperatures are affected by the local decay heat power (highest in the center) and the flow direction
  - During normal operation, the fuel at the exit (bottom) is the hottest
  - The exit becomes the coolest location (low power and cooler gases entering from the exit plenum)

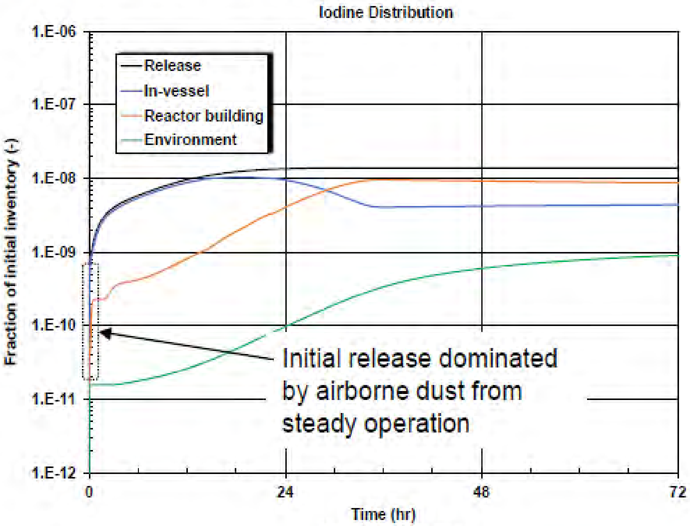


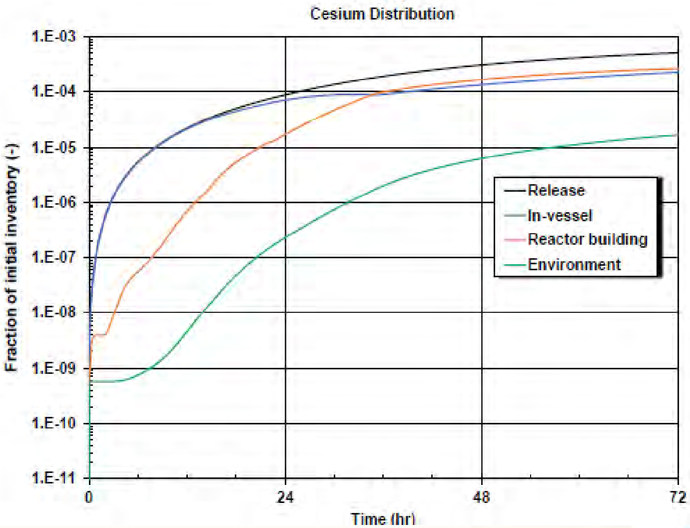


HTGR – loss of coolant accident

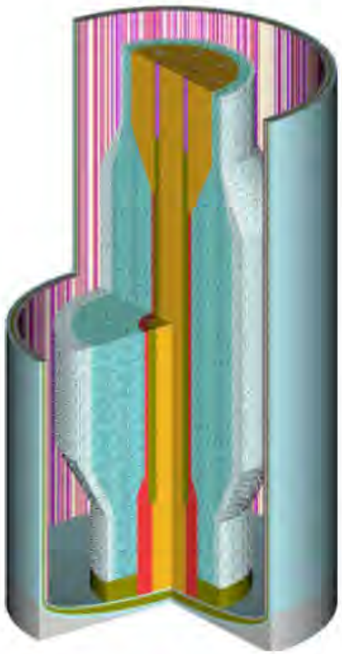
The impact of the low TRISO failure fraction leads to small releases

- Iodine diffusivity assumed to be same as krypton
- Assumes most iodine reacts with cesium
- Larger cesium release due its the higher diffusivity
- Ag release to the environment is $1.2x10^{-3}$ (highest diffusivity)



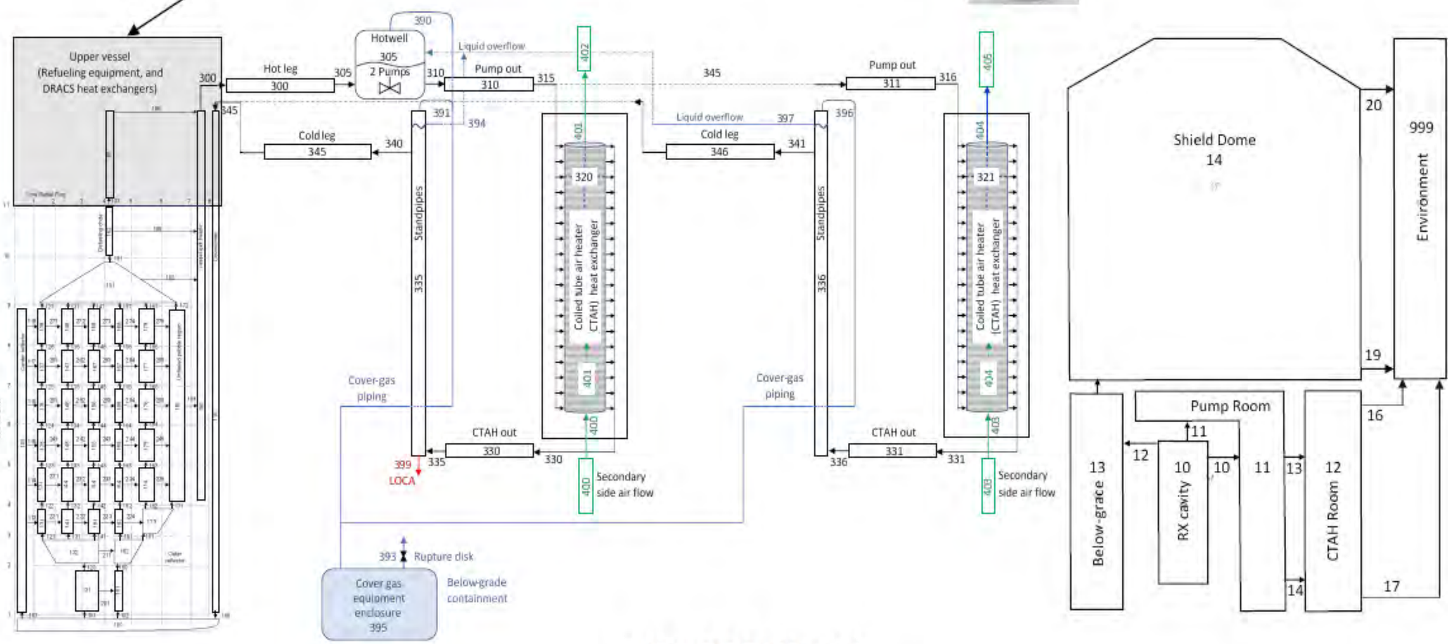


FHR model (UCB Mark 1)



SCALE Model

DRACS (direct reactor auxiliary cooling system)



MELCOR Model

FHR – ATWS

Loss-of-onsite power with failure to SCRAM

- Salt pumps shut off
- Reactor fails to SCRAM
- Secondary heat removal ends
- 0 to 3 trains of DRACS operating

Includes preliminary analysis with xenon transient

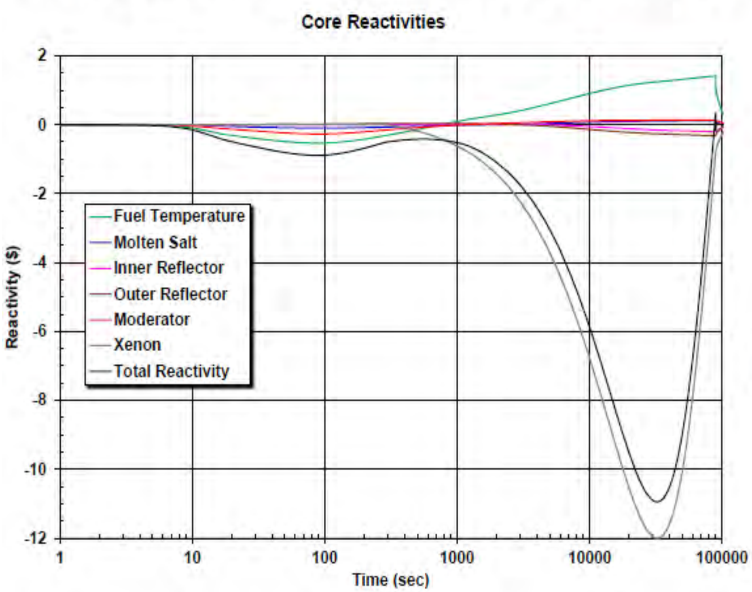
- Guided by ORNL calculations
- Xenon reactivity feedback model being implemented into MELCOR

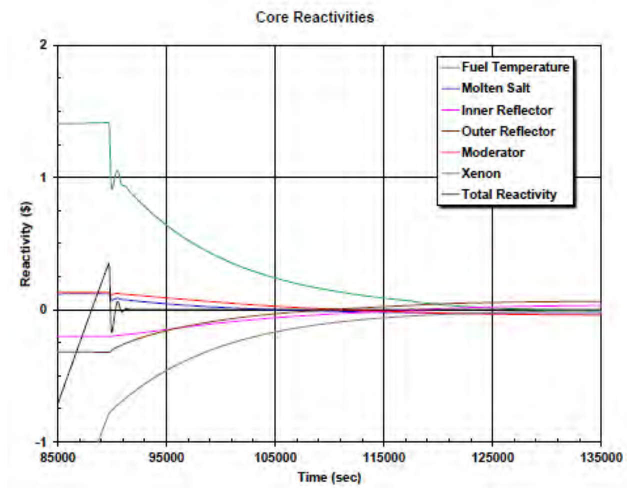
FHR – ATWS

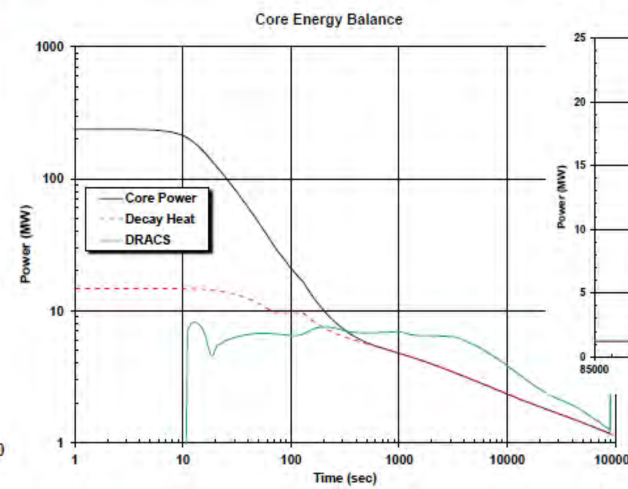
Initial fuel heatup has strong negative fuel and moderator feedback that offsets positive reflector feedbacks

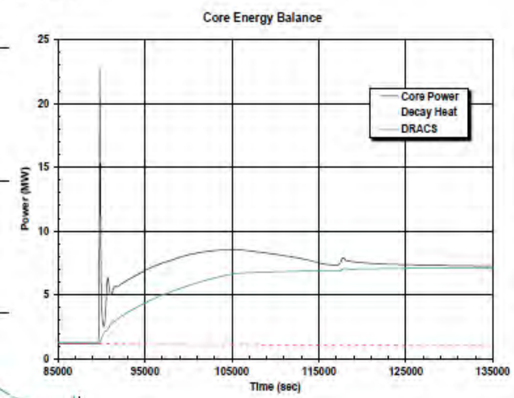
Strong negative xenon transient feedback

3xDRACS exceeds core power after 330 s









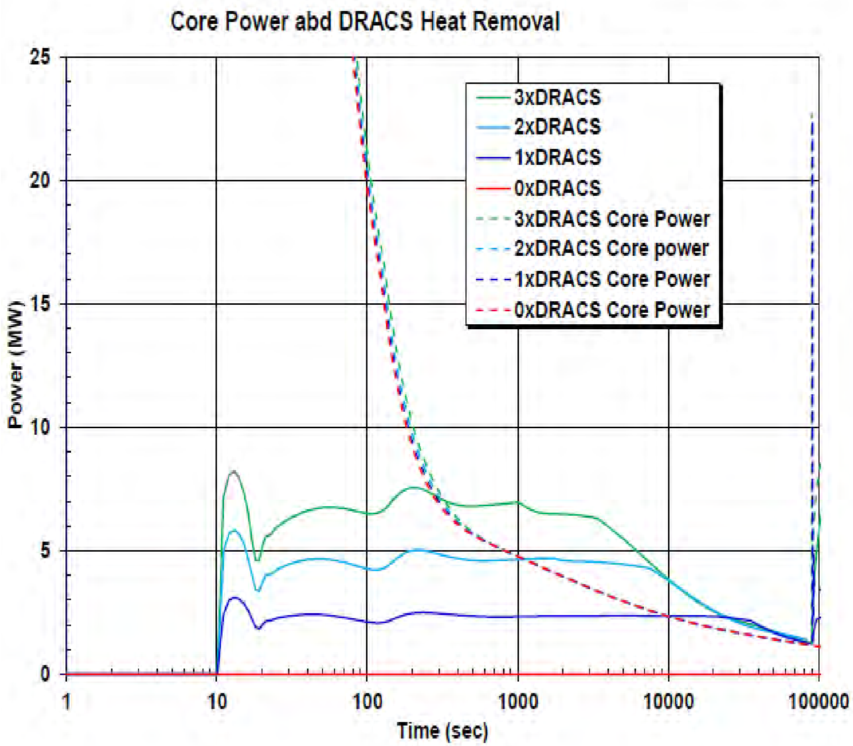
FHR – ATWS with variable DRACS

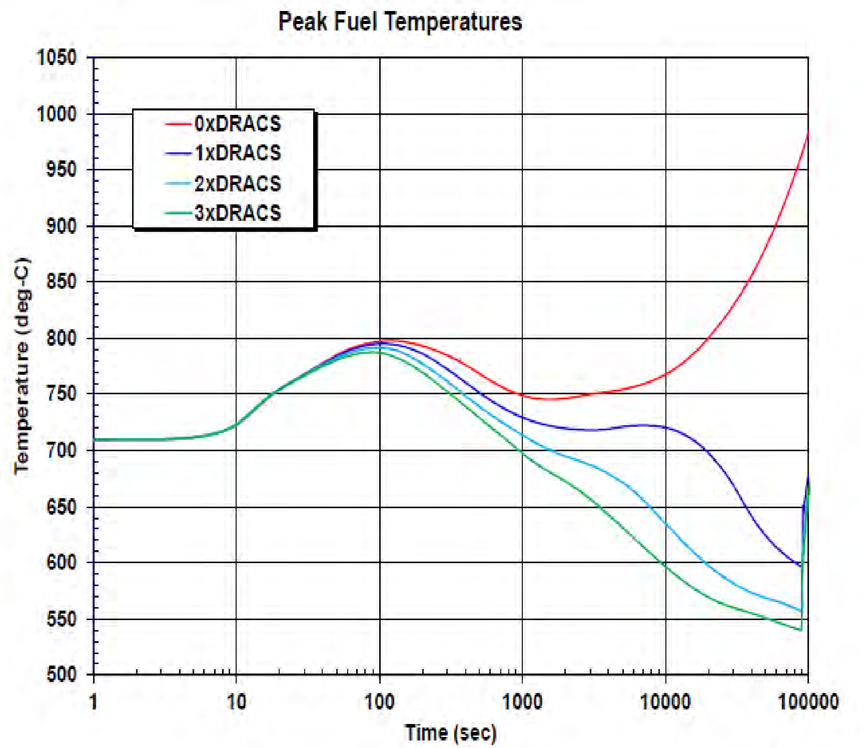
Early power decrease to decay heat level is similar for all cases

- 1xDRACS and 2xDRACS cases exceed decay heat later

Fuel temperatures cool down according to DRACS heat removal rate

- 0xDRACS peak fuel temperature = 990 °C at 10^5 s (T_{sat}~ 1350 °C)





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Summary

- Demonstrated use of SCALE and MELCOR for safety analysis for 3 classes of non-LWRs
  - Working on demonstrations for 2 more classes
- Simulated the entire accident starting with the initiating event
  - system thermal hydraulic response
  - fuel heat-up
  - heat transfer through the reactor to the surroundings
  - radiological release
- Evaluated effectiveness of passive mitigation features

References (www.nrc.gov)(1/2)

- **NUREG-2161**, Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling Water Reactor (2014)
- **NUREG-2206**, Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments (2018)
- **NUREG/BR-0524**, Cooperative Severe Accident Research Program (CSARP)(2015)
- **NUREG/CR-7143**, Characterization of Thermal-Hydraulic and Ignition Phenomena in Prototypic, Full-Length Boiling Water Reactor Spent Fuel Pool Assemblies After a Postulated Complete Loss-of-Coolant Accident (2013)
- **NUREG/CR-7144**, Laminar Hydraulic Analysis of a Commercial Pressurized Water Reactor Fuel Assembly (2013)

References (www.nrc.gov)(2/2)

- **NUREG/CR-7216**, Spent Fuel Pool Project Phase II: Pre-Ignition and Ignition Testing of a 1x4 Commercial 17x17 Pressurized Water Reactor Spent Fuel Assemblies under Complete Loss of Coolant Accident Conditions (2016)
- **NUREG/CR-7245**, State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses (2019)
- **NUREG/CR-7282**, Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases (2021)
- **NUREG/CR-7283**, Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions (2021)
- **SECY-16-0100**, "National Academy of Sciences Study of the Lessons Learned from the Fukushima Nuclear Accident for Improving Safety and Security of U.S. Nuclear Power Plants"

Abbreviations

ATF	accident tolerant fuel
ATWS	anticipated transient without scram
DBA	design basis accident
DW	drywell
EAB	exclusion area boundary
EU	European Union
DRACS	direct reactor auxiliary cooling system
FHR	fluoride salt-cooled high-temperature reactor
HALEU	high-assay low-enriched uranium (fuel)
HBU	high burnup (fuel)
HPR	heat pipe reactor
HTGR	high temperature gas-cooled reactor
INL	Idaho National Laboratory
IRSN	Institut de radioprotection et de sûreté nucléaire (France)
LPZ	low-population zone
MSR	molten salt reactor
NEA	Nuclear Energy Agency
OCP	operating cycle phase
ORNL	Oak Ridge National Laboratory
PBMR	pebble bed modular reactor
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
RPV	reactor pressure vessel
SAWA	severe accident water addition
SAWM	severe accident water management
SFP	spend fuel pool
SFR	sodium-cooled fast reactor
SMR	small modular reactor
SNL	Sandia National Laboratory
SOARCA	State-of-the-Art Reactor Consequence Analysis

NuScale EPZ Sizing Methodology Topical Report, Rev. 2
Light Water SMR Design Certification Source Term Approach
Source Term Approach for Early non-LWR Movers

Accident Source Term in Recent and Near-term Applications

Michelle Hart
NRR/DANU/UTB2

Outline

- SMR and non-LWR accident source terms recent experience
- Emergency planning zone size justification consequence analyses
- Example: SMR design certification source term approach
- Source term approaches for non-LWR early movers

SMR and Non-LWR Accident Source Terms Recent Experience

- SMR topical report reviews and SMR DC application review
- Advanced reactor pre-application interactions, topical report reviews, and license applications
- Source term development contractor reports

Emergency Planning Zone Size Justification Consequence Analyses

- Concept based on NUREG-0396
  - Technical basis for plume exposure and ingestion pathway EPZ radius of ~10 and ~50 miles, respectively
  - Identification of area within which prompt protective actions may be necessary to provide dose savings in the event of a radiological release
- Calculate dose at distance for a spectrum of accidents
  - Analysis includes design basis accidents and severe accidents

Emergency Planning Zone Size Justification Consequence Analyses

- No separate/unique source terms developed especially for EPZ size analysis
  - Re-use source terms and accident release information developed for safety analysis report and PRA

Emergency Planning Zone Size Justification Consequence Analyses

- Methodology to support exemptions to 10-mile requirement
  - Clinch River ESP EPZ size methodology described in SSAR
- Methodology to support plume exposure pathway EPZ size determination on case-by-case basis for reactors <250 MWt
  - NuScale EPZ sizing methodology topical report (under review)
- EPZ size determination required in EP for SMRs and ONTs alternative framework, once issued
  - SECY-22-0001 issued for Commission review and approval
  - Guidance on analysis in appendices to RG 1.242

NuScale EPZ Sizing Methodology Topical Report

- TR-0915-17772, Revision 2, submitted in 2020, currently under review
  - Not part of DC review
  - Applicable to light-water SMRs such as NuScale, although not limited to the NuScale designs
  - Rev. 3 under development
- Analysis methodology to determine plume exposure pathway EPZ size

NuScale EPZ Sizing Methodology Topical Report

- "Source term" refers to fission product release to the environment as a function of time
- Uses source terms from DBAs (DC FSAR Ch. 15) and PRA severe accident scenarios scoped into analysis
  - No separate/unique source terms developed especially for EPZ size analysis
  - Uses CDF from PRA to categorize severe accidents and select accident sequences to evaluate against relevant dose criteria

Example: SMR Design Certification Source Term Approach

- SECY-19-0079, August 16, 2019
  - Describes staff review approach to evaluate accident source terms for both the TR and the NuScale SMR DC application
  - Provides basis for using source term without core damage for environmental qualification

Example: SMR Design Certification Source Term Approach – NuScale TR

- NuScale TR-0915-17565, “Accident Source Term Methodology,” Revision 4, February 2020
  - Methods to develop accident source terms are consistent with RG 1.183 guidance for PWRs except for:
    - Core damage source term for Core Damage Event
    - Iodine spike design basis source term (no fuel damage)

NuScale TR: Core Damage Event

- Derive source term from range of accident scenarios that result in significant damage to the core
  - Informed by NuScale SMR PRA
- NuScale-design-specific analyses using MELCOR to be performed by applicant referencing the TR
- Radionuclide transport phenomena
  - Iodine retention in containment based on pH
  - Aerosol natural deposition in containment

NuScale SMR DC Application: Core Damage Event

- Implemented the NuScale TR methodology to determine the core damage source term
- Core inventory calculated using SCALE code
- Scenario selection
  - Based on NuScale SMR PRA, internal events
  - 5 surrogate scenarios
    - Various failures of ECCS, with decay heat removal system available
    - Intact containment

NuScale SMR DC Application: Core Damage Event

- MELCOR used to estimate release timing and magnitude for each scenario
  - Release onset and duration from scenario with minimum time to core damage
  - Core release fractions taken as median of scenarios
- Time-dependent aerosol removal rates calculated using STARNAUA code
  - Design-specific input thermal hydraulic conditions calculated by MELCOR for surrogate scenario with minimum time to core damage

Source Term Approaches for Non-LWR Early Movers

- Kairos Power
  - MST methodology TR (under review)
    - Methodology for applicants to develop event-specific radiological source terms
      - DBAs for siting and safety analysis
      - AOOs and DBEs for LMP
  - Hermes CP application (under review)
    - Evaluates MHA, deterministic
    - Refers to MST TR

Source Term Approaches for Non-LWR Early Movers

- X-energy
  - Proposed to use developer-made source term code (XSTERM) which includes modeling of radionuclides from generation to release (and dose)
  - TR was submitted, but withdrawn to clarify and resubmit in future (not currently under review)

Source Term Approaches for Non-LWR Early Movers

- Oklo Aurora COL application (review ended)
  - Proposed maximum credible accident without release
- TerraPower
  - Development of source term methodology described in 1/13/2022 public meeting (ML22011A072)
  - Topical report planned for April 2023
- Terrestrial, Westinghouse, Others
  - Source terms to be determined
  - Public website information on non-LWR pre-application activities

Acronyms

AOO	anticipated operational occurrence
CDF	core damage frequency
COL	combined license
CP	construction permit
DBA	design basis accident
DBE	design basis event
DC	design certification
ECCS	emergency core cooling system
EP	emergency preparedness
EPZ	emergency planning zone
ESP	early site permit
FSAR	final safety analysis report
LMP	Licensing Modernization Project
MHA	maximum hypothetical accident
MST	mechanistic source term
MWt	megawatts thermal
Non-LWR	non-light water reactor
ONTs	other new technologies
PRA	probabilistic risk assessment
PWR	pressurized water reactor
RG	regulatory guide
SMR	small modular reactor
SSAR	site safety analysis report
TR	topical report

LUNCH

Accident Consequence-Related Regulation Activities

Michelle Hart
NRR/DANU/UTB2

Petition for Rulemaking

- PRM-50-121, Voluntary Adoption of Revised Design Basis Accident Dose Criteria
  - Received 11/23/2019, docketed 2/19/2020 (85 FR 31709)
  - Under evaluation – no disposition yet
- Requests voluntary rule to allow power reactor licensees to adopt alternative to the accident dose criteria specified in § 50.67, ''Accident source term.''
- Proposes a uniform value of 100 milli-Sieverts (10 rem) for offsite locations and for the control room

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Final rule in development
  - New section 10 CFR 50.160, and related/conforming changes
  - ACRS meetings in September and November 2021
- RG 1.242 (to be issued with final rule)
  - Appendices
    - Generalized analysis methodology
    - Information on source terms

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Appendix A, "General Methodology for Establishing Plume Exposure Pathway Emergency Planning Zone Size"
  - Provides general guidance on the consequence analysis to support plume exposure pathway EPZ size determination
  - Discusses event selection and consideration of accident likelihood

Emergency Preparedness for SMRs and Other New Technologies Rulemaking

- Appendix B, “Development of Information on Source Terms”
  - Provides guidance to develop source terms for plume exposure pathway EPZ size evaluations

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Alternative Physical Security for Advanced Reactors Rulemaking

- Draft rule and guidance in development
- Voluntary alternative physical security requirements commensurate with potential safety and security consequences
- Analyses (guidance under development)
  - Develop relevant scenarios
  - Site-specific potential offsite radiological consequences

Acronyms

CFR	Code of Federal Regulations
EPZ	emergency planning zone
FR	Federal Register
PRM	petition for rulemaking
RG	Regulatory Guide
SMR	small modular reactor

Guidance and Information for Developing Source Terms for Non-LWRs

Michelle Hart, NRR/DANU/UTB2
Bill Reckley, NRR/DANU/UARP
Tim Drzewiecki, NRR/DANU/UTB1

Outline

- Accident consequence analysis for advanced reactors
- Mechanistic source term
- Recent reports on Non-LWR source term development
- Non-LWR PRA standard and source term
- Licensing Modernization Project and source term
- Overview of method in NUREG-2246, “Fuel Qualification for Advanced Reactors”
- Non-LWR accident source term information website

Accident Consequence Analysis for Advanced Reactors

- Regulatory nexus
  - Siting and safety analysis regulatory requirement
  - Newer uses for advanced reactors
    - LMP
    - Plume exposure pathway EPZ size determination
    - Alternative security requirements – ongoing rulemaking
    - Part 53 – ongoing rulemaking

Accident Consequence Analysis for Advanced Reactors

- Accident source term development considerations
  - Event selection, scenarios
  - Balance of prevention vs. mitigation
  - Relationship to functional containment
    - A barrier, or set of barriers taken together, that effectively limit the physical transport of radioactive material to the environment (SECY-18-0096)
  - Relationship to PRA
  - Uncertainty

Accident Consequence Analysis for Advanced Reactors

- Mechanistic or deterministic evaluation
  - LMP assumes MST and use of PRA
  - Some non-LWRs may choose to provide a postulated MHA, similar to non-power reactor licensees
- No current specific RG on MST or non-LWR source terms, however
  - RG 1.183, regulatory position C.2, "Attributes of an Acceptable AST," may be useful
  - SECY-93-092 included staff recommendations on non-LWR source terms

Mechanistic Source Term

- SECY-93-092 definition of MST

  A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

SECY-93-092: Provisions for Staff Assurance

- *The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.*
- *The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.*
- *The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties*

National Lab Non-LWR Source Term Reports

- Technology inclusive, what to do to develop accident source terms, not specific on how to do it
- No specific methods or phenomenological models
- Do not provide technology-related source terms or releases

Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities

INL/EXT-20-58717, Revision 0, June 2020, ML20192A250

- Summarizes a risk-informed, performance-based, and technology-inclusive approach to determine source terms
- Graded process
  - Conservative non-mechanistic approach
  - MST calculation methods
    - Design-specific scenarios for a range of licensing basis events
    - Best-estimate models with uncertainty quantification

MST Formulation

$$I(RN_j) * F(S_i, RN_j, t) * MR(S_i, RN_j, t) * PSR(S_i, RN_j, t) * LPF(S_i, RN_j, t) \quad = \quad ST(S_i, RN_j, t)$$

Figure 1-2 INL/EXT-20-58717, Revision 1. From Illustration of radionuclides retention and removal process for one non-LWR concept (reproduced from SAND2020-0402)

Technology-Inclusive Source Term Methodology Determination

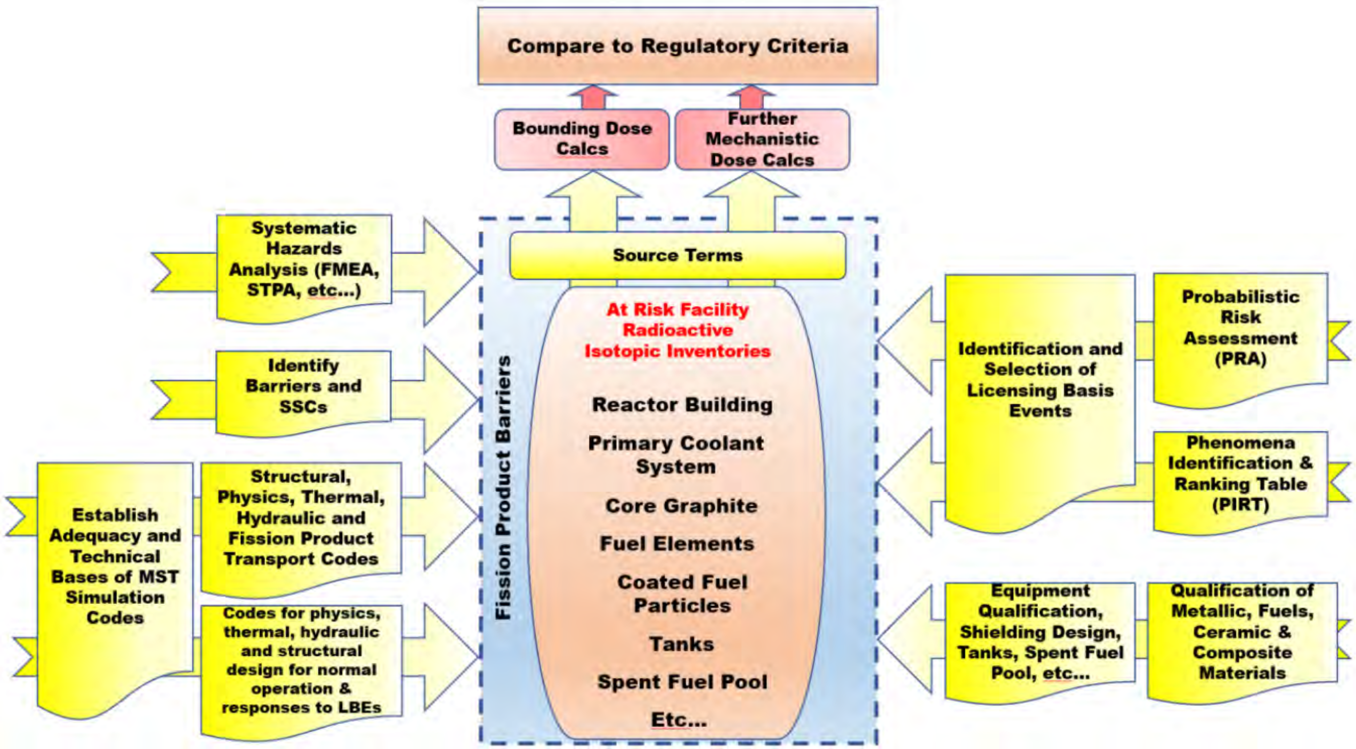


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

INL Report Methodology Steps

1: Identify Regulatory Requirements

2: Identify Reference Facility Design

3: Define Initial Radionuclide Inventories

4. Perform Bounding Calculations

5. Conduct SHA and Perform Simplified Calculations

6. Consider Risk-informed System Design Changes

7. Select Initial List of LBEs and Conduct PIRT

8. Establish Adequacy of MST Simulation Tools

9. Develop and Update PRA Model

10. Identify or Revise the List of LBEs

11. Select LBEs to Include Design Basis External Hazard Level for Source Term Analysis

12. Perform Source Term Modeling and Simulation for LBEs

13. Review LBEs List for Adequacy of Regulatory Acceptance

14. Document Completion of Source Term Development

Simplified Approach for Scoping Assessment of Non-LWR Source Terms

SAND2020-0402, January 2020, ML20052D133

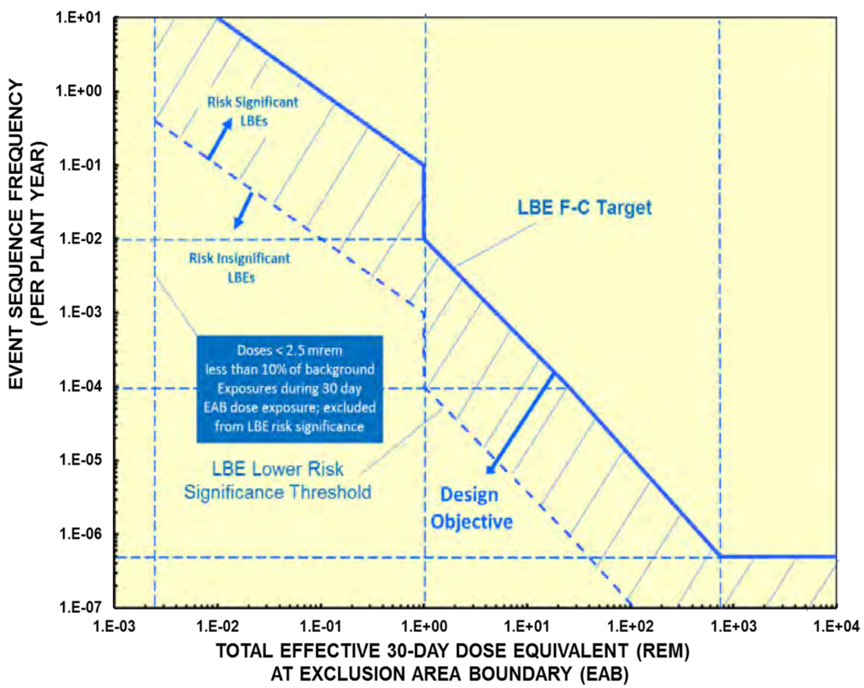
- Primarily qualitative means to identify the dominant considerations that affect a release mitigation strategy
- Classifies release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance under sample accident scenarios
- Did NOT develop quantitative estimates of radiological release magnitudes and compositions to the environment
- Looked at high temperature gas reactors, sodium fast reactors, and liquid fueled molten salt reactors

Non-LWR PRA Standard ASME/ANS RA-S-1.4-2021

- Full scope PRA (includes consequence analysis)
- Mechanistic Source Term Analysis (MS) element provides useful information on what to do to develop mechanistic source terms

Licensing Modernization

- Risk-informed approach to selection and analysis of licensing basis events
- Combined with assessment of cumulative risks
- Key roles for PRA and MST



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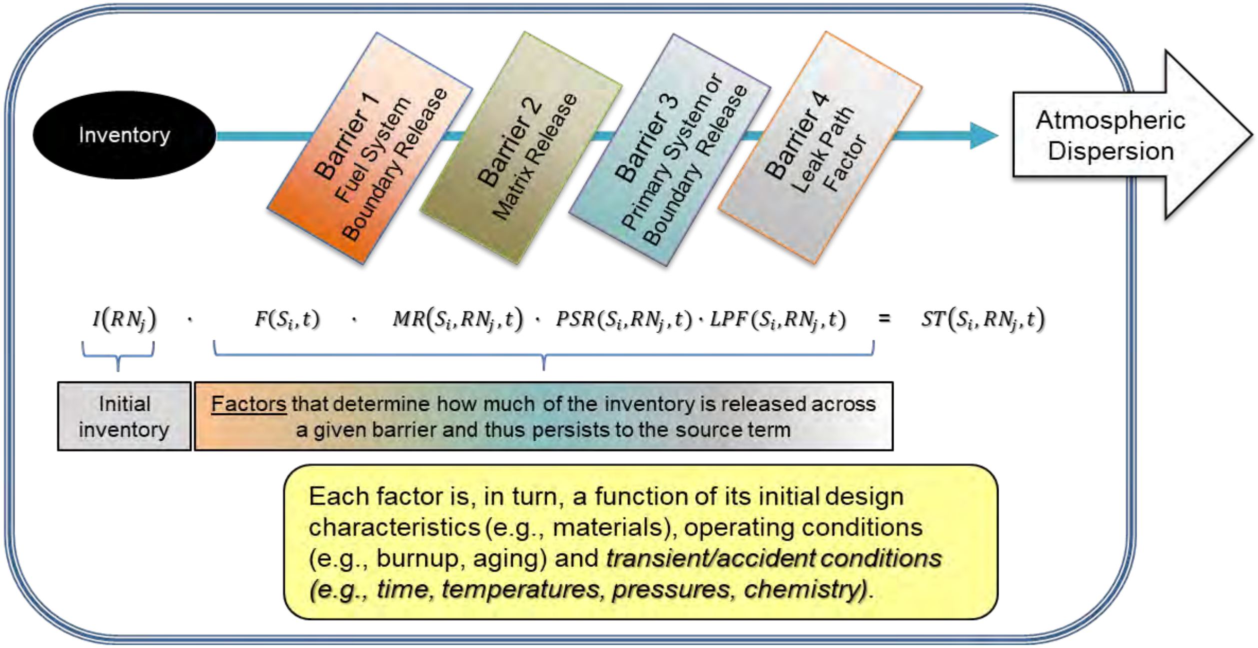
Licensing Modernization

Recent NRC activities related to advanced reactors (e.g., functional containment performance criteria, possible changes to emergency planning & security, and DG-1353) recognize the limitations of existing LWR-related guidance, which requires a return to first principles such as fundamental safety functions supporting the retention of radionuclides



See: SECY-18-0096, "Functional Containment Performance Criteria for Non-Light-Water-Reactors," and INL/EXT-20-58717, "Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities"

Licensing Modernization

- Flexibility provided on how to develop safety case
- NRC Advanced Reactor Policy Statement encourages use of passive and inherent features

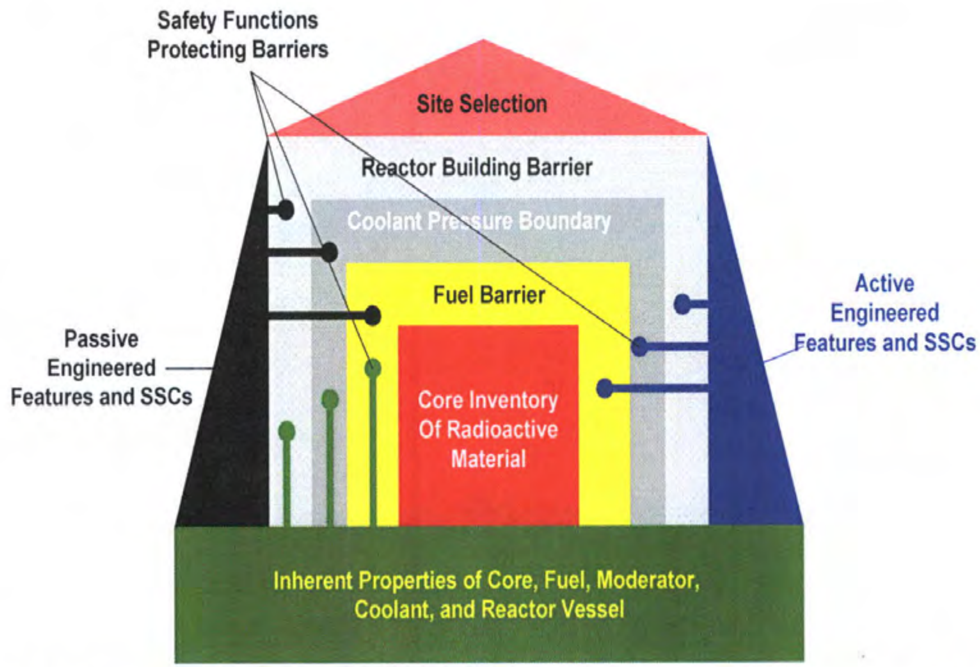
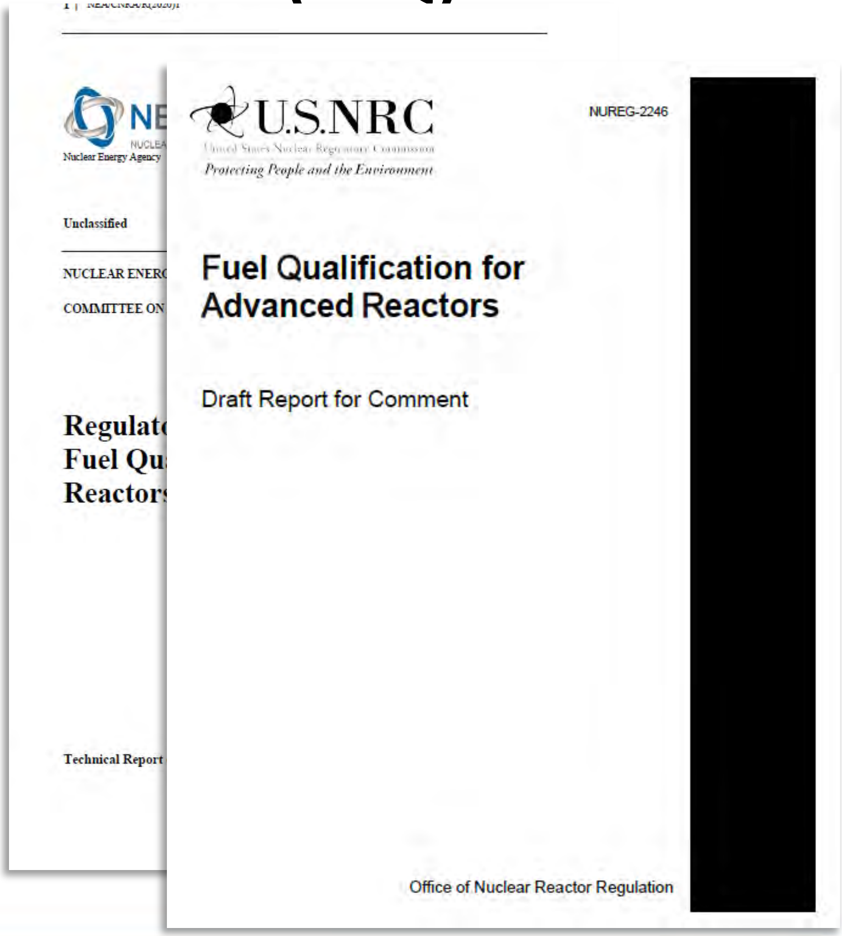


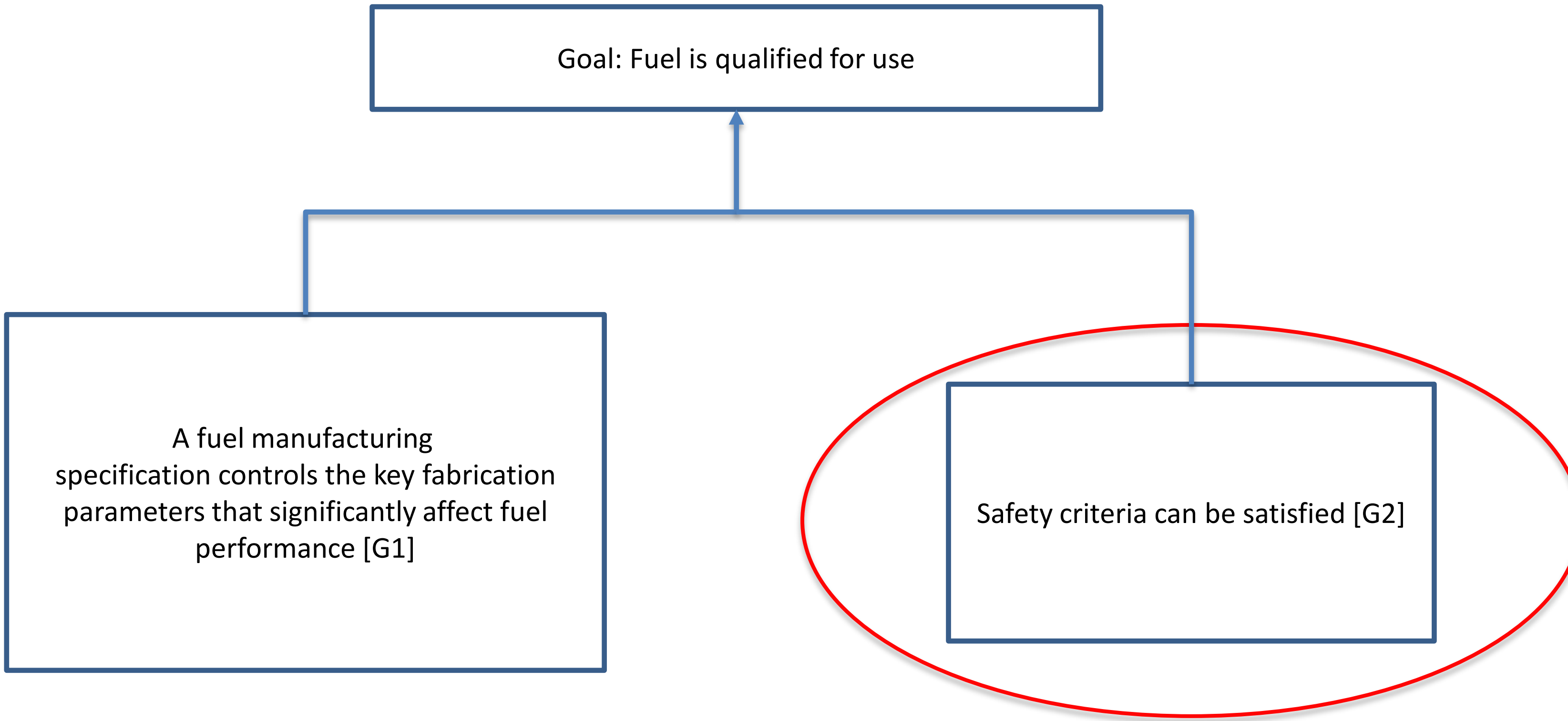
Figure 3-6. Elements of safety design approach incorporated into ***Plant Capability Defense-in-Depth***.

Assessment Frameworks
Fuel Qualification (FQ)

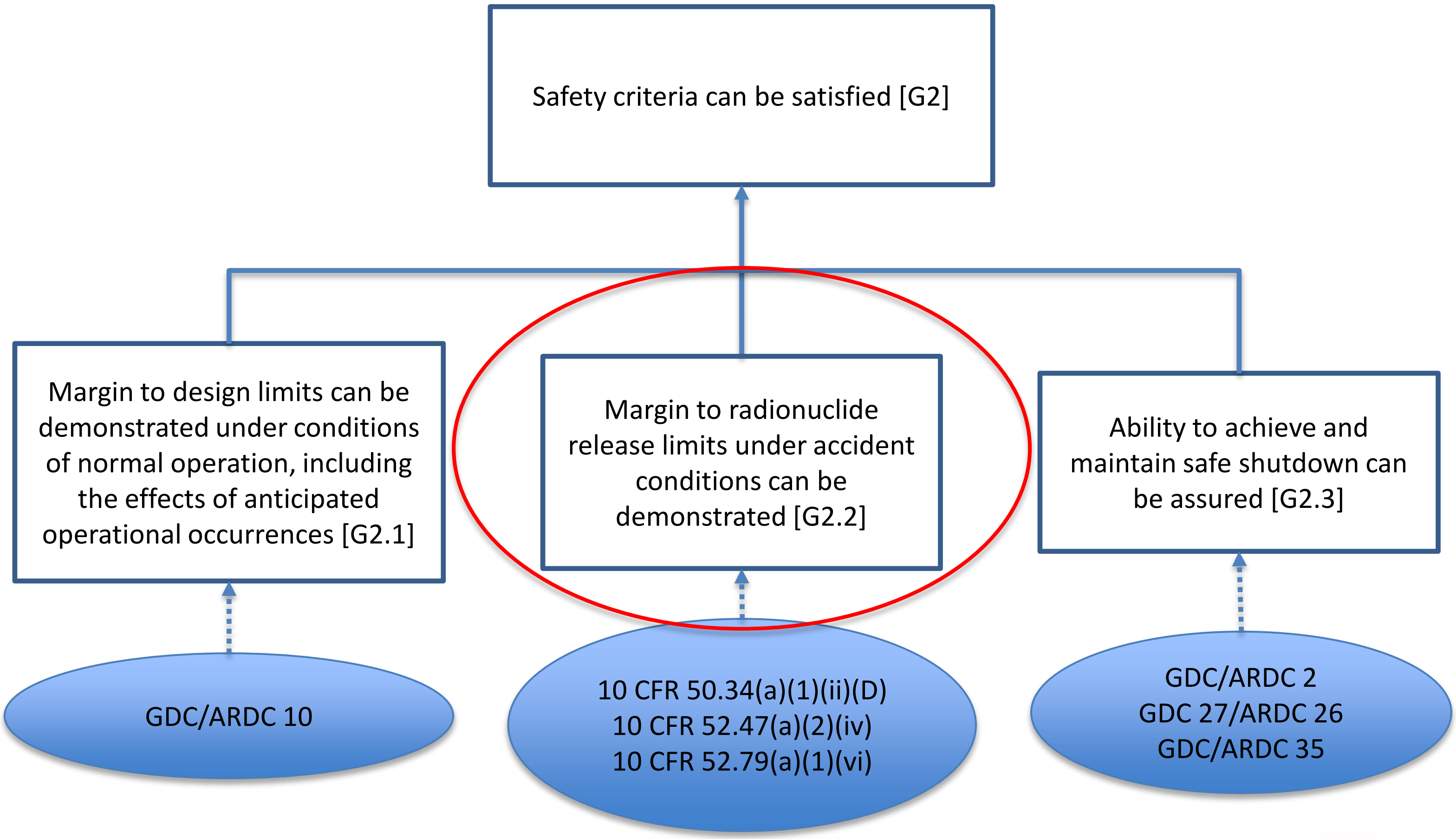
- Top-down approach to identify criteria (goals) to support a finding that "fuel is qualified"



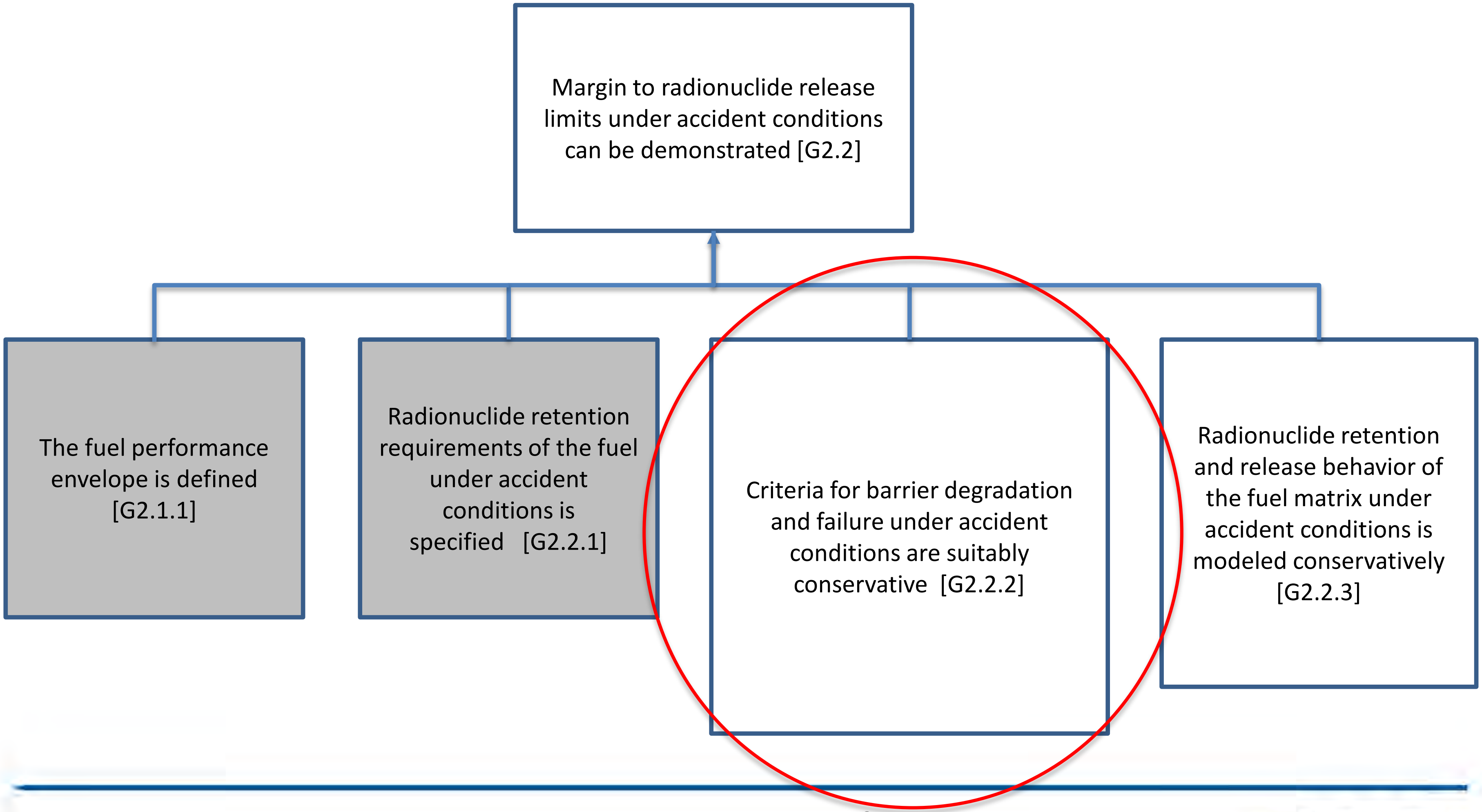
FQ Assessment Framework



G2: Safety Criteria



G2.2: Radionuclide Release Limits



G2.2.2 Criteria for Barrier Degradation

Criteria for barrier degradation and failure under accident conditions are suitably conservative [G2.2.2]

Criteria are shown to provide conservative prediction of barrier degradation and failure [G2.2.2(a)]

Experimental data is appropriate [G2.2.2(b)]

Note: Testing at environmental conditions consistent with accident conditions is expected (e.g., elevated temperatures)

Complete FQ Assessment Framework

GOAL				Fuel is qualified for use
G1				Fuel is manufactured in accordance with a specification
	G1.1			Key dimensions and tolerances of fuel components are specified
	G1.2			Key constituents are specified with allowance for impurities
	G1.3			End state attributes for materials within fuel components are specified or otherwise justified
G2				Margin to safety limits can be demonstrated
	G2.1			Margin to design limits can be demonstrated under conditions of normal operation and AOOs
		G2.1.1		Fuel performance envelope is defined
		G2.1.2		Evaluation model is available (see EM Assessment Framework)
	G2.2			Margin to radionuclide release limits under accident conditions can be demonstrated
		G2.1.1		Fuel performance envelope is defined
		G2.2.1		Radionuclide retention requirements are specified
		G2.2.2		Criteria for barrier degradation and failure are suitably conservative
			(a)	Criteria are conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
		G2.2.3		Radionuclide retention and release from fuel matrix are modeled conservatively
			(a)	Model is conservative
			(b)	Experimental data are appropriate (see ED Assessment Framework)
	G2.3			Ability to achieve and maintain safe shutdown is assured
		G2.3.1		Coolable geometry is ensured
			(a)	Criteria to ensure coolable geometry are specified
			(b)	Evaluation models are available (see EM Assessment Framework)
		G2.3.2		Negative reactivity insertion can be demonstrated
			(a)	Criteria are provided to ensure that negative reactivity insertion path is not obstructed
			(b)	Evaluation model is available (see EM Assessment Framework)

GOAL			Evaluation model is acceptable for use
EM G1			Evaluation model contains the appropriate modeling capabilities
	EM G1.1		Evaluation model is capable of modeling the geometry of the fuel system
	EM G1.2		Evaluation model is capable of modeling the material properties of the fuel system
	EM G1.3		Evaluation model is capable of modeling the physics relevant to fuel performance
EM G2			Evaluation model has been adequately assessed against experimental data
	EM G2.1		Data used for assessment are appropriate (see ED Assessment Framework)
	EM G2.2		Evaluation model is demonstrably able to predict fuel failure and degradation mechanisms over the test envelope
		EM G2.2.1	Evaluation model error is quantified through assessment against experimental data
		EM G2.2.2	Evaluation model error is determined throughout the fuel performance envelope
		EM G2.2.3	Sparse data regions are justified
		EM G2.2.4	Evaluation model is restricted to use within its test envelope

GOAL		Experimental data used for assessment are appropriate
ED G1		Assessment data are independent of data used to develop/train the evaluation model
ED G2		Data has been collected over a test envelope that covers the fuel performance envelope
ED G3		Experimental data have been accurately measured
	ED G3.1	The test facility has an appropriate quality assurance program
	ED G3.2	Experimental data are collected using established measurement techniques
	ED G3.3	Experimental data account for sources of experimental uncertainty
ED G4		Test specimens are representative of the fuel design
	ED G4.1	Test specimens are fabricated consistent with the fuel manufacturing specification
	ED G4.2	Distortions are justified and accounted for in the experimental data

* For illustrative purposes only. Please see Appendix A to NUREG-2246 for a legible list.

Non-LWR Accident Source Term Webpage Information

https://www.nrc.gov/reactors/new-reactors/advanced/related-documents/nuclear-power-reactor-source-term.html

- One-stop shop for existing information, on public website under advanced reactors
  - Discussion of accident source terms
  - Linked list of documents relevant to development of non-LWR accident source terms for licensing
- Staff will keep up to date

Acronyms

AST	alternative source term
EPZ	emergency planning zone
INL	Idaho National Laboratory
LBE	licensing basis event
LMP	Licensing Modernization Project
LWR	light water reactor
MHA	maximum hypothetical accident
MST	mechanistic source term
Non-LWR	non-light water reactor
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
RG	regulatory guide
SHA	system hazard analysis

Guidance for developing advanced reactor source term (long-term)

Bill Reckley
Michelle Hart
John Segala
NRR/DANU

General Approach

- Maintain traditional LWR approach (RG 1.183) as an acceptable option
- Technology-inclusive methodology available as an option
- Actual implementation is technology/design specific
- NRC not planning to provide analytical inputs to applicants (beyond making available NRC developed models)

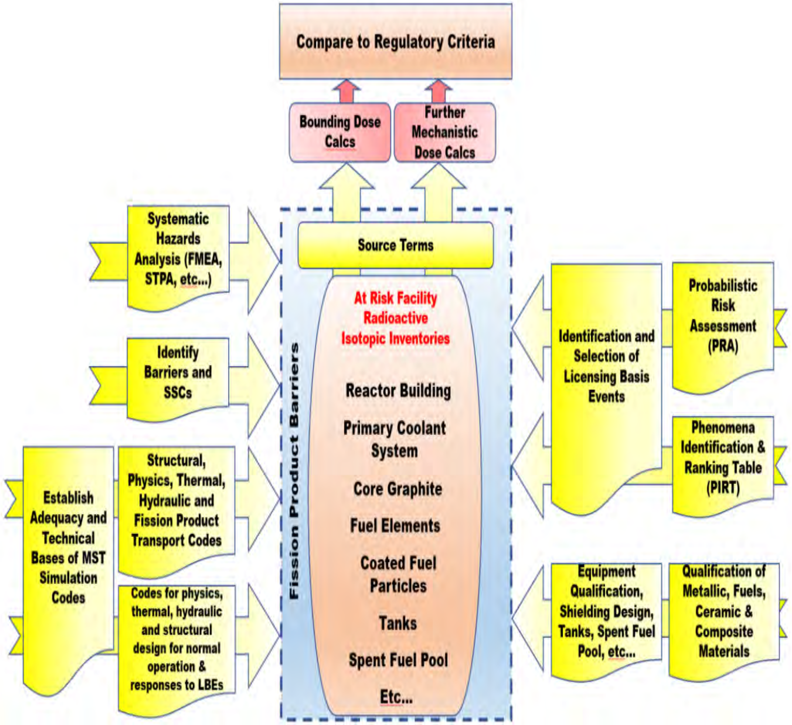
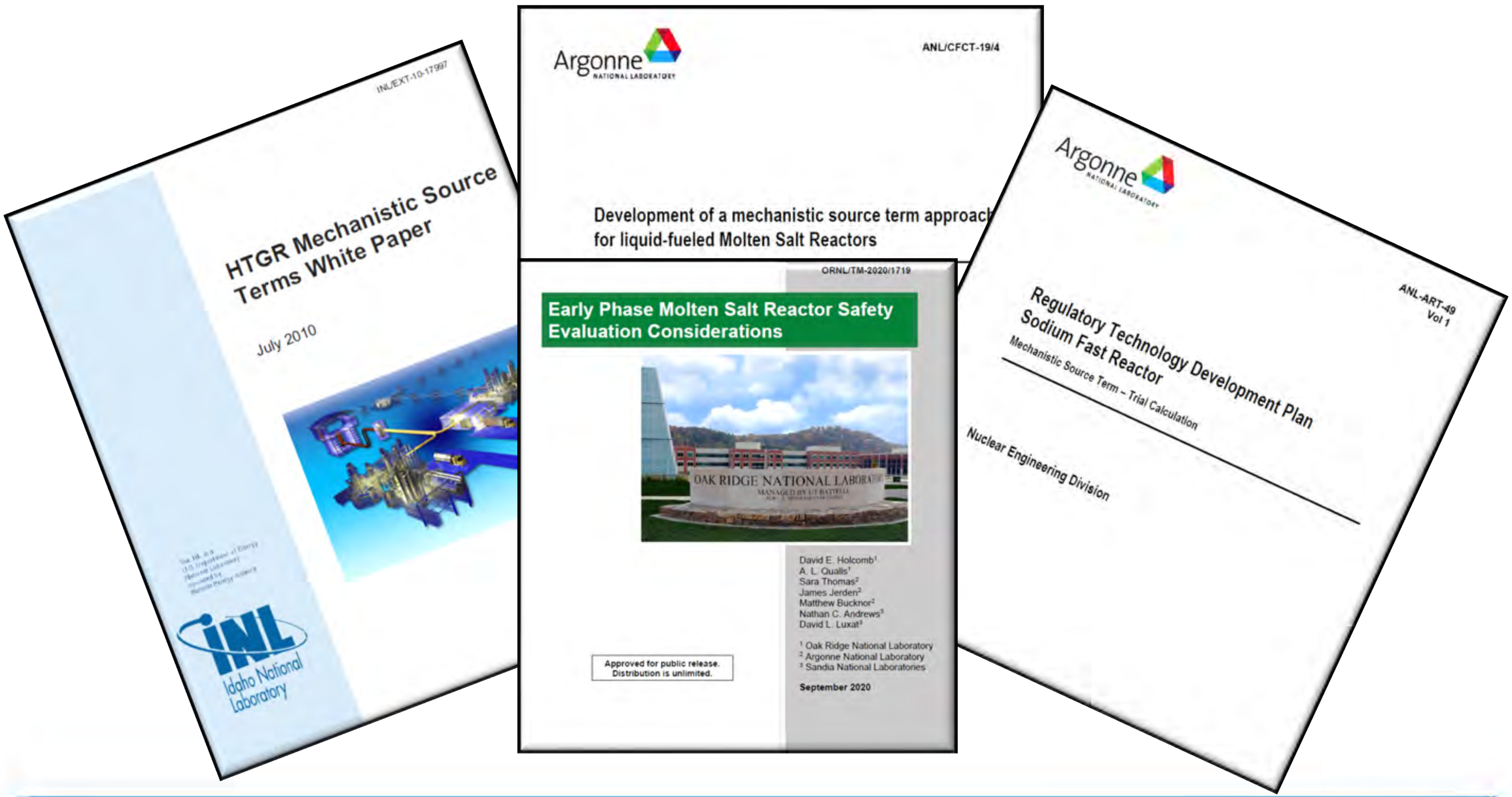
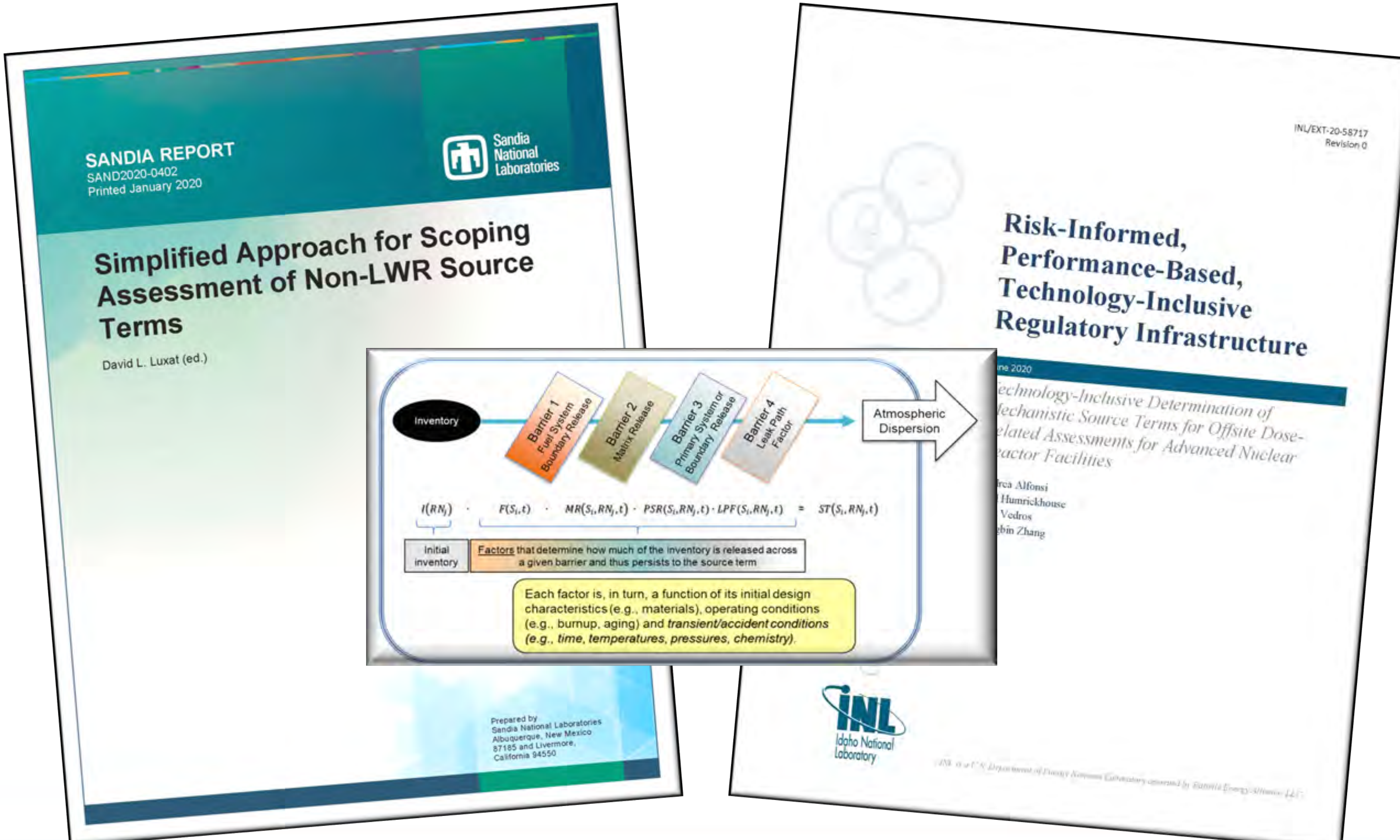


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).

DOE/National Laboratories



NRC Activities



Next Generation Nuclear Plant (NGNP)

Mechanistic Source Terms White Paper *INL/EXT-10-17997*

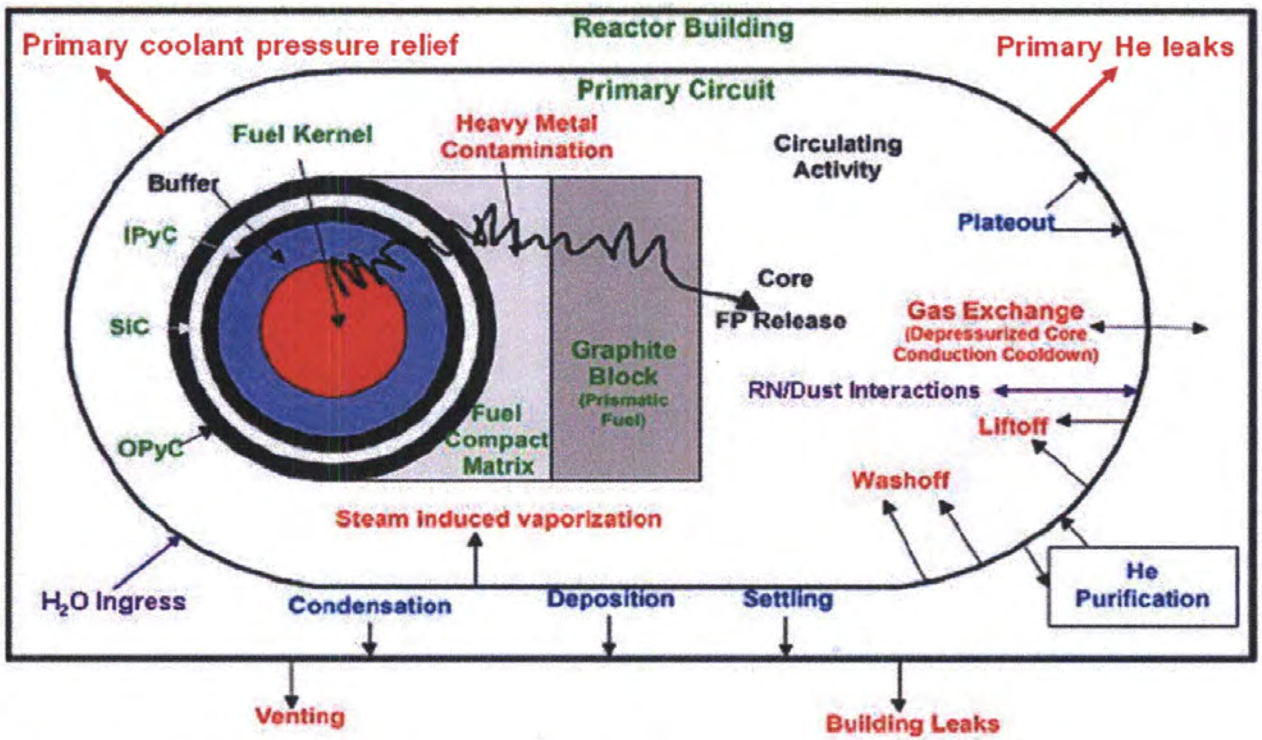
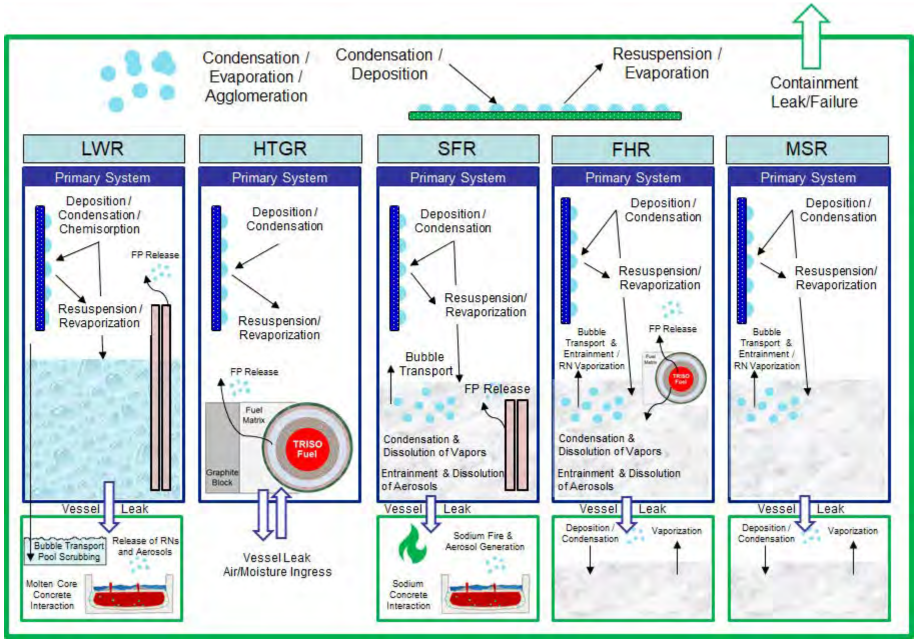


Figure 2-4. HTGR radionuclide retention system.

Model Development



Primer & User Guide
Reference Manual
Assessment Problems

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Applications & Pre-App Interactions

Moving Forward

- Following the scientific work being done by national laboratories and developers
- Engaging with developers
- Continuing to develop NRC models and identify related uncertainties
- Consider additional guidance based on experience with ongoing interactions
- Consider feedback on the new webpage

Opportunity for Public Comment

Member Discussion

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