

# **Official Transcript of Proceedings**

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Future Plant Designs Subcommittee

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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1 UNITED STATES OF AMERICA

2 NUCLEAR REGULATORY COMMISSION

3 + + + + +

4 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

5 (ACRS)

6 + + + + +

7 FUTURE PLANT DESIGNS SUBCOMMITTEE

8 + + + + +

9 THURSDAY

10 FEBRUARY 17, 2022

11 + + + + +

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

14  
15 COMMITTEE MEMBERS:

16 DAVID A. PETTI, Chairman

17 RONALD G. BALLINGER, Member

18 VICKI M. BIER, Member

19 CHARLES H. BROWN, JR. Member

20 VESNA B. DIMITRIJEVIC, Member

21 GREGORY H. HALNON, Member

22 WALTER L. KIRCHNER, Member

23 JOSE MARCH-LEUBA, Chairman

24 JOY L. REMPE, Member

25 MATTHEW W. SUNSERI, Member

1 ACRS CONSULTANTS:

2 STEPHEN SCHULTZ

3 DENNIS BLEY

4

5 DESIGNATED FEDERAL OFFICIAL:

6 MIKE SNODDERLY

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## 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.

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1 CHAIRMAN PETTI: Ahh, great.

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

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

6 Mike Snodderly is the designated federal  
7 official for this meeting.

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

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

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

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

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

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

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

9           The agenda for this meeting was posted  
10 there as well.

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

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

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

24           There will be an opportunity for public  
25 comment, and we set aside 10 minutes in the agenda for



1 comments from members of the public attending, or  
2 listening to our meetings.

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

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

10 Additionally, participants should mute  
11 themselves when not speaking.

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

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

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

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

25 MR. BURKHART: Okay, thanks. Just for

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

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

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

13 Thanks.

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

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

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

24 And, then somewhere in between that  
25 presentation, we will have a break that Dave Petti

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1 will tell us.

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

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

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

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

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

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

23 John Segala?

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

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

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

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

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

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

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

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

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

10          For example, the TID-14844 are source  
11          term.

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

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

22          Although we acknowledge that developing  
23          design and scenario specific source term will require  
24          substantial work, the staff believes that there is  
25          sufficient guidance available for applicants to

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1 develop design and scenario specific source terms, and  
2 for the NRC staff to complete its technical review.

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

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

16 We plan to showcase the webpage for you  
17 today.

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

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

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

4 If you could go to the next slide?  
5 Thanks.

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

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

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

21 That completes my opening remarks.

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

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1 the NRC regulatory process.

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

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

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

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

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

21 A critical component of the dose analysis  
22 is the source term. In the Regulations for Part 50,  
23 the NRC defines the source term as the magnitude and  
24 mix of the radionuclides released from the fuel,  
25 expressed as fractions of fission product inventory in

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1 the fuel, as well as the physical and chemical form  
2 and timing of their release.

3 Please go to slide 6 now.

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

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

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

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

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

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

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1 regulatory source terms.

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

6 Go to slide 7, please.

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

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

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

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

24 To determine the acceptability of these  
25 designs, at the time the Atomic Energy Commission

1 proposed using population densities to site reactors.

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

7 Please go to slide 8.

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

12 Nearly all the current operating reactors  
13 were licensed originally to 10 CFR Part 100, and TID-  
14 14844, which, as John said, provide guidance on a  
15 containment source term to be used for LOCAs involving  
16 significant fuel melt.

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

19 The source term involves the release of  
20 100 percent of the Noble gasses, 50 percent of the  
21 iodine, and 1 percent of the other radionuclides as  
22 particles.

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

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

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

8           You can go to slide 9 now, please.

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

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

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

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

25          The cooperative research resulted in the

1 development of the source term copackage to examine  
2 core melt progression, and fission product release and  
3 transport in LWRs.

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

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

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

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

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

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

22 It allowed plants that were licensed under  
23 10 CFR Part 100, to convert to the AST, 10 AST. An  
24 acceptable AST based upon NUREG-1465 was provided in  
25 a document known as Reg Guide 1.183.

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

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

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

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

15          And, go to slide number 11 now, please.

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

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

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

1 aerosols.

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

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

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

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

16 The NRC has decided not to finalize DG-  
17 1199 as Reg Guide 1.183 Revision 1, and is now issuing  
18 a replacement known as DG-1389.

19 You can go to slide 13 now, please.

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

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

1 in addressing those comments.

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

5 In 2017, the staff received the final  
6 responses from SNL, associated with that independent  
7 review. But then when we resumed our efforts late in  
8 2020, we had gained a significant number of insights  
9 since the issuance of that 1199.

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

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

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

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

24 DR. BLEY: Thanks, Mark.

25 MR. BLUMBERG: You're welcome.



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

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

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

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

15          Can you go to slide 14, now, please?

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

19          And, all of the changes you had on slide  
20          13, as well as what I'm seeing here, are LWR based,  
21          correct?

22          MR. BLUMBERG: That's correct.

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

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

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

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

13 CHAIRMAN PETTI: No, I've seen a draft --  
14 (Simultaneous speaking.)

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

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

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

25 But we're going to be, I'm assuming we're

1 all going to, that's all going to get wrapped up in  
2 the next meeting on 1.183.

3 MR. BLUMBERG: That will be a future --  
4 (Simultaneous speaking.)

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

6 MR. BLUMBERG: Go ahead.

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

11 But it's still all LWR based, right?

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

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

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

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

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

5 MEMBER REMPE: Okay, thank you.

6 MR. BLUMBERG: You're welcome.

7 CHAIRMAN PETTI: Steve, you have a comment?

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

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

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

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

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

25 And, also with respect to some of the

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

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

6 Thank you.

7 MR. BLUMBERG: Thank you.

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

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

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

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

25 Please go to slide 15 now.

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

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

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

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

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

21          MEMBER KIRCHNER: Mark, this is Walt  
22    Kirchner.

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

1 future?

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

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

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

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

17 So, the files is there.

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

21 It hasn't changed since then?

22 MR. BLUMBERG: Not substantially, no?

23 CHAIRMAN PETTI: Could you check on that?

24 (Audio interference.)

25 CHAIRMAN PETTI: Okay, thanks.

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

3 (Simultaneous speaking.)

4 CHAIRMAN PETTI: Yes. I think so.

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

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

12 But we haven't actually done that change  
13 yet.

14 CHAIRMAN PETTI: Okay.

15 MEMBER REMPE: Okay. And, then --

16 (Simultaneous speaking.)

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

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

24 MR. BLUMBERG: Good.

25 CHAIRMAN PETTI: So.



1 MEMBER REMPE: And, then --

2 (Simultaneous speaking.)

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

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

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

16 MS. HART: I am.

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

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

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

25 If you wanted to use the guidance that's

1 out there for non-power reactors, which is more  
2 focused on a maximum hypothetical accident as a, as a  
3 licensing aspect, you could also use that information.  
4 That is NUREG-1537, I think.

5 But there is nothing specific in Reg Guide  
6 1.183 that would give you a different licensing source  
7 term for that kind of assessment, if that's your  
8 question.

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

11 (Simultaneous speaking.)

12 CHAIRMAN PETTI: Yes, and --

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

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

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

23 But we are intending on keeping that  
24 website as up to date as possible, as we figure out  
25 new things that need to be on there. Or even old

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1 things that we, we don't recognize. It's a list of  
2 links. So, we'll talk about that some more later.

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

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

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

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

12 (Simultaneous speaking.)

13 CHAIRMAN PETTI: So --

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

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

20 CHAIRMAN PETTI: So Michelle, in terms of  
21 1.183, this guidance, is that the Appendix A as I  
22 remember?

23 MS. HART: So, we're --

24 (Simultaneous speaking.)

25 CHAIRMAN PETTI: That talks about the

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1 attributes?

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

6 CHAIRMAN PETTI: Ahh.

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

11 CHAIRMAN PETTI: Okay.

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

16 But I see a hand raised. Elijah?

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

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

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

25 And, so you can look at that document,

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

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

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

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

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

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

21 MEMBER REMPE: Okay, thank you.

22 MR. BLUMBERG: And, if you refer to slide  
23 9, that's the memo that Elijah was speaking to, the  
24 footnote there at the bottom. That provides the  
25 reference.

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

3           (Pause.)

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

7           Thank you.

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

11          Thank you.

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

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

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

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

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

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

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

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

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

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

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

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

24          MR. SEGALA: Yes, thank you.

25          Good comments there. I think that was one

1 of, that was in our thinking as we were putting  
2 together this, this new webpage on source term, was  
3 trying to provide, you know, members of the public  
4 sort of a here's where all the information is in one  
5 place.

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

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

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

15 But thanks for the comment.

16 DR. BLEY: Yes, thanks, John.

17 CHAIRMAN PETTI: Yes, to me --

18 (Simultaneous speaking.)

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

23 CHAIRMAN PETTI: Yes, I mean to me,  
24 something like a lead me first, right? The first  
25 document you should look at, you know, like the years



1       ago when you used to get software, that was the thing  
2       you had to do. So that they don't, the staff doesn't  
3       get, people don't get overwhelmed, and kind of guides  
4       them.

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

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

9               (Pause.)

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

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

17              Next slide, please.

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

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

24              These tools are under active development  
25       assessment. They are considered state of the practice

1 tools that is used worldwide, by many regulatory  
2 bodies, and research organizations.

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

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

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

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

23 Next slide, please.

24 So, first I will give an overview of the  
25 code MELCOR. This is our severe accident code. I

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1 will talk briefly about international collaboration,  
2 and this is important in how we understand severe  
3 accidents, and how we improve our code MELCOR.

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

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

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

16 Next slide, please.

17 So, well, next slide.

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

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

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

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

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

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

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

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

24          So, if you are interested in the in  
25          containment source term, then we can synthesize the

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1 results. This is what we see on the right hand side  
2 of the slides.

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

7 Next slide, please.

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

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

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

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

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

25 The staff is familiar with the code

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

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

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

10 Next slide, please.

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

13 Next slide, please.

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

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

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

24 We had the first release of MELCOR, this  
25 is MELCOR 1.80. This was a domestic release in 1986,

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1 and then followed by an international release in 1989.

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

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

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

14 And, spent fuel pool models. We started  
15 looking at spent fuel pools more closely after the  
16 9/11.

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

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

24 Okay, next slide, please.

25 All right, so MELCOR is an integral system

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1 level code. It models all phases of a severe accident  
2 from accident initiation to core heat up, and  
3 degradation leading to vessel and containment failure;  
4 tracks fission products through the RCS, and  
5 containment to the environment.

6 So, it does everything.

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

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

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

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

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

25 So, we are able to capture some of this

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1 phenomena through simpler models.

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

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

9 Am I live? Can everybody hear me?

10 CHAIRMAN PETTI: Yes, we can.

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

13 Okay. Next slide, please.

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

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

23 I would break it down to three main  
24 levels. So, going from the top, when it comes to the  
25 containment, there are phenomenon processes that are

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1 common. This has to do with high aerosols and vapors  
2 evolve and are removed. For example, aerosol-shaped  
3 factors may be different under dry and wet conditions  
4 in the containment.

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

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

18 Next slide, please.

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

25 Validation is targeted to a relevant

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

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

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

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

22 Next slide, please.

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

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

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

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

21                 CHAIRMAN PETTI: Hossein?

22                 MR. ESMAILI: Yes, sir.

23                 CHAIRMAN PETTI: Just a question. I  
24          assume -- I think it's true -- that, as the modeling  
25          sophistication has increased, those source terms have,

1 I guess, reduced. So, the estimates are more  
2 accurate, the uncertainty less. Is that a fair  
3 assessment generally?

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

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

17 CHAIRMAN PETTI: Yes.

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

20 CHAIRMAN PETTI: Yes. I'm just thinking  
21 about now you go to these advanced systems, where,  
22 hopefully, the physics is simpler; that, you know, the  
23 sledgehammer when you may only need to put the tack  
24 in. It may be more than is necessarily needed. But  
25 MELCOR is scalable in the sense that you could do

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1 something simple, if that's all you felt you needed,  
2 right? I mean, it's got that flexibility?

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

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

18 CHAIRMAN PETTI: Thanks.

19 MR. ESMAILI: Yes, thank you.

20 So, next slide, please.

21 So, for the past few years, we have  
22 conducted research to modernize MELCOR code with the  
23 key goal of enhancing the efficiency of the  
24 development and maintenance of this large, complex  
25 code base. Oh, yes, the code is huge. You know, in

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1 the earlier presentation, I always refer to it like it  
2 has almost 2 million lines of code. It's probably  
3 increasing by the day.

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

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

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

25 So, the idea in modernization, we still

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

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

12 All right. Next slide, please.

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

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

22 So, just like light water reactors, the  
23 model that we have developed requires some input data,  
24 and the data can come from all sources. It can come  
25 from codes, experiments, et cetera. So, I have listed

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1 some of the data needs in terms of what we need to put  
2 in the code. But I want to make sure that, as far as  
3 the code infrastructure is concerned, it can accept --  
4 you know, as data becomes available, we can put it  
5 into the code. We do not need to go and change  
6 basically the models. We can just change the input  
7 parameters.

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

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

19 MEMBER REMPE: Hossein, this is Joy.

20 MR. ESMAILI: Yes, ma'am.

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

23 (Laughter.)

24 I know when we talked about this a while  
25 back with the code readiness reviews documents, I

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1 thought we brought up the point that was you do the  
2 pilot plant applications. You'll also gain some  
3 insights about which data are very important and have  
4 more risk impact. And you can wait and answer this  
5 later in the pilot plant or reference plant  
6 discussions.

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

14 Do you understand where I'm going?

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

21 MEMBER REMPE: That's fine, but it's not  
22 just that you've identified it. I'm also curious if  
23 you're communicating that back. Is there a document  
24 on the website that says that -- you know, how do you  
25 inform the folks coming in? In this case, you're

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1 mentioning the gas reactors. Do the folks that are  
2 engaging in the pre-application discussions have a  
3 good feel for where the staff is with the progress  
4 they've made on the reference plant evaluations?

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

7 MEMBER REMPE: Uh-hum, right.

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

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

18 And so, you know --

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

21 MR. ESMAILI: Okay.

22 MEMBER REMPE: -- but I'm just thinking  
23 about readiness and guidance, and some of the things  
24 like Dave and Dennis brought up earlier, things that  
25 the staff has learned from these evaluations that not

1       only make them ready for incoming applications, but  
2       guidance, so that they focus.

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

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

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

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1 diffusivity is, right? So, this is a large effort to  
2 gain that.

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

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

16 Does that --

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

20 Thanks. Go ahead.

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

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

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

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

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

25 It is -- sorry?

1 MEMBER BIER: No, go ahead. Sorry.

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

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

18 MEMBER BIER: Great. Thanks.

19 MR. ESMAILI: Okay. Thank you.

20 Next slide, please, "International  
21 Collaboration."

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

24 But, next slide, please.

25 CHAIRMAN PETTI: Yes, Hossein, we would

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

2                   MR. ESMAILI:   Okay.

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

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

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

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

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

25                   And so, what you can see in the map here



1 is that we have approximately about 30 countries that  
2 are participating in this program -- you know, from  
3 North America to Europe to Asia.

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

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

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

25 There's also a large user base worldwide.

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

5 Next slide, please.

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

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

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

22 And from this experimental program, the  
23 international severe accident community did a series  
24 of separate effects experimental programs to study  
25 phenomena for which the code did not capture the

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1 behavior. And we are constantly monitoring, and we  
2 are involved in some of these studies.

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

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

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

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

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1 They are looking at uncertainties in severe accidents.

2 Next slide, please.

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

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

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

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

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

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

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

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

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

24          Next slide, please.

25          So, NRC joined a cooperative research

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

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

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

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

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1 Next slide, please.

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

4 Next slide, please.

5 So, MELCOR is used extensively in the  
6 design certification of new reactors. I'm showing you  
7 some pictures of what we have used in the past and  
8 what are the newer designs. It's used for both severe  
9 accident response -- you know, this is mostly when  
10 you're talking about the LWRs, Chapter 19, and source  
11 term, and the containment response to a design basis  
12 accident. This is what we cover usually under Chapter  
13 6.

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

21 And as far as severe accident mitigation  
22 is concerned, there are design differences between  
23 some of these designs. For example, the EPR and APWR.  
24 EPR, what you see here, has a special core catcher and  
25 spreading compartment, and the containment is equipped

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1 with passive autocatalytic recombiners for hydrogen  
2 control. And the APWR, it's more active systems. The  
3 debris shifts through a combination of cavity flooding  
4 and enhanced area for debris spreading, and hydrogen  
5 control relies on the igniter. So, MELCOR provides us  
6 a tool that we can look at these different systems.

7 Next slide, please.

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

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

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

15 Thank you all.

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

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

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

24 MR. SNODDERLY: This is Mike Snodderly.

25 I think you're like four slides behind.



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

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

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

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

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

24 And so, a few years ago, we started on  
25 performing the UA analysis for selected accident

1 scenarios in Surry, Beach Bottom, and also, for  
2 Sequoyah.

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

9 Next slide, please.

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

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

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1 modeling with SOARCA.

2 Next slide, please.

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

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

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

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

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

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

24          And this is understandable because the  
25          hydrogen that's produced in-vessel is generally more

1 in the time to hot leg failure and the hydrogen burns  
2 longer. Then, the pressurizer safety valve fails to  
3 close. And we need a longer time to allow more  
4 efficient venting of the hydrogen to the containment.

5 Next slide.

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

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

19 So, we did this clustering on the  
20 righthand side. This is an example of the calculation  
21 in the certainty analysis that we did for the AGR that  
22 Jason is going to talk in a little bit, and we'll look  
23 at some of the, you know, the immersivity of the RCCS,  
24 for example; the graphite conductivity; heat transfer  
25 from the reactor cavity cooling system, and also, the

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1 blockage; you know, how well we get air cooling of the  
2 cavity. And so, we can understand how these  
3 parameters affect the results.

4 Next slide, please.

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

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

18 Next slide, please.

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

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1 MELCOR for accidents in the spent fuel pools.

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

7 Next slide, please.

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

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

20 So, it is also a highly sensitive. So, we  
21 tried to model this with MELCOR. Another thing, I  
22 think, in general, the MELCOR is capable of predicting  
23 the conditions of the experiments. The results were  
24 highly sensitive to both oxidation kinetics and  
25 transition to breakaway. But, in general, we showed

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1 good agreement as far as peak cladding temperature,  
2 ignition time, and ignition propagation is concerned.

3 Next slide, please.

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

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

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

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

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1                   Next slide, please.

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

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

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

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1 boxes here.

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

10 Next slide, please.

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

21 We looked at different conditions: the  
22 boundary conditions; availability of DC power; how  
23 operators control reactor pressure; operation of RCSI  
24 from what we learned from the Fukushima accident, and  
25 the mitigation during post-core damage. You know,

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1 water injection, is it into the driver or the RPV?  
2 And how do we vent the containment?

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

8 Next slide, please.

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

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

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

3                   Next slide, please.

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

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

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

19                   Okay. Next slide, please.

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

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

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1 go into the containment.

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

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

9 Thank you.

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

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

16 I was there for some of the earlier  
17 analyses. The very first MELCOR analysis back in the  
18 '90s for steam generator tube rupture, severe  
19 accident-induced tube rupture, where we didn't have  
20 countercurrent flow in the hot leg, and somebody  
21 asked, well, how are the things going to get into the  
22 steam generator? How is the steam generator going to  
23 heat up? And fission products? So, anyway, this  
24 really has come a long way, I believe.

25 So, next slide, please.

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

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

12           Next slide.

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

19           The graphic here shows accident  
20 progression. This is kind of a graphical depiction.  
21 It starts on the upper left and moves around to the  
22 bottom left. The first one is the beginning of the  
23 accident. The next one is dryout of the steam  
24 generator and the core starts to crumble. The one on  
25 the bottom right shows the core pretty much all in the

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1 lower plenum of the reactor. And then, the one on the  
2 lower left shows the core after it's left the reactor  
3 vessel and it's sitting at the beginning of the core  
4 catcher. And this is the EPR.

5 Next slide.

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

12 Next slide, please.

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

20 NuScale did this with MELCOR. They used  
21 MELCOR to estimate source term from the reactor into  
22 the containment. And part of their analysis and,  
23 typically, one of these analyses is the deposition in  
24 containment. And for that, actually, NuScale turned  
25 from MELCOR and actually decided to use STARNAUA for

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1 that. It's a separate code. It's a aerosol code. I  
2 understand it was developed years ago in Germany, and  
3 they use that.

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

10 Next slide. Okay, next slide, please.

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

14 As Hossein mentioned, we actually started  
15 non-LWR work a long time ago. For example, back about  
16 10 years ago, we were doing some work on HTGRs. So,  
17 here's where we are today:

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

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

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

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

8           Next slide.

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

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

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

23          And also, during NUREG-1150, we developed  
24          something called the EXOR code package, which, again,  
25          wasn't really a -- it was not a phenomenological code

1       like MELCOR is. It was factors, you know, release  
2       fractions from the core, release deposited in the RCS,  
3       et cetera.

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

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

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

18               Next slide.

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

21               Something both you and Hossein talked  
22      about raises a question for me. And that is, over  
23      time, as you develop these codes, various people would  
24      raise questions about what's there and what's not  
25      there, and does this apply to the accident

1 randomizing. And it caused more evolution as a code  
2 in development.

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

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

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

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

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

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

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

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

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

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

25 MR. SCHAPEROW: Well, it looks like

1 Hossein would like to add something.

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

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

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

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

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

25 For the surface plant models, we are

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

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

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

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

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

25 And finally, we're demonstrating how one

1 could use SCALE and MELCOR for non-LWR analysis, and  
2 how they can be used to help identify characteristics  
3 that are important in an accident, and also,  
4 uncertainties that need to be addressed, or maybe that  
5 don't need to be addressed.

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

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

19 Next slide.

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

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

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

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

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

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

20           Next slide.

21           DR. BLEY: Jason, Dennis Bley again.

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

25           Can you talk about -- thinking about the



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

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

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

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

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

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

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

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

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

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

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

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

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

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

24 So, the third part is to select accident  
25 scenarios. Again, we're turning back to our LWR

1 experience. We're doing the usual suspects, as you  
2 say, the ATWS scenario, the station blackout scenario,  
3 and the LOCA with no makeup scenario, those types of  
4 scenarios, the, quote, "severe accidents" where  
5 everything goes wrong.

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

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

19 Next slide.

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

25 We have a QR code here. If you use your

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

4 I see a hand raised. Joy?

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

7 MR. SCHAPEROW: Oh, I'm sorry.

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

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

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

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

24 The second part is to say, well, okay, how  
25 do you use the code to look and see maybe what's

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1 important, what you really -- what are the results  
2 sensitive to? And so, we did demonstrate, I think,  
3 how you would do that; how you would use MELCOR to  
4 show what the results are sensitive to.

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

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

13 MEMBER REMPE: Sure.

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

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

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

25 MEMBER REMPE: So, again, when I see some

1 of the applications coming through, I see a lack of  
2 data because we all know getting the data is a long  
3 lead time and very expensive process. And so,  
4 communicating to the design developers what data you  
5 had to make up, because they are, also, by the way,  
6 making up.

7 MR. SCHAPEROW: Yes, yes.

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

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

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

25 MR. SCHAPEROW: No, thank you.

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1 Next slide, please.

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

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

15 Okay. Next slide, please.

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

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

8           Okay. Next slide.

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

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

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

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

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

12 So, what MELCOR predicts at that point is  
13 the temperature really goes up even faster. At  
14 2200 K, we have an assumption that the control rods  
15 are -- the shutdown rods are inserted. So, that's it.  
16 So, the rods go into the core and all that's left is  
17 decay heat. And so, after 2200 K, you see the  
18 temperature going down because the heat from the fuel  
19 is now moving outwards, and the fuel into other  
20 materials in the reactor, and then, finally, into the  
21 cavity surrounding the reactor. This is a passively  
22 cooled reactor. The heat just goes from the outside  
23 of the reactor vessel into the cavity, and then,  
24 there's air circulation in the cavity.

25 Next slide.

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

5           Anyway, turning to fission products, this  
6           slide talks about fission products. Well, first of  
7           all, the top righthand graph shows the pressure in the  
8           heat pipes. So, when the heat pipes heat up to  
9           2200 K, the pressure goes up from 1 bar to almost 6  
10          bar. And then, as the heat pipes fail, then the  
11          pressure comes back down to 1 bar.

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

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

25          The next curve down is the green curve.

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

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

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

15 Next slide.

16 CHAIRMAN PETTI: Jason? Jason?

17 MR. SCHAPEROW: Yes?

18 CHAIRMAN PETTI: Just a question.

19 MR. SCHAPEROW: Sure.

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

25 MR. SCHAPEROW: Okay. Yes, so to answer

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

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

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

25 So, I don't know if that addresses your

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1 question.

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

5 MR. SCHAPEROW: Yes, really.

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

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

11 CHAIRMAN PETTI: Yes.

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

20 CHAIRMAN PETTI: Yes.

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

25 So, the next two sets of slides I have

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

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

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

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

20 Next slide.

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

25 So, the pressure dropped, as you can see

1 on the top, and then, the bottom viewgraph shows the  
2 countercurrent flow in the pipe between the reactor  
3 and the break location.

4 Next slide.

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

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

17 Next slide.

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

24 Next slide.

25 The third and final MELCOR model we

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

6 Next slide.

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

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

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

22 Next slide.

23 On the left side, I show the reactivity  
24 predicted in the MELCOR point kinetics model using,  
25 again, the SCALE inputs. The first thousand seconds,

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1 the reactor is heating up because of the accident and  
2 you're getting negative reactivity insertion. And  
3 then, after about a thousand seconds, you really see  
4 an effect of xenon. The xenon really takes over. And  
5 for the next 100,000 seconds, the reactivity is really  
6 down because the xenon is adding all kinds of negative  
7 reactivity.

8 And so, the next graph over to the right  
9 shows the core power going down the first thousand  
10 seconds, and then, continuing to go down. At about  
11 100,000 seconds -- well, 85,000 to be more precise  
12 -- the xenon is kind of gone. And guess what? The  
13 reactor's power starts coming up again.

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

23 Next slide.

24 CHAIRMAN PETTI: Oh, for these sorts of  
25 systems, that's exactly what I've seen in the HGTRs,

1 too. You know it's right when the power and the heat  
2 removal balance. Then, it's the new steady state  
3 until the rods get in, yes.

4 MR. SCHAPEROW: Okay. Next slide.

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

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

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

21 Next slide.

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

25 And again, to be a little redundant with

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1 what Hossein says, MELCOR simulates the entire course  
2 of the accident starting with the initiating event,  
3 with the hydraulic response, the fuel heatup, and  
4 then, the heat transfer out of the reactor into the  
5 reactor building, and then, into the environment. And  
6 then, last, but not least, the radiological release.

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

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

21 That's it.

22 CHAIRMAN PETTI: Hey, Jason, just a  
23 comment here. To me, the results also show the  
24 importance of the design of the building, right? And  
25 there's a message that has to go back to designers,

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1 because I think a lot of the advanced reactor  
2 designers, you know, everything's passive and I've got  
3 this functional containment, and I've got to make it  
4 by seismic, but there's an important fission product  
5 mitigation capability there that, as they are  
6 designing it, they should think about.

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

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

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

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

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

7 CHAIRMAN PETTI: Hold on, Michelle.

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

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

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

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

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

20 Next slide.

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

23 So, the outline of this is:

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

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

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

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

7           Next slide, please. Slide 82.

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

17          Next slide, please.

18          So, when we talk about the Emergency  
19          Planning Zone size justification consequence analyses,  
20          the concept is based on NUREG-0396, which is the  
21          technical basis for the current regulation which says  
22          you should have an Emergency Planning Zone for plume  
23          exposure of about 10 miles in radius and an ingestion  
24          pathway Emergency Planning Zone of about 50 miles in  
25          radius. And that is in the regulation, and it is

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1 based on the analysis in NUREG-0396. And it's based  
2 on identification of an area within which prompt  
3 protective actions may be necessary to provide dose  
4 savings in the event of a radiological release.

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

9 Next slide, please.

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

16 Next slide, please.

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

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1 they have not selected an Emergency Planning Zone for  
2 the Clinch River site.

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

10 And we also have a rulemaking underway for  
11 Emergency Planning Zone size determination for  
12 emergency planning for SMRs and other new  
13 technologies. And that's in a proposed new regulation  
14 50.160. We have just issued, for Commission review  
15 and approval, the SECY that goes along with the final  
16 rule, and there guidance on analysis in Appendices to  
17 the related Regulatory Guide 1.242.

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

21 Next slide, please. This is slide 86.

22 So, the NuScale EPZ Sizing Methodology  
23 Topical Report, which, as I said, is currently under  
24 review, it was not part of the design certification  
25 review, but it was submitted at the same time. It's

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1 currently on Revision 2. It's applicable to light  
2 water reactor SRMs such as NuScale, although it's not  
3 limited to the NuScale design.

4 And, in fact, in a response to a recent  
5 RAI, the scope of Revision 2 was reduced to advanced  
6 light water reactor SMRs. So, it's not for any  
7 advanced reactor, as originally envisioned in Revision  
8 2.

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

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

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

22 Next slide, please.

23 So, in this Topical Report, source term  
24 refers to the fission product release to the  
25 environment as a function of time. Unlike with

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1 discussions of Reg. Guide 1.183 or TID-14844, in which  
2 the source term is released to containment, it  
3 includes the containment transport in the  
4 determination of the source term.

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

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

20 MEMBER HALNON: Michelle, this is Greg  
21 Halnon.

22 MS. HART: Uh-hum.

23 MEMBER HALNON: My understanding is there  
24 was a question about the threshold of that CDF and  
25 what was below regulatory concern, since there was

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1 such a large spectrum of accidents looked at. Was  
2 that ever resolved or is that still a question on the  
3 table?

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

12 MEMBER HALNON: Okay. Because, during the  
13 50.160 discussions, that was one of their sticking  
14 points on trying to get the Reg. Guide and some  
15 guidance there. We kind of punted towards this  
16 Topical Report coming up with some resolution to that  
17 issue. So, hopefully, that will be a resolution, but  
18 we'll be looking at that in the future, I'm sure.

19 MS. HART: Right. I think there should be  
20 a caution, though, about the methodology resolution of  
21 the issue may not be a general resolution, because we  
22 are taking into account the specific design and the  
23 specific design interaction with the environment that  
24 may affect that. Because the major concerns that we  
25 have are about how to account for external hazards and

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1 the probability of those events that are initiated by  
2 external hazards. And that may not be a general  
3 concern for everyone. And so, there may not be a  
4 single way to look at that, but, like I said, it is  
5 still something that's still under review.

6 MEMBER HALNON: Okay. Thank you.

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

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

17 That SECY talks about how we're going to  
18 evaluate their change that they have just recently  
19 proposed -- "recently" being at the beginning of 2019  
20 -- which moved away from mostly doing what was in Reg.  
21 Guide 1.183 with some specific information from the  
22 plant to something a little bit more unusual, in that  
23 they wanted to use a design basis source term without  
24 core damage for environmental qualification of  
25 equipment, and for other purposes in the plant. But

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1 they were still using a Reg. Guide 1.183-like source  
2 term for the offsite analysis and control room  
3 habitability analysis.

4 Next slide, please.

5 So, to talk about the source term  
6 methodology, it is an approved methodology for the  
7 NuScale SMR design. Its method is to develop accident  
8 source terms that are consistent with the Reg. Guide  
9 1.183 guidance for PWRs, except for these two special  
10 events that they included, which is the core damage  
11 source term for the core damage event, which is the  
12 equivalent to the maximum hypothetical accident LOCA  
13 that we do in Reg. Guide 1.183, Appendix A, and the  
14 iodine spike design basis source term, which included  
15 no fuel damage, but release of the coolant to the  
16 containment.

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

25 The ACRS has reviewed the staff's

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

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

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

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

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

17           NuScale's core damage event is similar to  
18 a maximum hypothetical accident, in that it is not a  
19 specific scenario, but is intended to represent an  
20 accident with major damage to the core. NuScale's  
21 methodology is, in concept, similar to that used to  
22 develop NUREG-1465 and the Reg. Guide 1.183 LOCA  
23 source term. It is derived from a range of accident  
24 scenarios that result in significant damage to the  
25 core, informed by the PRA, and it is intended to be

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1 representative or bounding of a dominant majority of  
2 intact containment core damage events for the NuScale  
3 nuclear power module.

4 Next slide, please.

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

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

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

18 Next slide, please.

19 In the NuScale design certification  
20 application and in their Topical Report methodology,  
21 there was one release phase, unlike in Reg. Guide  
22 1.183, which has a gap release phase and an in-vessel  
23 phase.

24 MELCOR was used to estimate the release  
25 timing and magnitude for each scenario of those five

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1 scenarios, and the release onset and duration from the  
2 scenario with the minimum time to core damage was used  
3 for the core damage event source term.

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

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

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

17 Next slide, please.

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

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

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

7 MS. HART: Right.

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

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

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

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

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

14                  MS. HART:    Yes.

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

17                  MEMBER KIRCHNER: Michelle?

18                  Okay. Go ahead, Walt.

19                  MEMBER KIRCHNER: Yes, thank you, Dave.

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

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1 MS. HART: Right.

2 MEMBER KIRCHNER: Then, how do you  
3 proceed?

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

21 And depending on the information that they  
22 use to justify that, you know, if they have a full  
23 structured analysis of the accidents and their  
24 facility, and the risk from their facility, they may  
25 be able to justify an exemption from that particular

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1 requirement. Because, as we were talking about  
2 earlier, you know, there is this idea that, with like  
3 the Licensing Modernization Project, that you would do  
4 that full assessment, and in conjunction with  
5 functional containment, the features of the  
6 assumptions behind the regulation, that you would have  
7 this large release into a containment, but with a  
8 demonstrable leak rate that you would evaluate for  
9 this assessment, you know, none of those features may  
10 be there for your particular design.

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

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

18 MS. HART: Right.

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

22 MS. HART: Correct.

23 MEMBER KIRCHNER: And then, you're at a  
24 juncture where, does the staff have in its evaluation  
25 -- like you say, the pre-application meetings between

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1 the applicant and the staff will be very important in  
2 resolving it. But I see that as the stumbling block  
3 going forward with some of the non-LWR concepts.

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

6 Are there any more questions about this?

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

12 MEMBER KIRCHNER: Yes, that is what I  
13 mean.

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

24 MEMBER KIRCHNER: Precisely, Dave, yes.

25 CHAIRMAN PETTI: Right. And I guess the

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

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

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

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

25 I say that guardedly because we have to

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1 evaluate what potential things may actually appear.

2 Yes, Dr. Rempe?

3 MEMBER REMPE: Just call me Joy.

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

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

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

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

25 CHAIRMAN PETTI: Yes, I agree with you,

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

5 MEMBER REMPE: But a lot of them are using  
6 Part 50 and 52. So, I'm not sure, you know, if Part  
7 53 is the only place to think about it.

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

13 Okay. Let's keep going then.

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

15 John Segala?

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

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

24 CHAIRMAN PETTI: Okay.

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

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

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

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

24 Kairos, in their construction permit  
25 application, did use the SRP for non-power reactors,

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1 NUREG-1537, in preparing its application. And the  
2 maximum hypothetical accident which they developed is  
3 intended to bound all potential accident source terms.

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

13 Next slide, please.

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

19 The Topical Report was submitted in April  
20 2021, after they had previously submitted a white  
21 paper on the concepts that they were talking about for  
22 the source term verification and validation for the  
23 code suite, which they had submitted back in 2019 and  
24 we had given them comments on, or not comments, but  
25 feedback. When they submitted their Topical Report in

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1 2021, they did address some of those comments.

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

11           Next slide, please.

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

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

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

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

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

19          Regulatory engagement plans and other  
20          information on pre-application activities, including  
21          meetings and submittals on source term, may be found  
22          on the NRC's public website under "Nuclear Reactors,"  
23          and then, "New Reactors," then "Advanced Reactors,"  
24          then "Licensing Activities," then, finally "Pre-  
25          application Activities". And if you do have a copy of

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1 the slides, the blue underlined is a link to that  
2 site, and it will tell you what information we have  
3 in-house or are expecting.

4 Next slide, please.

5 And that ends my presentation on this  
6 particular topic.

7 CHAIRMAN PETTI: Thank you, Michelle.  
8 Very helpful.

9 Anybody have any questions before we break  
10 for lunch?

11 MEMBER DIMITRIJEVIC: Yes, I do.

12 CHAIRMAN PETTI: Go ahead, Vesna.

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

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

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

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1 there's no specific reason for it.

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

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

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

25           MS. HART: I think maybe the only thing



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

3 (Laughter.)

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

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

20 But thanks.

21 MS. HART: Yes, I think the thing that  
22 we're aiming for in that regulation, in that  
23 discussion, in that footnote is, you know, let's try  
24 to see what would be a bad thing that could happen to  
25 test the containment. And if you don't have a

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1 specific containment you're testing, well, what would  
2 be the bad thing, the worst thing that you would  
3 expect to use in your design basis for your siting and  
4 for your safety systems that you're crediting in  
5 retaining the radionuclides? It's a topic that is  
6 very nuanced and may depend on the information that  
7 you have in front of you.

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

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

19 MEMBER HALNON: Okay.

20 CHAIRMAN PETTI: I see some hands.

21 John? And then, Steve.

22 MR. PARILLO: Yes, this is John Parillo.

23 I just wanted to point out to the  
24 Committee that the Draft Guide 1389 does contain a  
25 definition of an MHA which is similar to the wording

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1 that Michelle has used, but I can read it.

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

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

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

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

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

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

9 CHAIRMAN PETTI: Steve?

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

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

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

25 And typically, when we're thinking of a

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

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

24 Then, when we start looking at facilities  
25 that might be somewhat larger, and just using examples

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1 of some of the medical radioisotope facilities that  
2 are designing to be less than 1 rem at the site  
3 boundary, that's when we've started looking at  
4 considerations for maximum credible accidents. And  
5 we've looked at that as being derived from the design  
6 basis accidents at the facility, where there is still  
7 an expectation that an applicant has looked at  
8 initiating events and all of the accident sequences  
9 within families, and then, they select a bounding  
10 event from those. And that credible accident is  
11 still, more or less, used as a conversation piece and  
12 a way of demonstrating the overall risk of the  
13 facility, but does not replace any other accident  
14 analysis that would otherwise be expected for the  
15 facility.

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

21 Thank you.

22 CHAIRMAN PETTI: Joy?

23 MEMBER REMPE: Sure. I'm glad you brought  
24 this up, Steve, and I'm glad you went and actually  
25 mentioned the fact that sometimes a design basis

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1 accident that's been evaluated by an applicant might  
2 suffice for the maximum hypothetical accident.  
3 Because I think there are some good insights, and I'm  
4 not sure that those insights are yet on the web page,  
5 or whatever guidance the staff's developing. And I am  
6 not sure that the design developers that are coming  
7 for Part 53 will be looking at the non-power  
8 applications, the impact applications. And so, I do  
9 think that there's a synergy that could be explored  
10 here, especially, again, if they do a systematic  
11 approach to come up with the initiating events.

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

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

16 Anyone else?

17 (No response.)

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

20 Thank you all.

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

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

1 Michelle?

2 MS. HART: Hi. Welcome back.

3 So now, for the next topic, we'll be  
4 talking about the accident consequence-related  
5 regulation activities, and there are several  
6 regulation development activities ongoing where source  
7 terms will be used. And that's in addition to Part  
8 53, which, of course, is happening.

9 Next slide, please.

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

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

24 Next slide, please.

25 CHAIRMAN PETTI: Michelle, does it say who



1 petitioned this?

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

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

7 MS. HART: Yes.

8 CHAIRMAN PETTI: Thanks.

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

12 CHAIRMAN PETTI: Okay. Thanks.

13 MS. HART: So, the next topic is the  
14 emergency preparedness for small modular reactors and  
15 other new technologies rulemaking. The stage we're in  
16 now is the final rule is in development. And as I had  
17 mentioned previously, there is this new section,  
18 50.160, and related and conforming changes to other  
19 portions of Part 50.

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

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1 1.242, to be issued with the final rule, currently  
2 discusses accident consequence analysis and  
3 information on source terms.

4 We did present the draft for Reg. Guide  
5 1.242 to the ACRS, which, when issued for public  
6 comment, had two appendices: Appendix A, on the  
7 methodology for analyses to support determination of  
8 plume exposure pathway EPZ size, and Appendix B, which  
9 provided information on source terms.

10 Reg. Guide 1.242 includes further  
11 clarifications and consideration of the public  
12 comments. The changes incorporated into Reg. Guide  
13 1.242 were presented to the ACRS last fall.

14 Next slide, please. This should be slide  
15 101.

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

23 In selecting events for the Emergency  
24 Planning Zone size evaluation, an applicant should use  
25 licensing basis events, such as design basis events,

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1 beyond design basis events, and DBAs that they would  
2 determine through the LMP process, or the MHA, if they  
3 chose to use that in their licensing scheme, and  
4 severe accidents, as candidate radiological release  
5 scenarios.

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

11 Next slide, please.

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

23 Next slide, please.

24 Another rulemaking activity that's ongoing  
25 right now is the alternative physical security for

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1 advanced reactors. It is a draft rule and guidance in  
2 development right now.

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

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

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

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

24 CHAIRMAN PETTI: Michelle?

25 MS. HART: Yes?

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

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

6 CHAIRMAN PETTI: Okay.

7 MEMBER BROWN: This is Charlie Brown.

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

12 MS. HART: Right.

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

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

23 MEMBER BROWN: I guess I still don't  
24 understand. I mean, you're still going to have  
25 guards. You've got to have some type of

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1 administrative requirements in buildings.

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

4 MEMBER BROWN: Okay.

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

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

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

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

12 MEMBER REMPE: Michelle?

13 MS. HART: Yes, Joy?

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

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

25 MEMBER REMPE: Okay. That helps. Thank

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1       you.

2                   MS. HART:   Uh-hum.

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

6                   MR. SEGALA:   Hey, Michelle?

7                   MS. HART:   I'm sorry.   Yes?

8                   MR. SEGALA:   Yes, this is John Segala.

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

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

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

15                   MS. HART:   Yes.   Thank you, John.

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

23                   So, I think you would start with that same  
24       basis for the security rule, but there are other  
25       considerations you would have to bring into play, such

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1 as potential different release pathways and different  
2 initiators. So, that will be something that we would  
3 expect to discuss in the Reg. Guides that would go  
4 with that rule.

5 Next slide, please.

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

8 (No response.)

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

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

16 Next slide, please.

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

21 Then, I'll describe mechanistic source  
22 term.

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

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

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

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

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

14          Next slide, please.

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

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1 analysis.

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

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

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

25 And as I just discussed, there's an

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

6 Next slide, please.

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

20 When you go through determining which  
21 phenomena you need to model, and which systems,  
22 structures and components, you would credit in these  
23 analyses, there is a balance of prevention versus  
24 mitigation of the event. So, certain SSCs are safety-  
25 related or must be managed in a different way because

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1 they prevent the accidents from occurring. Then,  
2 other SSCs are there to reduce the radionuclide  
3 release or increase the containment of radionuclides.  
4 And so, they have special treatment as well.

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

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

23 Next slide, please.

24 So, as I had just kind of discussed,  
25 there's no requirement that you have a mechanistic or

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1 deterministic valuation. That's a choice that a  
2 designer can make. If you do use the Licensing  
3 Modernization Project, that assumes that you would  
4 develop a mechanistic source term and use a PRA.

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

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

22 I also note that the guidance in Reg.  
23 Guide 1.183 on radiological assessment, in other  
24 words, the portions that do not describe the light  
25 water reactors source term itself, such as guidance on

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1 dose conversion factors, briefing rates, atmospheric  
2 dispersion, things like that, is technology-inclusive  
3 in manner, and non-light water reactor applicants may  
4 find it useful in performance consequence analyses.  
5 And we are planning to clarify that in the revision to  
6 Reg. Guide 1.183.

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

13 If there are any questions?

14 (No response.)

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

17 So, this is the SECY-93-092 definition of  
18 mechanistic source term that the staff put forward.  
19 "A mechanistic source term is the result of an  
20 analysis of fission product release based on the  
21 amount of cladding damage, fuel damage, and core  
22 damage resulting from specific accident sequences  
23 being evaluated. It's developed using best-estimate  
24 phenomenological models of the transport of fission  
25 products from the fuel to the reactor coolant system

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1 for all holdup volumes and barriers, taking into  
2 account mitigation features, and finally, into the  
3 environs."

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

14 Next slide, please.

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

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

24 Transport of fission products can be  
25 adequately modeled for all barriers and pathways to

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1 the environs; and that the calculation should be as  
2 realistic as possible, so that the values and  
3 limitations of any mechanism or barrier are not  
4 obscured.

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

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

13 Next slide, please.

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

23 They are technology-inclusive and they  
24 tell what to do to develop accident source terms, and  
25 not specifically on how to do it. There are no

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1 specific methods or phenomenological models that are  
2 presented as "the way to do things." And they do not  
3 provide technology-related source terms or releases.  
4 In other words, they don't provide specific values  
5 that you would expect to see.

6 Next slide, please.

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

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

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

12 Next slide, please.

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

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

24 Next slide, please.

25 This is another figure that is taken from

1 the INL report. And it shows all of the different  
2 considerations and information that you would use to  
3 develop source terms to compare to regulatory  
4 criteria. And I know it's a nice picture, and it is  
5 described in more detail, or the concepts are  
6 described in more detail in the paper.

7 Next slide, please.

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

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

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

25 This is intended for facilities that have

1 a small enough initial inventory of source terms to  
2 meet radiological requirements upon a full release of  
3 the initial inventory. So, we don't expect it's  
4 something that the majority of designs may be able to  
5 use.

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

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

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

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1 go through this assessment.

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

5 (No response.)

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

7 Now I will talk about the lab report that  
8 we've gotten, which is a "Simplified Approach for  
9 Scoping Assessment of Non-Light Water Reactor Source  
10 Terms." And this was issued by Sandia in January of  
11 2020. And that's the ADAMS Accession Number for that.

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

20 The scoping method is applied to different  
21 reactor concepts, considering the performance of  
22 barriers to fission product release for these concepts  
23 under sample accident scenarios. The accident  
24 scenarios and sensitivity evaluations are selected in  
25 this report to evaluate the role of different fission

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1 product barriers in reducing the source term to the  
2 environment and associated offsite consequences.

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

9 Do you have any questions about this  
10 report?

11 (No response.)

12 Okay. The next slide. This is slide 118.

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

20 MS describes the objectives and the  
21 characteristics and attributes of the mechanistic  
22 source term analysis for the PRA element. It provides  
23 information on what to do to develop mechanistic  
24 source term, not specific assumptions, methods,  
25 models, or computer codes. And it describes such

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1 considerations such as timing, location, amount  
2 released, and the radionuclide transport barriers and  
3 transport mechanisms and associated uncertainties.

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

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

10 (No response.)

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

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

17 MS. HART: Yes. Sure.

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

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

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

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

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

22                  And so, I think it's like a pre-step to  
23                  you determining how you're going to do that. And  
24                  then, you could come talk to us and say, "This is what  
25                  we're planning." And we're saying, okay, yes, if you

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1 use this information from these guides, we understand  
2 where you're coming from.

3 MEMBER BROWN: How does that help an  
4 applicant? I mean, to me, it would be just confusing.  
5 I wouldn't know what would be acceptable and what  
6 would not be acceptable.

7 This is Charlie Brown.

8 MS. HART: Right. I think --

9 MEMBER BROWN: You've gone through a whole  
10 mess of possibilities, and, I mean, I'm confused. I  
11 apologize.

12 MS. HART: So, they all have -- I'm sorry  
13 -- they all have similar features. They all talk  
14 about considering the same things. They all talk  
15 about, you know, you need to quantify these things  
16 -- "these things" being, you know, whatever they are.

17 I think there are some issues, but there  
18 are so many different technologies; there are so many  
19 different approaches that they can take to licensing  
20 -- I mean, like they can determine to do a full  
21 mechanistic source term or they can determine to do  
22 something in between. Or they can determine to do a  
23 maximum hypothetical accident, which has a non-  
24 physical, obviously, bounding analysis, where they  
25 wouldn't have to do as much, I guess, justification of

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1 the assumptions that they had done, or detailed  
2 analysis to supply the justification, is probably a  
3 better way to say it.

4 So, I think we're trying to remain  
5 flexible for these different approaches for these  
6 different designs and different goals that these  
7 applicants would have.

8 MEMBER BROWN: Well, one of them is  
9 qualitative. Some of the other ones are more  
10 quantitative.

11 MS. HART: Uh-hum.

12 MEMBER BROWN: It just seems --

13 MEMBER HALNON: So, this is Greg. Let me  
14 try to frame it just a little bit.

15 These small reactors are technologies that  
16 are usually -- I'll just say they're probably  
17 commercial reactors or commercial facilities that will  
18 be out there trying to make a profit. And in that,  
19 there's going to be a tremendous economic pressure to  
20 have extremely certain design parameters.

21 And when you get into these situations --  
22 and I'm not sure if there's an answer to it or not --  
23 where you get this sort of "bring me a rock, and we'll  
24 let you know if we think it's okay," without that  
25 bright line to design to, there's inefficiencies and

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1       there's potential economic issues.

2               So, there's going to be this drive to push  
3       back. And I would hope that at some point we would  
4       get, after enough technologies have come in, that we  
5       could see the similarities in all of these and come up  
6       with a process that gives more certainty to the  
7       designers. That's the pressure, and there is  
8       frustration amongst the designers because we don't  
9       have, you know, how do you slice and dice the  
10      licensing basis events, you know, get rid of the 10-  
11      to-the-minus-12 events and talk to 10-to-the-minus-5  
12      events, and that sort of thing?

13              So, that's the frustration I think that at  
14      least I'm experiencing through this. It came out in  
15      droves when we were doing the 50.160 P rule. And I  
16      think you're going through the same type of  
17      discussions with NuScale.

18              MS. HART: Right. I think, you know, this  
19      transition period we're in -- or I don't even know  
20      that that's the right term -- but, yes, I mean, it's  
21      hard to be efficient right out of the gate. We're  
22      trying to be as flexible as possible and allow for  
23      different concepts to come in. So, that does leave  
24      some room for future refinements. And if we do see,  
25      as we do see commonalities and obvious -- well, I

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1 won't even say, "obvious." As we learn lessons as we  
2 go through, we may find in the future that there is  
3 some guidance that we can give that could help refine  
4 things and bring things into a more efficient space.

5 MEMBER HALNON: Well, I think it's  
6 essential that we look for them and we have a project  
7 in play that every time we do find one, that we can  
8 refine the guidance, so that it gets more certain, we  
9 should do that. We shouldn't wait five years, and  
10 say, okay, now what did we learn? Each time we find  
11 something that's going to be advantageous to the  
12 certainty of a designer, we should allow that to be  
13 used.

14 And I think the nature of Topical Reports,  
15 and being able to use them across the board, is good,  
16 except I keep on hearing, you know, this is very  
17 specific to NuScale, or very specific to Kairos, or  
18 it's very specific, and it causes a problem in the  
19 transposition of these types of methodologies to other  
20 reactors.

21 MR. SEGALA: This is John Segala with the  
22 staff in NRR.

23 I just wanted to add on the question that  
24 Joy had on the endorsement, I did want to add that,  
25 for the non-light water reactor PRA standard, we're

1 getting very close to putting out a trial use Reg.  
2 Guide endorsing that standard. So, yes, we're not  
3 endorsing the other two industry guidances, industry  
4 documents -- I'm sorry -- the National Lab documents  
5 that Michelle just presented on. Those are really  
6 just being provided for information to help the  
7 developers.

8 MEMBER REMPE: I get it, but it would be  
9 nice if -- again, it's their bailiwick. If they're  
10 not asking for it, I guess all you can do is say that  
11 and be very careful to say, "These are just out here  
12 for information. Use it. It doesn't mean that we're  
13 going to approve it." But, anyway, that's what I'm  
14 hearing.

15 MEMBER BROWN: Well, in my mind, this is  
16 similar -- one of the concerns I've had with Part 53,  
17 it's more generalized requirements as opposed to  
18 general design criteria. And if somebody submits  
19 that, then, all of a sudden, the NRC decides, well,  
20 that's not quite good enough.

21 And the source term issue, it seems to me,  
22 is going in the same direction: "Tell us what you'd  
23 like to do and we'll figure out whether it's okay or  
24 not." Flexibility is great, but that's really tough  
25 to design hardware, plants, anything to get that work

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1 done. It's almost like nobody wants to have any  
2 specific requirements at all. I'm exaggerating  
3 slightly, but that seems to be, if you're listening to  
4 these slides, it seems to be where we're going. I'm  
5 glad I'm not a designer.

6 So, I'll pass. I just think this is a  
7 concern. I've spoken my piece.

8 CHAIRMAN PETTI: So, to me, my bigger  
9 concern is the diffused nature of what's called,  
10 quote, "requirements." They're there, but you've got  
11 to go read six, seven, or eight documents -- and it's  
12 only the source term, the most important thing we care  
13 about in terms of protecting the public.

14 And that's why the idea of consolidating  
15 it, you know, a website, whatever, so that someone  
16 doesn't overlook something, (a) I think is good, but  
17 (b) it would provide confidence, right, that, yes, the  
18 NRC is doing its job and doing it well, and here's  
19 what it looks like.

20 I would think NEI would be interested in  
21 trying to put out something. Yes, it may be  
22 functional, it may be higher level, because it goes  
23 across technologies, but, as it is now, it is somewhat  
24 diffuse. And this has to do with the NPUFs versus  
25 power reactors. It has to do with the history of how,

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1 what's called the "source term tree" has grown, but it  
2 seems like it's worth stepping back and asking these  
3 sorts of questions at this juncture, given what the  
4 future may hold in terms of advanced reactor  
5 applications coming in.

6 Keep going, Michelle.

7 MS. HART: I'm sorry. Okay.

8 So, I will be handing off the presentation  
9 to Bill Reckley now.

10 MR. RECKLEY: Okay. Thanks, Michelle.

11 So, thank you, everyone.

12 And I only have a few slides, but I'll go  
13 back and visit a few of the comments, and not  
14 disagreeing with anything anybody has said.

15 Part of the challenge is, as you provide  
16 flexibility and try to make things technology-  
17 inclusive, addressing a variety of reactor  
18 technologies, it gets harder to be specific and  
19 produce anything like a TID document or even  
20 NUREG-1465 and say, "Use this source term and it's  
21 going to be used as a confirmation of a specific  
22 barrier like the primary containment for a BWR or  
23 PWR."

24 But, that said, just to kind of summarize  
25 licensing modernization -- and we've talked to the

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1 Committee and this Subcommittee a few times on this  
2 -- it provides a risk-informed approach to the  
3 selection of licensing basis events and categorizes  
4 them into the categories we've talked about  
5 -- anticipated operational occurrences, design basis  
6 events, beyond design basis events, and then, another  
7 category of design basis accidents that are used kind  
8 of in a traditional deterministic method.

9 Those event categories are assessed and  
10 compared to acceptance target figures to test specific  
11 barriers and to assess the margin to the frequency  
12 consequence target and design objectives, such as a  
13 lower-dose value to justify an Emergency Planning  
14 Zone, as Michelle talked about under the 50.160  
15 rulemaking.

16 And this also includes an assessment  
17 against cumulative risks, using the QHOs and some  
18 other measures introduced by the NEI 18-04.

19 And then, just a summary that there are  
20 key roles in the LMP, as we've talked about before,  
21 for probabilistic risk assessment and, also,  
22 mechanistic source term. So, I don't know.

23 If we can go to 120?

24 This is a figure similar to what Michelle  
25 showed from the lab reports, and that's not a

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1 coincidence. We were all starting from the same  
2 fundamental source document, which is a DOE order on  
3 assessing radiological releases from any facility.  
4 And so, it's simply modeled as an inventory, and then,  
5 various barriers, and then, the integrity or the  
6 performance of those barriers for particular scenarios  
7 as it relates to particular radionuclides, and as it  
8 relates to time, as the transients are being modeled.

9 So, Michelle had the same equation.  
10 Obviously, we don't have it down yet to being able to  
11 model things using an equation like this. But you  
12 can, if you go back to Hossein's presentation and  
13 Jason's presentation, you can see that these things  
14 are what's being modeled in things like MELCOR, the  
15 transient response of particular barriers and the  
16 transport of radionuclides against those barriers.

17 So, one way to look at this, for example,  
18 when Michelle talked about the Sandia report, in  
19 looking at a first systematic way to look at it, it  
20 would be, for a particular design, where am I going to  
21 put the emphasis on retaining fission products? So,  
22 under the current light water reactor model, a lot of  
23 emphasis is put on barrier four, the containment  
24 building. Under HTGRs, there's a much larger focus on  
25 barrier one, the fuel, and maybe barrier two, the

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1 matrix around the particles. In other cases, or for  
2 other transient scenarios, since mechanistic source  
3 term is done on a scenario basis, it might be the  
4 pressure boundary or some other consideration.

5 And I know it's complicated. And I have  
6 said, when it comes to helping the developers, if they  
7 don't know it's complicated, they're getting into the  
8 wrong business. They have to understand that, when  
9 they're given these design choices and flexibility,  
10 then they have responsibilities.

11 So, when you go to how they implement this  
12 maze, if you will, let's say they are going to  
13 emphasize barrier one in their design. That's where  
14 they want to put the emphasis.

15 Then, Arlon, if you could do me a favor  
16 and go back up to 115? Okay.

17 This is another complicated figure, but  
18 what does it mean? It means, where are they going to  
19 have to do their research? Where are they going to  
20 have to put their emphasis? What does the R&D have to  
21 go? Where are they going to try to minimize the  
22 uncertainties in any particular barrier? If they're  
23 putting the emphasis on the TRISO particle, there's  
24 got to be a lot of work by DOE and by the developers  
25 to show that they've addressed that qualification of

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1 the TRISO fuel. On the other hand, should they choose  
2 to say, "I'm going to put my emphasis on another  
3 barrier," then, they're going to have to do the R&D on  
4 that other barrier.

5 So, again, it is complicated and it's  
6 iterative, how a designer is going to have to walk  
7 through this. But, again, given that we're trying to  
8 address a lot of different technologies, and we're  
9 trying to give flexibility to the designers as to  
10 where they put the emphasis in their design and in  
11 their analysis, it's just the nature of the beast that  
12 it won't be as clear as giving them a spreadsheet with  
13 radionuclides and saying, "Here, put this into this  
14 volume and model your containment."

15 And I oversimplified. I don't mean to --  
16 even the current process is not as simple as that,  
17 but, anyway --

18 So, Joy, I see you've got your hand up.

19 MEMBER REMPE: Yes. You're right, and  
20 what you just said would be good to have in a document  
21 somewhere.

22 CHAIRMAN PETTI: Amen, amen, amen.

23 MEMBER REMPE: The other thing I have to  
24 say is --

25 CHAIRMAN PETTI: I completely agree.

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1 Bill, it was good.

2 MEMBER REMPE: Let me finish, Dave. Let  
3 me finish, Dave. Okay?

4 CHAIRMAN PETTI: Okay.

5 MEMBER REMPE: Old farts like me can  
6 recognize this equation really comes from NUREG-1150.  
7 This is not something new. But perhaps the new design  
8 developers are not old farts like myself. Okay?

9 (Laughter.)

10 So, yes, and what you're saying is very  
11 good, Bill, and as Dave is saying, "Amen," and singing  
12 a chorus here. It ought to be written somewhere. Is  
13 it written somewhere?

14 MR. RECKLEY: I think the reports that we  
15 commissioned talk about it in this sense, when you  
16 look at them. And, for example, the Idaho report,  
17 then, goes on to further say things like Michelle was  
18 describing. You have an option in there at some point  
19 to say, "I'm going to address the uncertainty by being  
20 conservative. There's things I don't know in terms of  
21 the behavior of the radionuclide, or the behavior of  
22 a barrier. I'll just be conservative," and that's an  
23 option that's available to the designer as well to get  
24 through the licensing process.

25 So, whether it's as clear as it should be,

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1 I'm not sure. As John was saying, one thing we can do  
2 -- or even some of the members were saying -- bring  
3 this up at a public meeting and ask the developers,  
4 who are the ones we're trying to do this for, is it  
5 clear; is this understandable? Between the website,  
6 the reports, the history, is it clear? I don't know.

7 MEMBER REMPE: Okay. I think you've got  
8 our point. But Dave may want to expand --

9 CHAIRMAN PETTI: No, no, I agree. That's  
10 why I keep thinking of this, sort of an executive  
11 summary that pulls all this together. I'm thinking  
12 three-, four-, five-page manual, heavily annotated.  
13 So, it's kind of a roadmap, but the important things  
14 get put there, so that it helps them put all these  
15 different documents in context.

16 Because, you know, it is diffused right  
17 now, and I'm just trying to figure out, is there a way  
18 to better focus it, just to help them? I agree with  
19 you that you still need the flexibility, and it's  
20 never going to be as cut and dry as "Do X, and then,  
21 do Y, and then, do Z." But if we can make sure that  
22 they know what the considerations are in each step,  
23 and what's important, that's, I think, the best you  
24 can do at this point.

25 MR. RECKLEY: Okay. Well, I guess -- and

1 John will pick this up maybe at the end with the  
2 website -- that's a step; that's our effort to take a  
3 step in that direction. And you guys haven't had a  
4 chance to look at it yet, and neither have the  
5 external stakeholders. Maybe the next step is to see  
6 if that provided additional clarity where we're  
7 seeking it.

8 So, Arlon, if you can go back up to 120?

9 MEMBER REMPE: While you're going up  
10 there --

11 MR. RECKLEY: Yes?

12 MEMBER REMPE: -- again, I'm thinking  
13 about websites and how they can easily be changed.  
14 And it's nice to have the website, but you always have  
15 to say, "Last accessed," and what date you did,  
16 because --

17 MR. RECKLEY: Yes.

18 MEMBER REMPE: -- it can easily be  
19 changed. And so, again, I think something that's a  
20 little more concrete might be a good idea.

21 MR. RECKLEY: Okay.

22 MR. SEGALA: Yes, this is John Segala,  
23 Bill.

24 I just wanted to add, I mean, now that we  
25 made the website, the web page on source term,

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1 publicly available just earlier this week, you know,  
2 we were planning to share that at an upcoming  
3 stakeholder meeting, just to make everybody aware that  
4 we have this new page. So, just throwing that out  
5 there.

6 MR. RECKLEY: Okay. And, Arlon, if we can  
7 go to 121?

8 It's basically a very similar  
9 representation, just coming at it from a different  
10 direction. And as I just mentioned, and the bullet  
11 emphasizes here, there is flexibility provided on how  
12 they want to develop the safety case, both to reflect  
13 the design and within the analysis.

14 And that involves combinations of active,  
15 engineered safety features, passive safety features,  
16 and increasingly, reliance on inherent properties of  
17 materials in the reactor cores. And that's  
18 consistent, as it says here in the second bullet, with  
19 the Advanced Reactor Policy Statement encouraging  
20 passive and inherent features.

21 But sometimes, those also bring in  
22 uncertainties and things you need to address in both  
23 the design and in the analysis.

24 So, that's just a quick summary, again, of  
25 licensing modernization; the key role that modeling

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1 the mechanistic source term has in that.

2 So, with that, if there's no other  
3 questions or suggestions, we can go on to fuel  
4 qualifications and Tim.

5 MEMBER KIRCHNER: Dave, may I make an  
6 observation? This is Walt Kirchner.

7 CHAIRMAN PETTI: Sure.

8 MEMBER KIRCHNER: You know, Dave, and  
9 going back to this morning, we have this DG -- let me  
10 make sure I get the number right -- DG-1199. Is that  
11 an appropriate place to at least lay out this generic  
12 approach to source term analysis, identification and  
13 analysis?

14 CHAIRMAN PETTI: I think that's RWR-based.

15 MEMBER KIRCHNER: I know it is, but,  
16 conceptually, it seems to me, at least what I do is,  
17 each time we see a new document for non-LWRs, my first  
18 proof of concept or test of the document is, would it  
19 work for an advanced LWR? And I think they will, most  
20 of them.

21 And the advanced LWR, of course, is where  
22 we've got the most experience. We've got very mature  
23 PRAs. We've got equipment reliability to feed into  
24 the PRAs. We've got a good understanding of the  
25 technology and vulnerabilities, et cetera, et cetera.

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1           So, you know, in my mind, when we look at  
2           these new approaches, I'm testing them on an advanced  
3           LWR, and I think they will work. So, conceptually,  
4           the charts we were just looking at on mechanistic  
5           source term, I mean, those could be used for an  
6           advanced LWR as well.

7           I thought what Michelle was covering, that  
8           synopsis of the methodology from the INL report was  
9           useful because you started off trying to bound the  
10          problem, and then, you worked further down. If you've  
11          got the data, if you've got the experimental data, if  
12          you've got the experience with the technology, you can  
13          probably go down further. As Bill was saying, this is  
14          complicated, especially for newer technologies where  
15          you don't have the kind of databases, and such, and  
16          experience.

17          It just seems to me that this DG-1199  
18          might be a good place to put the conceptual approach  
19          in, in a technology-inclusive manner, and then, go on  
20          to --

21          CHAIRMAN PETTI: My view is, if there was  
22          a place that it needed to be put, it would be in an  
23          appendix to the NUREG on fuel qualification that Tim's  
24          going to talk about. Sorry, Tim. To me, that was  
25          where it made most sense. But I would leave that up

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1 to the staff to decide where to put it. But there's  
2 a lot of --

3 MEMBER KIRCHNER: But that's just fuel.  
4 To do the source term --

5 CHAIRMAN PETTI: No, the source term is --

6 MEMBER KIRCHNER: -- you've got to have  
7 the barriers.

8 CHAIRMAN PETTI: It's talked about in  
9 there. You'll hear it. You'll hear it from him.  
10 That stuff is in there. But, again, I'd leave those  
11 details sort of to the staff, but --

12 MEMBER KIRCHNER: That was just a comment.  
13 No need -- no answer required.

14 MR. DRZEWIECKI: Okay. So, this is Tim  
15 Drzewiecki from the staff.

16 Sorry, can you guys hear me?

17 CHAIRMAN PETTI: Yes, go ahead.

18 MR. DRZEWIECKI: Okay. Wonderful. Okay.  
19 Yes. So, yes.

20 So, I've got a few slides to just kind of  
21 give an idea of what's in this document. This is the  
22 fuel qualification guidance that we came out with and  
23 we talked about at the end of last year.

24 And what it kind of lays out is a top-down  
25 method to kind of highlight a list of goals or

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1 criteria the staff could evaluate to say that this  
2 fuel is qualified and it does have certain aspects  
3 that I'm going to touch on source term. So, I'll kind  
4 of walk through how it relates to the source term in  
5 the next couple of slides.

6 So, may I have the next slide, please?

7 So, how it starts out is we're trying to  
8 make a finding that the fuel is qualified for use.  
9 And there are two kind of conceptual things that we  
10 want to make findings on for that. One, that you have  
11 the manufacturing specification to control the key  
12 fabrication parameters, and the other puts that the  
13 safety criteria can be specified. Obviously, the  
14 safety criteria are not well-defined. So, we break  
15 that down more on the next slide.

16 So, the next slide, please.

17 And in here, specific criteria are coming  
18 from the NRC regulations. The box on the left, that's  
19 talking about margin on the safety limits under normal  
20 operation and the AOOs. And the one in the middle,  
21 which is the focus of this presentation, is showing  
22 that you can maintain the margin of a radionuclide  
23 released when it's under accident conditions.

24 Now the point is, for this report, we were  
25 focused on the fuels role, but it does talk about

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1 other barriers, too. And this goes into the role that  
2 fuel plays in the protection against the release of  
3 radionuclides.

4 And the box on the right, that just goes  
5 into, basically, showing that you can maintain a safe  
6 shutdown condition.

7 Next slide, please.

8 Okay. And so, that box that was in the  
9 middle, it's broken down into four other criteria. I  
10 want to say that the boxes that are gray are gray  
11 because they're considered these base goals, in the  
12 sense that we would expect to be able to have enough  
13 evidence to make a finding on that item and not have  
14 to break it down any more.

15 So, in terms of the box on the left, we  
16 want to make that the fuel performance envelope is  
17 defined. In other words, for this goal, it's knowing  
18 what kind of accidents that the fuel is going to be  
19 credited to perform under. So, whether it's under-  
20 cooling events, overpower events, things like that.

21 The box second to the left is, basically,  
22 specifying what's the radionuclide retention  
23 requirements of the fuel under accident conditions.  
24 This is going into the role of fuel as it plays in the  
25 safety case, in the sense that, if you're not

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1 crediting your fuel, if you have other barriers,  
2 that's going to have an impact on the kind of data  
3 that you would need to support that.

4 These other two boxes on the right, these,  
5 I think, line up pretty well with the source term, a  
6 slide that was shown on -- both Bill and Michelle had  
7 shown it. So, these are associated with, one, the  
8 barriers to the release of radionuclides, showing that  
9 you're conservative on criteria there. So, this  
10 should be things like propellant/clad/mechanical  
11 interaction, your limits, or it would be other things,  
12 too, like whether it's, say, like melting, and things  
13 like that.

14 And then, the box on the right, that's  
15 talking about the release of radionuclides or the  
16 migration of radionuclides within the fuel matrix.

17 So, I was going to break down the box,  
18 that whole G2.2.2 on the next slide. Yes.

19 Essentially, this has two boxes. One is  
20 that you can show the criteria are conservative, which  
21 we think is something that can be shown, as long as  
22 you have data. This assumes that you're comparing it  
23 against experimental data.

24 Then, the box on the right is just making  
25 a statement that experimental data is appropriate.

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1 And that box is blue because there's a whole separate  
2 framework that's going to break that down. And that  
3 goes into things like data independence. Is that data  
4 applicable to the scenario, things like that?

5 And so, what I have on this next slide  
6 -- may I have the next slide, please? -- so, this is  
7 not really meant to be read, but it kind of shows you  
8 what this framework is. It takes a high-level goal  
9 and it breaks it down into, effectively, like a  
10 checklist, a bunch of things that we want to make  
11 findings on. And if we can make findings on all of  
12 those gray boxes, then we can say that the fuel is  
13 qualified.

14 And there is some overlap here. And what  
15 I mean by that is the main framework is the one on the  
16 left, and that relies on things like having evaluation  
17 models in a couple of sections and experimental data  
18 is used to support multiple goals as well.

19 So, that's all that I had for NUREG-2246,  
20 unless there's any questions.

21 (No response.)

22 Arlon, can you hear me? Okay.

23 MR. COSTA: Yes, we can hear you.

24 MS. HART: I guess if there are no  
25 questions, we can go to the next slide.

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1           So, I wanted to talk a little bit about  
2           this web page that we keep referring to, the non-light  
3           water reactor accident source term web page  
4           information. At the top, there's the link to it.  
5           It's on the public website and it's underneath the  
6           advanced Reactor-related documents.

7           And it's a one-stop shop for existing  
8           information, and it's discussion of accident source  
9           terms. There's a link list of documents relevant to  
10          development of non-light water reactor accident source  
11          terms for licensing. It includes some of the light  
12          water reactor information. We're not trying to be  
13          comprehensive. Right now, we are putting information  
14          out there and we will keep it up-to-date and keep  
15          adding to it, as we continue to go through this  
16          process.

17          Arlon, if you could open the website, so  
18          they can take a look at it?

19          So, we have a little bit of a preamble  
20          that talks about what a source term is and a little  
21          picture of the barriers and kind of the barriers  
22          assessment.

23          And next, we go through the history and  
24          evolution of the light water reactors source term.  
25          So, you have some of that history.

1           And then, it kind of follows along the  
2           lines of what we've been talking about today. We do  
3           have the links in the text here to TID-14844 and Part  
4           100, Reg. Guide 1.183.

5           We talk a little bit about the analytical  
6           tools and past studies. So, we have references to  
7           RADTRAD and MELCOR.

8           And then, we talk about information on  
9           SMRs and non-light water reactors. So, we do have a  
10          discussion to our mechanistic source term and how it  
11          interacts with the regulatory analysis and the siting  
12          analysis that we talked about in SECY paper 16-0012.

13          And then, we talk about the SECY-93-092,  
14          which was kind of the base for the information on what  
15          a mechanistic source term is and the considerations  
16          for what the staff would look at for a good  
17          mechanistic source term.

18          And then, further down on the page, we  
19          have a section where we would talk about accident  
20          consequence-related regulation activities. We don't  
21          have links to those right now at the moment, but it  
22          says that there are things that we're going through  
23          now.

24          And then, we have the guidance and  
25          information for developing advanced reactor source



1 term with reference to the LMP, is the first thing,  
2 the Regulatory Guide that endorses the LMP.

3 And then, we go through document links.  
4 And so, you can see that we have a link to TID-14844,  
5 NUREG-1465, Reg. Guide 1.183. And then, Hossein's  
6 discussion about the history of source terms.

7 We have a link to our vision and strategy  
8 code development plans for severe accident analyses.

9 We have some links to those demonstration  
10 projects that Hossein and Jason discussed earlier  
11 today. So, we do have the slides and a video  
12 recording of those presentations from the workshops  
13 available on this site.

14 A little bit further down, another link to  
15 SECY-94-302, which is about source term-related, you  
16 know, the information on source terms for evolutionary  
17 and passive light water reactors. So, it's not non-  
18 light water reactors, but it's that kind of  
19 information.

20 And we have the wrong SRP here. It should  
21 be 15.0.3 for new reactors. We can fix that.

22 We do have a link to the approved NuScale  
23 Topical Report for Accident Source Term Methodologies.  
24 So, if somebody wanted to see an example of an  
25 approved methodology for light water reactors SMRs,

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1 and we do have the SECY, our staff approach to  
2 evaluating the source terms for NuScale.

3 And then, here, we have some SECYs and  
4 some NUREGs and other information on the source term  
5 approach for non-light water reactors, including  
6 SECY-93-092 and it's SMR; some pre-application Safety  
7 Evaluation Reports for several of the designs, and  
8 then, an assessment of a white paper from NGNP on fuel  
9 qualification and mechanistic source terms. So,  
10 that's our staff assessment.

11 In the next section, we have some guidance  
12 on developing advanced reactor source term. And so,  
13 for that, we have the endorsement Reg. Guide for  
14 NEI 18-04.

15 We do have the Reg. Guide for the  
16 performance-based emergency preparedness for small  
17 modular reactors and other new technologies.

18 And then, we have the lab reports that I  
19 described today and a link to the ANS standard, which,  
20 of course, we are working on an endorsement Reg. Guide  
21 for that as well.

22 And then, the next section has several  
23 reports from labs and from other places about source  
24 term information on designs that we've seen in the  
25 past.

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1                   And then, the next section is  
2 presentations on non-light water reactor source term,  
3 and I expect that we will probably be putting this  
4 presentation and transcript, a link to that here as  
5 well.

6                   Like I said, this is what we put so far.  
7 And we do intend on keeping that up-to-date. And I do  
8 like the idea that Joy was talking about, that maybe  
9 we need to have a "last updated" note on there.

10                  CHAIRMAN PETTI: Yes, the other thing you  
11 could do is, if something's new, sometimes like  
12 websites put a little tag, "New" next to it.

13                  MS. HART: Uh-hum.

14                  CHAIRMAN PETTI: So, it draws the eye of  
15 people visiting it.

16                  MS. HART: Yes.

17                  CHAIRMAN PETTI: I'm just wondering,  
18 nothing is there about NUREG-1537 for the NPUFs, and  
19 that might be useful, to consider bringing in some of  
20 those documents, too.

21                  MS. HART: Right. And I think that's a  
22 good piece of information. I think there's also been  
23 some discussion about maybe some information that they  
24 had used when they developed the source terms for like  
25 SHINE, or whatever. We can look into what information

1 we can include on that.

2 CHAIRMAN PETTI: Right, right. I think  
3 that would be good.

4 And, I mean, I like that introduction  
5 stuff. That's what I was kind of thinking of as  
6 needed. So, I think it's a good flow of all that.

7 MS. HART: Yes. And it's a very high  
8 level right at the moment. As we go through learning  
9 things again, you know, and as we get some comments  
10 from folks as they look at it, not that we're  
11 requesting comments officially, or anything like that,  
12 but if there's missing pieces of information or  
13 information that doesn't make sense, certainly, we'll  
14 take that comment into consideration.

15 MEMBER REMPE: The intro is good, but some  
16 of the things that Bill mentioned about where you  
17 emphasize and the barrier diagram might be good to  
18 consider --

19 MS. HART: Uh-hum.

20 MEMBER REMPE: -- in the actual formation,  
21 the introduction. But, again, that's just one  
22 member's comment.

23 MS. HART: Sure.

24 MEMBER REMPE: And we're getting into  
25 designing your web page, which I don't think ACRS

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1 members should be doing. But it's just something to  
2 think about.

3 MS. HART: Right. No. No, I understand.  
4 You know, there could be more pictures and more  
5 discussion. Or they could be more callouts to other  
6 documents, as necessary.

7 So, as we go through this, and if we hear  
8 from industry or anybody about what they think is  
9 useful, we can take that into consideration. I can't  
10 make any promises necessarily, because we're not  
11 asking for guidance on how to do our website  
12 necessarily, but we thought that this would be a good  
13 way to gather this information. And it may be a  
14 little bit more flexible than trying to write like a  
15 series of white papers or SECY papers, or something,  
16 which takes a lot of time and effort to try to get all  
17 the language correct.

18 MEMBER REMPE: And I do want to say, I  
19 were a developer, having all the documents in an  
20 accessible place would help big time, and sharing your  
21 knowledge this way is good. It's just it's very  
22 fluid, I guess. I'm still thinking that some sort of  
23 paper document might be good to be thinking about.

24 MS. HART: Yes, we can also think about  
25 this as we go through things. I will say, you know,

1 a certain amount of what is on this web page is  
2 constrained by the way that the NRC website is  
3 generally configured. So, there may be some of these  
4 good ideas that you have that maybe aren't  
5 implementable or generally haven't been implemented on  
6 our website. So, I'll just add that as a thought to  
7 throw out there.

8 MEMBER REMPE: Good point.

9 MS. HART: Are there any other questions?

10 DR. BLEY: Michelle, yes, this is Dennis  
11 Bley.

12 After you introduced it, I went over and  
13 found it and have been poking around in it. And I  
14 think it's a great start. And it's the kind of thing  
15 I think the Committee has been looking for for a  
16 while, and I think it will be very helpful to people.  
17 Thanks.

18 MS. HART: Right. And somebody did tell  
19 me that there is a timestamp on the page near the  
20 bottom that says the last time it was updated. So,  
21 it's not in big print or anything like that, but it  
22 does have it.

23 MEMBER REMPE: It is on there. But I  
24 guess I was thinking, if a person came in and said,  
25 "Well, I used document X that was available on January

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1 15th, 2021," or 2022, and in the ensuing two years  
2 before they came in, that document has been removed  
3 because the NRC has decided that wasn't a good idea to  
4 post it, they're kind of stuck. If you say, I used  
5 Reg. Guide 1.-whatever that was Rev. 3, the fact  
6 you've updated it, well, it was there at that time and  
7 there's more certainty with the document.

8 MS. HART: Right. I think, right now, we  
9 only have things that already existed. We haven't  
10 added new, different information, other than this  
11 introduction/preamble stuff. So, if you're using a  
12 link that's on the page, you should be able to find  
13 that again later, if you've taken down the  
14 information. Most of them are ADAMS documents; not  
15 all of them are, though.

16 And, of course, there are some pluses and  
17 minuses to that as well. I'm thinking of like  
18 specifically Reg. Guide 1.183. Right now, we have a  
19 link to Rev. 0. What happens when we have Rev. 1? We  
20 have to make sure that that's up-to-date.

21 MEMBER REMPE: And again, these are just  
22 comments, and you've emphasized ACRS members shouldn't  
23 be designing a website. But it's just, you know, we  
24 can't help ourselves sometimes.

25 MS. HART: No, I get it.

1 Are there any other questions or concerns?

2 (No response.)

3 All right. If not --

4 DR. BLEY: Well, yes. Now that you  
5 mention that there are constraints on NRC websites,  
6 I'm reminded of the briefings we've had on NRR's  
7 Venture Studios, and maybe they can be of help to you  
8 guys. And maybe you've already used them to make this  
9 more flexible and easier to use, and flexible in areas  
10 where you're looking for more flexibility.

11 MS. HART: All right. Thank you.

12 So, if there are no further questions, I  
13 guess we can move on to the next topic, and that will  
14 go to Bill Reckley.

15 MR. RECKLEY: Okay. Thanks, Michelle.

16 And I think the only reason I'm doing this  
17 is because I was gone in January. So, I stepped  
18 forward or everyone stepped backwards, or however that  
19 works.

20 So, just kind of summarizing things -- and  
21 I'll ask Michelle or John to jump in whenever it's  
22 appropriate -- but if we want to go to 131, Arlon?

23 Just kind of, as a summary and a general  
24 approach, using that same kind of complicated figure  
25 that Michelle used, but it does emphasize the level of

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1 effort that needs to go into this.

2 And we understand the general observation  
3 that reactor developers, and especially those that may  
4 be somewhat newer to this kind of interaction with the  
5 NRC, it's complicated and maybe it could be made  
6 simpler. On the other hand, this is really what it  
7 comes down to, right? I mean, this is, as I think  
8 Mark said in the very beginning, this is why the NRC  
9 is here, and that's to address the potential hazards  
10 of nuclear reactors and radioactive materials, and  
11 ensure that they aren't released to the environment.

12 And so, as you look at how that evolved  
13 for light water reactors since the 1950s or the TID in  
14 1962 to present day, that reflected the way that it  
15 was done, the importance of the containment, all the  
16 way back to the genesis of that, as part of the siting  
17 decisions of the Atomic Energy Commission. And it's  
18 worked over that period of time.

19 And so, as the first bullet says, if a  
20 developer wants to use that kind of an approach, if  
21 they're a non-light water reactor, they're going to  
22 have to show a conservative source term, and it may  
23 not be the same as NUREG-1465 because of the  
24 technology differences. But if they want to put an  
25 emphasis on a barrier like a containment and use that

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1 same approach, that remains available to them.

2 The other thing that the second bullet  
3 starts to go into is they can, also, decide to use a  
4 more mechanistic source term and a more LMP-type  
5 approach, where you are assessing the performance of  
6 barriers and the behavior of radionuclides for  
7 particular scenarios and assessing the system overall.  
8 That's the functional containment concept. You're,  
9 basically, looking at all of the barriers and their  
10 performance for various scenarios, as the LMP would  
11 allow you to do.

12 In either case, the third bullet, when  
13 you're dealing with a variety of technologies and  
14 designs, the actual implementation is going to be  
15 specific to that technology and that design. The  
16 source term for a molten salt reactor, and what  
17 radionuclides stay in the salt and what gets released,  
18 and what barrier they're going to rely on, is going to  
19 be different than a gas reactor or a sodium fast  
20 reactor. And that will go to my next slide to some  
21 degree.

22 The last bullet is the NRC is not  
23 currently planning to do the equivalent of a TID  
24 assessment or a 1465, where we give source terms to  
25 applicants and say, "If you use this, it's

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1 acceptable."

2 As Jason and Hossein mentioned, the work  
3 we do, we share, but that's somewhat different than  
4 giving them something in a Regulatory Guide and,  
5 basically, saying, "Use this as an input." And the  
6 reason we're not planning to do that is for a couple  
7 of reasons. One, the variety of technologies, and  
8 two, a general trend where that kind of scientific  
9 work, the burden of doing that is put over onto the  
10 applicants.

11 And if you want to go to 132, Arlon?

12 Obviously, the Department of Energy and  
13 the National Laboratories play a role in this. And  
14 so, as you know, the National Labs are organized into  
15 various campaigns for different technologies. All the  
16 labs associated with taking the lead for a particular  
17 technology are looking at the behavior of  
18 radionuclides and barriers and materials, and how that  
19 plays into source term.

20 I just threw up a couple of reports. The  
21 HTGR, that dates back from the NGNP, Next Generation  
22 Nuclear Plant, project from 10 years ago. Argonne and  
23 Oak Ridge are doing work on molten salt. Argonne is  
24 doing work to develop models for sodium fast reactors.

25 So, if we go to 133.

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1           The NRC's activities have been more  
2           focused on the methodology and how to use that  
3           scientific data, or how an applicant might use that  
4           scientific data, that they do by their own experiments  
5           or through reference to DOE activities, to support  
6           their design and to support their safety case, again,  
7           going back to the previous discussion of where they're  
8           going to put their emphasis. And both of these  
9           reports kind of echo the flexibility that designers  
10          have and where they might want to put their emphasis.

11           Joy?

12           MEMBER REMPE: So, I'm hearing a little  
13          bit different from you in the prior slide discussion,  
14          which maybe you're right, but I also think that the  
15          design developers need to understand that, too. I  
16          still think I like the discussion about, if you rely  
17          on that barrier, you'd better have more research.  
18          That's technology-independent, and I think that to be  
19          vetting guidance.

20           But you said, I believe -- and I'm  
21          paraphrasing -- that, because we've got so many  
22          technologies coming in and they're not using the  
23          standard LWR stuff, the burden is on the new design  
24          developer to come up with a source term. That's just  
25          the way it's going to be. If you're going to pick a

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1 new technology, you've got to realize you've got to  
2 have enough money to get the data and the approach for  
3 your source term. And maybe that ought to be in some  
4 guidance. Or is it communicated well to them, so they  
5 understand that?

6 MR. RECKLEY: Well, yes.

7 MEMBER REMPE: Am I misquoting you, by the  
8 way? Because I am kind of paraphrasing what I'm  
9 hearing.

10 MR. RECKLEY: No, no. I think it's an  
11 accurate paraphrase. The only thing -- Arlon, if you  
12 can go back to 132? -- is, and I think you guys are  
13 aware of this, all of these are complex relationships,  
14 right?

15 The Department of Energy is not an  
16 innocent bystander. They are a key player. And they  
17 do this through their own work. They do it through  
18 working with developers. They solicit input from the  
19 developers on where to do the research through things  
20 like Project GAIN. They work with individual  
21 developers, and some of the Project GAIN grants have  
22 related directly to developing source terms.

23 And we're not just standing by, either.  
24 We're interacting with DOE on where to do some of the  
25 research. And so, you do have all parties kind of

1 trying to stay cognizant of what each other is doing  
2 to make sure this works out. I don't want to make it  
3 sound like we're just stepping back. We're talking to  
4 the developers, as Michelle and others have said, in  
5 this pre-application discussions. Those same  
6 developers are talking to DOE. We're talking to DOE.  
7 And so, the hope is that, although it is complicated,  
8 that it is kind of an organized approach to this.

9 So, I'll leave it there, unless John or  
10 Michelle want to weigh-in on the interactions we have  
11 with the developers and the laboratories and DOE.  
12 Okay?

13 MEMBER REMPE: It helps. Again, it's just  
14 something I'm thinking about. I'm reacting to things  
15 I see in the popular press.

16 MR. RECKLEY: Right.

17 MEMBER REMPE: And I'm thinking that some  
18 points are worth emphasizing.

19 MR. RECKLEY: No, and I don't think any of  
20 us really have much different thoughts. I mean, in  
21 terms of the development of a technology, it's like a  
22 3-dimensional chess game here -- working with the  
23 developers, the DOE, the NRC, and other entities. As  
24 Hossein was mentioning, you can bring in the  
25 international elements. It's a complex set of

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1 activities to support any of these technologies.

2 MEMBER REMPE: Go ahead. Thank you.

3 MR. RECKLEY: Sure.

4 So, Arlon, if we want to go to 134?

5 You know, this is just kind of reinforcing  
6 that complexity. This is maybe the most famous  
7 simple, well, single figure representing a source  
8 term, and it goes back to the NGNP and HTGR  
9 discussions.

10 But, if you look at those DOE reports, you  
11 will see similar representations for molten salt  
12 reactors, similar representations for sodium-cooled  
13 fast reactors. And you can do this for any of the  
14 designs and the technologies. And it's just another  
15 way to try to represent sort of what was going into  
16 the barrier diagram, and just another way to try to  
17 represent mechanistic source term; and, again, the  
18 complexities of trying to, for individual scenarios,  
19 say which of these is going to come into play; which  
20 of these are going to contribute? In this particular  
21 case, when are the contaminants going to come off of  
22 the primary circuit and get released into the reactor  
23 building? When might they be retained, and so forth?

24 So, I didn't really have much of a  
25 message, other than just using this somewhat familiar

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1 capsule diagram to kind of reinforce the number of  
2 things going on and the complexities of developing a  
3 source term.

4 So, go to 135.

5 This is just, again, summarizing what  
6 Jason and Hossein were saying. Our activities in  
7 regards to this, developing the models, which we  
8 share, for at least the consideration of the  
9 developers, whether they're using MELCOR or some other  
10 system analysis code to support the design and  
11 licensing of their plant.

12 So, 136.

13 Again, this is, to some degree, just  
14 summarizing the previous discussions, as Michelle went  
15 through. We have had both applications and pre-  
16 applications with NuScale, along with Kairos, and  
17 then, pending discussions with Westinghouse,  
18 TerraPower, X-energy. And we'll consider those in  
19 terms of a step forward; what is useful to share.

20 As another developer comes in, we can talk  
21 to them about at least the public versions of the  
22 interactions with these other vendors, to say, you  
23 know, this is one approach that was taken. And if  
24 it's an approved Topical, then so much the better;  
25 that this approach was found to be acceptable.

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1 Slide 137.

2 So, moving forward, we, the staff, will  
3 continue to follow the scientific work being done by  
4 the National Labs and the developers. We continue to  
5 engage with those developers on the approaches they're  
6 taking in both the design and for presenting a safety  
7 case.

8 We'll continue, as Hossein and Jason  
9 mentioned, to develop, refine what we've done to date,  
10 and to develop models for the last two technologies we  
11 plan to do.

12 And in terms of the MELCOR workshops, we  
13 will engage stakeholders, including the meeting here  
14 today, and consider additional guidance, if we get  
15 feedback like the last bullet: the web page maybe was  
16 useful. But if stakeholders are commenting that it  
17 needs to either be revised, we'll take that into  
18 account, or we need to do something more, then we'll  
19 consider that, based on the feedback we get from our  
20 engagement with stakeholders, individual developers,  
21 and so forth.

22 So, I think that's the last slide.

23 Questions?

24 MEMBER HALNON: Bill, this is Greg.

25 Given that last bullet there, maybe an

1 inclusion on the web page would be a link that emails  
2 a certain person that could take those kind of  
3 comments. I don't know if that's possible in the  
4 public world, but it seems it would be an easy way to  
5 get some feedback on it, rather than just wait for  
6 public meetings or someone to get so frustrated that  
7 they call you. It might be a good way to get some of  
8 that feedback.

9 MR. RECKLEY: All right. We'll put  
10 Arlon's name right on there.

11 MEMBER HALNON: That's what I was  
12 thinking.

13 MR. COSTA: Appreciate that.

14 DR. BLEY: Hey, Bill, it's Dennis Bley.

15 I haven't looked up what's on the public  
16 web about Part 53 lately, but have you thought about  
17 doing anything similar to support Part 53?

18 MR. RECKLEY: We have quite a bit on Part  
19 53. I think this one might flow a little better than  
20 our current Part 53. So, we'll take a look at it.  
21 But we do have a lot of information on Part 53  
22 currently on the public website.

23 DR. BLEY: And I was just thinking about  
24 the roadmap, and we've talked about that before.

25 MR. RECKLEY: Yes.

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1 DR. BLEY: But that might be a place to do  
2 that.

3 MR. RECKLEY: Okay. Thank you.

4 Hey, John, I'm not sure if you had any  
5 additional kind of --

6 MR. SEGALA: Yes, this is John Segala.

7 I mean, I appreciate all the discussions  
8 and the feedback, and I appreciate the opportunity  
9 that we had today to present on all the activities  
10 that we've been working on, and the new web page,  
11 everything.

12 I think I'll let you guys conclude and how  
13 you want to move forward. But I guess I'm just kind  
14 of hearing that some of the things that we're thinking  
15 about as sharing the new web page at our upcoming  
16 stakeholder meeting in March, you know, reaching out  
17 to stakeholders at that meeting and see if there's a  
18 need for something else in terms of source term.

19 It seems like, from what I've heard, that  
20 there seems to be a general flavor that there's a lot  
21 of good information out there. It's just, you know,  
22 there's a lot of information scattered. So, it's hard  
23 for a new developer to look at in more of a  
24 consolidated manner.

25 Our attempt with the website was an

1 attempt to try to pull it all together and make it  
2 easy for people to find. And as Michelle and Bill  
3 said, we're going to continue to refine that web page  
4 and enhance it, and add to it as new information comes  
5 out.

6 So, anyway, again, I appreciate the  
7 discussion and even Joy saying that ACRS providing  
8 feedback on our websites -- we're open; any sort of  
9 feedback for things that we can do better to  
10 communicate externally, you know, that's something  
11 that we're interested in.

12 With that, I think that sort of concludes  
13 the staff remarks.

14 MEMBER REMPE: Again, it was just  
15 individual member comments. This is a work-in-  
16 progress. John, do you think it's appropriate, I  
17 mean, knowing that this is a work-in-progress, do you  
18 think an ACRS letter is really going to help the  
19 staff? You've got our individual comments.

20 But I heard you say at the beginning of  
21 this meeting, "We hear there's going to be a letter."  
22 You guys were not asking for a letter. What's your  
23 thought on this? I mean, we're doing individual, fly-  
24 by-night comments here. Do you want us to be more  
25 measured and come up with a letter? Or do you think

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1       you've got enough?

2                   MR. SEGALA:     Well, I think, from our  
3       perspective, we feel like we have enough for  
4       developers. We feel that the early-moving developers  
5       that were engaging with or developing source term, we  
6       haven't really heard any strong interest from the  
7       developers through our interactions with them in pre-  
8       app, or even in the public meetings, that there's a  
9       need for additional guidance.

10                   And so, we kind of feel like we don't  
11       really need a letter. We tried to listen to the  
12       feedback that you all provided to us at earlier  
13       meetings on EP and other things. And we've tried to  
14       do all sorts of activities, like develop guidance and  
15       develop the MELCOR-SCALE demonstration projects, and  
16       make the videos and information available, and the  
17       contractor reports.

18                   So, I think what we decided sort of at the  
19       end was, you know, that that's where we kind of ended  
20       up with a web page. We thought, well, maybe we do  
21       need to pull all this information together and make it  
22       easy access. And so, that was sort of what we thought  
23       was kind of the piece that was needed to kind of pull  
24       it all together. And it is a work-in-progress, and  
25       we're going to continue to engage with these

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1 developers and applicants, as they submit applications  
2 and do the reviews.

3           You know, we've been developing guidance,  
4 as you can see from this whole presentation, we've  
5 been developing guidance for LWRs since the early  
6 '60s, and we continue to develop guidance for LWRs.  
7 And I imagine that we will continue to do that for  
8 non-light water reactors, as well as SMRs and other  
9 new technologies that come forward.

10           So, I think we're trying to remain  
11 flexible. We're trying to remain agile. We're trying  
12 to optimize and enhance our guidance, as we move on  
13 and as we learn new information.

14           So, I'm not sure if that answered your  
15 question, but that's kind of our view.

16           MEMBER REMPE: It helps to understand.  
17 Again, we do this now with the research folks; that  
18 we're having more frequent meetings, which takes more  
19 staff time to come and talk to us, and individual  
20 members provide comments. And we hear from your  
21 personnel that's more helpful to them than having  
22 formal letters coming out. And I just wondered where  
23 you guys were on it.

24           Personally -- and again, I'm sure Dave  
25 will have us go around -- but I think you guys have

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1 done a great job. You've made good progress. But,  
2 you know, interacting, we have some ideas. I think it  
3 will be interesting to see what the public comments  
4 are at the end of this meeting, as well as what you  
5 learn in the middle of March, when you go back to them  
6 and talk to them.

7 Anyway, that's just my thoughts.

8 CHAIRMAN PETTI: Other members?

9 MEMBER DIMITRIJEVIC: Yes, Dave. Hi.  
10 This is Vesna.

11 I wanted to make one general comment in  
12 the end about something which is very important for  
13 me, sort of my hangup. And that was not addressed so  
14 much in the source data or it was stressed in the  
15 beginning of looking. And that's uncertainty  
16 associated with results.

17 So, we saw some of your results where  
18 there was a play with the parameters in Monte Carlo  
19 and produced some uncertainties. But is just a  
20 thought for an ice break. There is some much  
21 uncertainties in these results which are associated  
22 with models, you know, the radionuclides, barriers,  
23 the facilities, the methodology which is used. So,  
24 uncertainties are so much bigger than these parametric  
25 uncertainties. And maybe there we see results

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1 presented with uncertainties, the brackets.

2 So, I would like, actually, you know, and  
3 this is also my personal opinion, but I think that  
4 everybody agrees the uncertainties is a necessary part  
5 of the game and something starting to be used in  
6 regulation. You know, in my opinion, my personal  
7 opinion, the uncertainties from the PRA Level 1, I  
8 don't know if they necessarily are handled well, but  
9 they are much better and astute, and they always  
10 address and we talk about mean values and 95 and  
11 median and point estimates.

12 However, as we progress through the Level  
13 II and Level III, in my opinion, uncertainties triple.  
14 They become a much bigger and much larger, and we  
15 don't really see that yet, because these uncertainties  
16 are not as well and astute, and they're not looking to  
17 in the level of details necessary.

18 So, I understand that this is just in the  
19 beginning and maybe it's too early to want a lot of  
20 attention to uncertainty, but I definitely think that  
21 that's something we should be very seriously looking  
22 to.

23 All right. That was my comment. You  
24 know, I don't think we really understand the  
25 uncertainties associated with this and we need to

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1 devote much more attention to understand these  
2 methods, models, and methodology associated with it.

3 CHAIRMAN PETTI: Other members?

4 Oh, go ahead, John. No, John, go ahead,  
5 if you want to respond.

6 MR. SEGALA: Yes, I was just going to say  
7 that I think we agree that uncertainties are  
8 important. I think modeling and understanding the  
9 uncertainties and accounting for the uncertainties are  
10 something that's acknowledged in the Licensing  
11 Modernization Project, as well as the non-light water  
12 reactor PRA standard. I think that that is something  
13 that is going to have to be accounted for in the  
14 design margins, and whatnot, as we move forward.

15 So, thank you for the comment.

16 CHAIRMAN PETTI: Anyone else?

17 MEMBER KIRCHNER: Dave, this is Walt.

18 CHAIRMAN PETTI: Yes?

19 MEMBER KIRCHNER: I misquoted the number  
20 of the Draft Reg. Guide earlier. I should have been  
21 saying Draft Guide 1389. Is that the topic of your  
22 Subcommittee's meeting in March? Are we going to  
23 review that document?

24 CHAIRMAN PETTI: Yes, I think so. That's  
25 -- we gave up on 1.138 one -- what it will evolve to,

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1 I believe.

2 MEMBER KIRCHNER: Yes. Well, just a  
3 thought and observation. I want to go away and study  
4 it more closely. But it seems that, echoing an  
5 earlier observation, that the methodology used to  
6 develop an alternate source term for an LWR, where we  
7 have a lot more data, we have a lot more experience,  
8 should, in general, and conceptually, apply for  
9 advanced reactors.

10 And I'll take a look at the Reg. Guide  
11 Draft, but it seems to me the LWR specifics could be  
12 in appendices and the methodology or the recommended  
13 approach, much like was covered earlier from the INL  
14 report, could find its way into at least an executive  
15 summary or an introduction to the Reg. Guide. I'll go  
16 and take a look, and I'll give you some comments  
17 perhaps that could be used for that interaction.

18 CHAIRMAN PETTI: Okay.

19 MEMBER KIRCHNER: And the other thing,  
20 Vesna, I agree with you. Having had some experience  
21 in this modeling and simulation business, once you  
22 lose the fuel geometry, then the uncertainties in the  
23 models are much greater than the parametric  
24 uncertainties. You get into stochastic and random  
25 kind of results.

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1           So, once you're in that territory, so to  
2       speak, once the uncertainties expand, and bounding  
3       analyses are probably a wise choice to try and get  
4       some sense of the consequences -- this is true of even  
5       thermal hydraulics for existing plants. Once you go  
6       into two-phase flow and the phases separate, then the  
7       uncertainty in the model results increases  
8       substantially. And that has nothing to do with  
9       parametric uncertainties. It's just the uncertainty  
10      in terms of our ability to model what is happening  
11      physically. Fuel fragmentation is a good example of  
12      that.

13           MEMBER DIMITRIJEVIC: Yes. You know,  
14      Walt, I was thinking, how can this be addressed? I  
15      was thinking that, maybe since this is international,  
16      there are other programs which do similar maybe  
17      computing results, or in the case of Fukushima, we  
18      have real data to compare with the results of the  
19      models.

20           I don't know how to estimate that. It's  
21      not an easy thing to do. I mean, even I don't think  
22      the bounding things always is a way to address  
23      something, especially as these become a more important  
24      part in the regulation. I mean, how are we going to  
25      do the risk-informed applications based on, you know,

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1 source term, and the dose and dispersions, and things  
2 like that?

3 So, I'm not sure how to address that, but  
4 it's exactly what I thought, what you said. Thanks.

5 MEMBER KIRCHNER: And, Dave, I would just  
6 like to thank the staff for this great presentation  
7 today. There's a lot of information here.

8 CHAIRMAN PETTI: Yes.

9 MEMBER KIRCHNER: So, I look forward to  
10 browsing or grazing on their website.

11 CHAIRMAN PETTI: Yes.

12 MEMBER KIRCHNER: Thank you.

13 CHAIRMAN PETTI: You took my comment. I  
14 really want to reiterate what Walt just said. This  
15 presentation, you know, the depth and breadth is  
16 fairly unique from what we've heard, I think, in other  
17 presentations. It required a lot of different folks  
18 to come together, and I really appreciate it, given my  
19 background, of course, in this area. I've thought  
20 about these things as well, and I really do appreciate  
21 the slides.

22 That said, we should discuss, among  
23 ourselves I guess, the need for a letter. My letter  
24 was going to be --

25 MEMBER REMPE: Can we do public comments

1 first, though?

2 CHAIRMAN PETTI: You want comments first?

3 Okay. Sure.

4 MEMBER REMPE: Yes. Because I think that  
5 I'd like to hear what the public has to say.

6 CHAIRMAN PETTI: Okay.

7 MEMBER REMPE: Is that okay?

8 MR. SNODDERLY: Yes, that's the way we had  
9 -- we had it set up that way, before member  
10 discussion.

11 CHAIRMAN PETTI: Okay. So, for members of  
12 the public, \*6 to unmute yourself. Give us your name  
13 and your comment.

14 MR. SNODDERLY: I'm sorry. Dave, we did  
15 something different today. So, the members of the  
16 public are on. So, you just have to unmute yourself.

17 CHAIRMAN PETTI: Oh.

18 MR. SNODDERLY: Yes. Yes.

19 CHAIRMAN PETTI: Okay.

20 MR. SNODDERLY: I don't know if Ms. Fields  
21 is still there, but I saw some members of the public  
22 in.

23 MS. FIELDS: Hi. Yes, this is Ms. Fields.  
24 Can you hear me?

25 CHAIRMAN PETTI: Yes, we can.

1 MS. FIELDS: Oh, okay. Because of another  
2 appointment, I didn't hear the whole presentation, but  
3 what I did hear and see was very, very informative.  
4 So, I just have a few comments.

5 As you know, there's still no permanent  
6 repository for irradiated nuclear fuel. And to me, it  
7 does not make sense for the United States to keep  
8 making used or spent nuclear fuel when there is no  
9 permanent repository and none anticipated in the near  
10 future, and I believe none in my lifetime.

11 Also, the NRC should not use the term  
12 "advanced" in these various rulemakings and documents.  
13 "Advanced" is a public relations term. It doesn't  
14 have a regulatory or a statutory or a technical basis.

15 The NRC has already dropped the term  
16 "advanced" in the Part 53 rulemaking. The rulemaking  
17 is not for advanced nuclear reactors; it's just for  
18 commercial nuclear reactors. And the NRC has taken  
19 the definition of "advanced" out of that rulemaking.

20 And there are some various reasons for  
21 that -- in part, I imagine, because the NEIMA  
22 definition of "advanced" just does not make sense.  
23 There's just no real basis for that.

24 Also, I'm still concerned that the NRC has  
25 not been able to accession properly the documents

1 related to these pre-application and application  
2 submittals to the NRC. Many of the documents have  
3 been accessioned to ADAMS, but they have no docket  
4 number, or the docket number should be on an  
5 application document for an application docket, but  
6 it's still on the pre-application docket. The NRC has  
7 been paying attention to this and made some changes,  
8 but there's still a big issue.

9 I also think that the NRC and the ACRS  
10 should pay more attention to the needs of the public.  
11 Obviously, this whole process and the discussion today  
12 was really to satisfy the needs of the industry, as  
13 they move forward with these new reactor designs and  
14 proposals.

15 But there's little information available  
16 to the public that would give them an idea of what  
17 exactly they should look for, if there is an  
18 application to site one of these new reactor designs  
19 in their community. There's just really not much  
20 there that would guide the public, as they review  
21 these applications and they consider whether this is  
22 appropriate for their community.

23 So, I definitely think that the NRC should  
24 think -- well, maybe even have a public meeting for  
25 public input on what exactly the public, information

1 the public needs, not so that the industry can site  
2 something in their community, but so the community has  
3 the tools it has to actually evaluate what's going on,  
4 like the amount of water that might be needed. Or is  
5 it really appropriate to reduce the EPZ and not have  
6 it at the traditional 10 miles?

7 So, it's just something for you all to  
8 think about in the future. Thank you.

9 CHAIRMAN PETTI: Thank you.

10 I believe Kati Austgen is next.

11 MS. AUSTGEN: Yes, thank you. Kati  
12 Austgen with the Nuclear Energy Institute.

13 First, I just wanted to thank the ACRS and  
14 the NRC staff for trying something new and letting us  
15 participate in the Teams webcast this time around. It  
16 was much easier to follow along on the slides and  
17 understand which members and which staff were making  
18 which comments. So, we very much appreciate that.

19 And then, I agree with the ACRS that this  
20 was a very informative presentation, and I look  
21 forward to sharing the information about the new  
22 website with our members, and continuing our  
23 conversations with the staff on guidance in this area  
24 and how we can all make sure we're on the same page  
25 about what's required.

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1 Thank you.

2 CHAIRMAN PETTI: Thank you.

3 Jan Boudart?

4 MS. BOUDART: Jan Boudart, and I'm a Board  
5 member of the Nuclear Energy Information Service in  
6 Chicago.

7 And apropos to what Sarah Fields said,  
8 there is being considered an experimental reactor at  
9 the University of Illinois in Champaign-Urbana. And  
10 the question that NEIS is asking is, how do people  
11 feel about sending their kids to that University if  
12 there's going to be a fission project on the  
13 University campus? And how is the NRC, or whatever  
14 appropriate agency -- maybe it's the DOE -- how are  
15 they going to present this to the public and to the  
16 people who are considering sending their kids there?  
17 And is there going to be a public hearing? And will  
18 the ramifications of having this on the campus be  
19 explored, et cetera, et cetera?

20 But I think this is very appropriate to  
21 what Sarah said, because the people who are going to  
22 be affected by these small modular nuclear reactors  
23 and the experimental reactors, like the one that's  
24 being considered with the TVA in Tennessee, the people  
25 in the neighborhood and the people who are going to be

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1 in the area need to be consulted as to whether they  
2 think it's appropriate to have these things near them.

3 That's the end of my comment.

4 CHAIRMAN PETTI: Thank you.

5 Any other comments from members of the  
6 public?

7 MEMBER REMPE: Dave, am I -- oh, I'm  
8 sorry, I interrupted. Yes.

9 CHAIRMAN PETTI: Please go ahead.

10 MR. STEIN: This is Adam Stein from the  
11 Breakthrough Institute. I have a comment and a  
12 clarifying question, if you are willing to entertain  
13 the question.

14 To what extent does the staff expect to  
15 require a MELCOR model to be developed for specific  
16 advanced reactors? Because, as was presented today,  
17 the models that have been created so far are not  
18 direct replicas of the various developers; they are  
19 based on other similar technologies, or similar  
20 designs, I should say. But, as the staff mentioned  
21 today, a significant amount of time and effort was put  
22 into developing those models. So, to what extent is  
23 the staff expecting to require a MELCOR model that is  
24 similar to an applicant's design to be developed? And  
25 if not, how closely resembling results are they

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1 expecting between the existing MELCOR models or the  
2 ones that are being developed and an applicant's  
3 submission of results?

4 CHAIRMAN PETTI: Thank you. We don't  
5 respond to comments from the public in our meeting.  
6 But you can send an email to Mike Snodderly, the  
7 Designated Federal Official. Mike can tell you his  
8 email, and take it from there.

9 MR. SNODDERLY: Yes, Adam, I think if you  
10 want to send me a written comment, then we can add  
11 that to the record. But the other option is you just  
12 turn your question into a comment, and that also is  
13 being transcribed.

14 So, what I heard could be your comment is  
15 that MELCOR models, as they've been developed, are  
16 valuable and should be required or part of any review  
17 of a confirmatory analysis of a review of advanced  
18 reactors.

19 But you just need to form it in that kind  
20 of a phrasing. And if you want more time to do that,  
21 you can do that, and then, send me an email to  
22 michael.snodderly -- S-N-O-D-D-E-R-L-Y -- @nrc.gov.

23 MR. STEIN: Thank you.

24 I would also like to comment that it is  
25 important to consider that, when you are looking at

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1 the differences between a MELCOR model and a,  
2 potentially, applicant-developed model, that it is  
3 important to consider and disposition why and how they  
4 may differ in results. Because, as was discussed  
5 today, there are still various considerations or  
6 physical parameters that are not in both models  
7 identically, and there will still be dispositions.

8 Thank you for your time.

9 CHAIRMAN PETTI: Any other public  
10 comments?

11 I see a hand. Jan?

12 MS. BOUDART: It's me again.

13 I wanted to ascertain whether Mr.  
14 Snodderly's email, does it say @acrs.nrc.gov or is it  
15 just nrc.gov?

16 MR. SNODDERLY: Yes, Jan, it's @nrc.gov.

17 MS. BOUDART: Oh, good. Oh, thank you.  
18 That's what I thought. Thanks.

19 MR. SNODDERLY: Thanks.

20 CHAIRMAN PETTI: Okay.

21 MEMBER REMPE: Mike, is your hand up?

22 MR. SNODDERLY: Yes. If you don't mind,  
23 Dave, I just wanted to remind the Committee, as you  
24 start your member discussion, I went back to your  
25 presentation on October 8th to the Commission. And in

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1 that, just to remind everybody, in your slides that  
2 you presented -- let's see -- the second-to-last slide  
3 that you presented, you said that, "Numerous recent  
4 and upcoming source-term-related activities. A  
5 roadmap showing how all the pieces fit together would  
6 be worthwhile"; and that "Many different pieces are  
7 coming together, and that the ACRS plans an integrated  
8 review later this year."

9 So, I just wanted to remind the Committee  
10 that, well, you didn't make a commitment, but you told  
11 them that you were going to have this interaction.  
12 You've had it. It doesn't require a letter, of  
13 course. But I think the staff has done something to  
14 provide somewhat of a roadmap, and you may want to  
15 comment on how good or bad it is. But you, of course,  
16 can say nothing. But I just thought that might be a  
17 good kickoff point, and now, I'll be quiet.

18 MEMBER REMPE: Mike, also, while you're on  
19 the line, just so I understand the new process,  
20 everyone can just -- does anyone need to press \*6 now?  
21 Because there are a lot of phone lines out there.

22 MR. SNODDERLY: No, they're on just like  
23 you and I, and it's mute and unmute.

24 MEMBER REMPE: Okay.

25 MR. SNODDERLY: Now I can mute folks, any

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1 participate, but --

2 MEMBER REMPE: Right, but the phone line  
3 folks --

4 MR. SNODDERLY: Yes.

5 MEMBER REMPE: I understand the folks that  
6 are linked in. Like Sarah has a real link.

7 MR. SNODDERLY: Yes.

8 MEMBER REMPE: But the phone line folks  
9 can it themselves? Okay. Just wanted to make sure we  
10 weren't leaving anyone out. Thank you.

11 MR. SNODDERLY: That's my understanding,  
12 but Tom Daschle (phonetic) -- you know, correct me if  
13 I'm wrong -- but I think that --

14 MR. BURKHART: This is Larry.

15 One of the commenters, our last commenter  
16 was on the phone. So, I think it's working.

17 MR. SNODDERLY: Yes. Thank you, Larry.

18 MEMBER REMPE: Great. Thank you.

19 CHAIRMAN PETTI: Okay. So, Members, let's  
20 talk a little bit about the need for a letter; what  
21 your thoughts are.

22 I have, as I always have, put together  
23 what I call the guts of a letter. It's just a bunch  
24 of points and starting points.

25 While I don't think the staff needs a

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1 letter, my opinion is that, in light of everything  
2 that's going on right now in the area with non-LWRs,  
3 that a pointer letter, which is what the letter would  
4 be, pointing out some of the important things that  
5 they've done, might be quite helpful. And I'm  
6 certainly willing to write the letter, given the  
7 importance of the topic, as we heard from (audio  
8 interference) for the NRC.

9 So, comments, Members? And consultants?  
10 I'd be interested in hearing their perspectives, too.

11 MEMBER BROWN: It's Charlie.

12 Based on my listening, this was kind of a  
13 potpourri-type presentation. I head a lot of comments  
14 from us, but unless we have some specific things that  
15 should go on in this general, multiple opportunities  
16 for people, I don't see where a letter would really  
17 add a whole lot, other than just saying, "Continue."  
18 My opinion is I wouldn't write a letter based on this  
19 meeting.

20 MEMBER KIRCHNER: Dave, this is Walt.

21 I guess I would concur with Charlie. I  
22 would wait until you heard, we hear about DG-1389 and  
23 its status. Because what I'm thinking is, although  
24 this set of presentations today primarily is  
25 addressing, quote-unquote, "advanced" reactors and

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1 applicants coming in, it's likely that they're going  
2 to go in the near term, while 10 CFR 53 rulemaking is  
3 in progress, that they are going to come in under  
4 10 CFR 50 or 52. And if they do so, then they can  
5 either do exceptions to guidance and the regulations  
6 or they'll follow the spirit of the guidance that's  
7 out there. And that's why I'm very interested in what  
8 DG-1389 contains and whether it's a broader template  
9 for other advanced reactor applications.

10 Thank you.

11 CHAIRMAN PETTI: Go ahead, Ron.

12 MEMBER BALLINGER: This is Ron.

13 I think that, unless we want to reinforce  
14 something that we've already communicated on at least  
15 two occasions -- the first with the Commission  
16 presentation and the second, at least the second today  
17 -- I think that we should probably -- I think the  
18 staff has got the message. And so, I think I'm in  
19 agreement with Walt -- waiting until we have something  
20 which we can actually review in detail.

21 Thank you.

22 CHAIRMAN PETTI: My only concern, having  
23 read that Draft Guide, is that it's extremely light-  
24 water-reactor-specific and we're going to focus on  
25 that. So, I don't see us getting into non-LWRs in

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1       that review because that's not where that is.

2                   MEMBER BALLINGER: That would not keep us  
3       from making a comment, does it?

4                   MEMBER KIRCHNER: That's what I was  
5       thinking, Dave. You can build on that to say whatever  
6       we feel -- more, less, or keep on going, or whatever  
7       -- for the non-LWR reactors or something new,  
8       completely new and different is needed. That would be  
9       a good juncture to make that kind of observation.

10                  MEMBER HALNON: Dave, this is Greg.

11                  CHAIRMAN PETTI: Dennis? Oh, go ahead,  
12       Greg.

13                  MEMBER HALNON: No, well, Dennis had his  
14       hand up. I couldn't see that.

15                  Go ahead, Dennis. I'll come after you.

16                  DR. BLEY: Okay. Yes, Dave, well, first,  
17       I'd say I found today very useful, and I've played  
18       around on that website already this afternoon. I  
19       think that's going to be especially nice.

20                  Now we sometimes get tricked in things we  
21       read, as Joy said, in the popular press and in papers  
22       people send to us to look at. I'm not sure how much  
23       of what I read is politics and how much is real, where  
24       people are technically.

25                  There's a lot of things passing around

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1       that kind of say, "Gee, the NRC is stuck in the Dark  
2       Ages and they ought to just say, almost, you don't  
3       need to do anything with these new reactors. Just let  
4       them operate."

5               I think what we saw today ties a lot of  
6       very important information together. And if I had a  
7       new design and it had extremely low consequences and  
8       likelihood of damage, I think I could do something not  
9       terribly elaborate to prove that point within those  
10      constructs.

11             So, I think what they've put together  
12      today is really useful, integrating all these pieces.  
13      I would have agreed with Walt and Ron that we ought to  
14      wait for the meeting next month, and there's when it  
15      would be most useful for you folks to write a letter.  
16      But I didn't read the draft and you have. And if they  
17      aren't closely coupled, then, maybe it makes sense  
18      separately.

19             I mean, this isn't at a point where this  
20      rates a 10-page letter. A one-page letter might be  
21      appropriate to acknowledge what's being done and its  
22      value. So, I think we shouldn't let this pass without  
23      -- I don't think you should let it pass without some  
24      comment.

25             CHAIRMAN PETTI:     So, yes, my talking

1 points are all of 80 lines. You know, it's not a big  
2 letter. But we're viewed as sort of an independent  
3 look at things, and there's these questions out there  
4 by all sorts of folks about NRC, as you say, being in  
5 the Dark Ages and they don't really understand these  
6 non-LWRs. And I think nothing could be further from  
7 the truth. They've done an awful lot of work to get  
8 themselves ready to accept an application. And to  
9 recognize that, I thought, personally, was of value.

10 MEMBER HALNON: This is Greg.

11 I look at it from a perspective of the  
12 Commissioner sitting in his or her office, and  
13 somebody walking in wanting a drop-in meeting and have  
14 discussion about some iterations they're going through  
15 with reviews, and whatnot.

16 And I guess there's a couple of nuggets  
17 that came out today. And notwithstanding the one  
18 about the burden of developing the methodologies is on  
19 the developer and expect it to be complicated, and  
20 those sorts of things, there's a couple of messages  
21 that I think the Commissioners need to hear, so that  
22 they're armed with some information.

23 And I'm not sure if it gets to them or  
24 not, but the fact that there is an integrated website  
25 and it's in process; it's in really pretty good shape,

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1 but we're taking comments on it; the fact that there's  
2 a couple of nuggets in there that we heard today that  
3 we're either expecting or anticipating or have learned  
4 -- I think those type of things, going to the  
5 transcripts, might be useful.

6 And I'm not talking about a long letter,  
7 but a couple of sentences here and there that  
8 acknowledge that we're in an iterative-plus process  
9 that's going to be getting better every time we have  
10 interaction with a new technology and we all learn  
11 more about it. I think those types of things are  
12 pretty important.

13 So, I'm kind of in with you and Dennis,  
14 Dave, that there could be a very short letter that has  
15 some of the things that we heard, some of the things  
16 that we anticipate, and wouldn't negate having another  
17 letter down the road for a specific Reg. Guide, but,  
18 certainly, from the topic itself, could use some, I  
19 guess, collation into a concise letter.

20 CHAIRMAN PETTI: Steve? I see your hand  
21 up.

22 DR. SCHULTZ: Yes. For the Committee, I  
23 do appreciate the discussion here about a letter, but  
24 I don't think it's letter or no letter. Because I  
25 think the Committee has made these comments to the

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1 Commission in their last meeting; that this is a topic  
2 that is extremely important to the advancement of the  
3 new reactors; and also, to the work that is being done  
4 associated with their development and application.

5 And I think a followup to just those  
6 comments to the Commission would be very useful,  
7 especially in light of the excellent presentations  
8 that were made today. A very well-integrated  
9 presentation and very useful to the Committee and,  
10 also, in the public forum.

11 My other comment would be that not only is  
12 this of interest in the United States, but the  
13 international community. The regulatory agencies in  
14 those areas are also working on the same thing. And  
15 for the ACRS to not come out with a continuing  
16 statement about what is ongoing and what is  
17 appreciated and associated with this work -- I think  
18 that's important for the Committee to do. Whether  
19 it's one letter or two letters, it's up to the  
20 Committee. I think that an individual letter on this  
21 topic would be not the last letter the Committee  
22 writes, but very important to continue to endorse.

23 CHAIRMAN PETTI: Joy?

24 MEMBER REMPE: So, I'm torn. Clearly, we  
25 need to decide because the month is coming up and the

1 staff needs to know if they're going to be coming back  
2 to us.

3 But, listening today to the public  
4 comments, it seemed to me, and what the staff said,  
5 that they're going to have an upcoming stakeholder  
6 meeting in March, where they'll maybe learn more about  
7 whether more guidance is needed -- I mean, clearly,  
8 they've made a lot of good progress and they've got a  
9 good -- you know, I really do like the reference plant  
10 evaluations. I'd like to see a little more coming out  
11 of it, which I think the staff was receptive to.

12 I think that some of our comments about  
13 the need for guidance is important, but I'm not sure  
14 whether we should be telling the staff to do that,  
15 when I really appreciate that there's so many  
16 different designs out there, that maybe the staff is  
17 at the right -- that what they're doing is correct.  
18 Because why waste resources on doing guidance for so  
19 many different designs and flexibilities?

20 And one of the public comments, to me,  
21 indicated that the person didn't understand the  
22 staff's broad perspective. I don't think the staff is  
23 trying to say, by any means, that a MELCOR model is  
24 needed in what we've heard today. So, I think  
25 communication with the stakeholders is very important.

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1 But that was my impression of it and all,  
2 but I don't know if the timing is right. I definitely  
3 would put "Interim Letter on Staff Progress," or  
4 something like that, as the title of whatever is  
5 issued. But I'm open to whatever the Committee  
6 decides, but I don't know if this is the right time or  
7 not.

8 I'm not sure that's a very helpful  
9 comment.

10 CHAIRMAN PETTI: My biggest concern is,  
11 you know, there's probably never a good time. And we  
12 know the bow wave of what's coming in front of us,  
13 starting in April. There just seemed to be a window  
14 of opportunity here, given what's on the plate for  
15 March.

16 MEMBER REMPE: But what will we say other  
17 than, "Good job."? And we can do that in the minutes  
18 of the meeting summary, when you come back and do  
19 something like what Jose often does. I don't know; I  
20 just am wondering if we're making them come back to  
21 present to us and things like that. So, anyway, I'm  
22 just kind of thinking about it.

23 CHAIRMAN PETTI: Anybody else? Matt?  
24 Jose?

25 MEMBER MARCH-LEUBA: Yes, Jose. Jose has

1 his hand raised because Joy used my name in vain, and  
2 she took my argument out of it.

3 We have an intermediate method of dealing  
4 with this. First, let me say that I'm ambivalent on  
5 the letter. On the one side, the person that knows  
6 and cares more about this is Dave, and he thinks we  
7 should have a letter. And that makes me want to  
8 support it. On the other side, I see all the other  
9 arguments from the people that maybe we should wait  
10 for a real Reg Guide.

11 But we have a middle ground, which is  
12 writing a couple of paragraphs to be included in the  
13 summary of this meeting. And they get presented to  
14 P&P, and they become the property of ACRS. It's not  
15 a subcommittee any longer. I mean, once P&P approves  
16 those two or three paragraphs, it's half a letter, and  
17 those paragraphs can be submitted by email to  
18 whomever.

19 So, if we are voting, I will vote for the  
20 halfway and go for the summary.

21 MEMBER REMPE: And that avoids the staff  
22 having to come back and do another presentation.

23 CHAIRMAN PETTI: Matt, you've been silent.

24 MEMBER SUNSERI: I've been listening,  
25 Dave. You know, I think it was a very important

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1 topic. We've been asking for how these various source  
2 term activities integrate each other towards the  
3 topic. And so, I'd be in favor of a letter.

4 MEMBER BIER: This is Vicki.

5 It's kind of out of my area. I feel sort  
6 of similar, in between Charlie and Dennis; that, you  
7 know, on the one hand, there was nothing burning today  
8 that inspired me that we have to write a letter, but  
9 I could certainly go with a short letter that just  
10 said, "We support the progress being made," et cetera.

11 CHAIRMAN PETTI: Okay.

12 MEMBER DIMITRIJEVIC: Dave?

13 CHAIRMAN PETTI: Yes?

14 MEMBER DIMITRIJEVIC: You know, I believe  
15 in Jose's philosophy. You know most about that and  
16 you put the letter together, and you think it should  
17 be written. So, I support that.

18 MEMBER SUNSERI: Dave, this is Matt. One  
19 more thing.

20 I think that we have precedents, also, for  
21 not having to have staff come back and make another  
22 presentation to the Committee. I think maybe almost  
23 everybody from the Committee is here, and that you  
24 could make some kind of presentation, a short one, at  
25 the full Committee meeting, and we could write the

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1 letter from that. I think we've done that before on  
2 a lot of the license reviews that we did for design  
3 certifications.

4 MEMBER REMPE: We did it for MELLA+,  
5 actually, and that would make it easier for the staff.  
6 I don't know.

7 CHAIRMAN PETTI: Yes. I certainly don't  
8 -- I mean, we would need a very, very, 100,000-foot  
9 level presentation, because this here, I'm thinking I  
10 would have thought no more than five slides sort of  
11 thing. You know, what were the things you touched on?  
12 But if we can even get rid of that, maybe, then, I  
13 could put a draft together, and then, when we see it,  
14 that might even inform us better. To bring it to the  
15 full Committee, and then, people make a decision? Is  
16 that allowable?

17 MR. SNODDERLY: Yes, I believe that's  
18 consistent with the Bylaws. The Subcommittee needs to  
19 make a recommendation to the full Committee on whether  
20 to write a letter or not. This would also give some  
21 more time to digest it and think about it.

22 But, Dave, you would, when we come to this  
23 item on the agenda, you would make, as a member, a  
24 recommendation to the Committee, and then, the  
25 Committee would vote whether to take it up further and

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1 write the letter or not, or to do some hybrid thing.

2 And it's already been discussed or the  
3 precedent has been set. The staff could be there or  
4 not be there. But, of course, we would like them  
5 there to support letter writing as usual, anyway. But  
6 I don't even know what, if any, presentation would be  
7 needed, unless you requested it, what you think you  
8 need to support the letter.

9 CHAIRMAN PETTI: All right. I don't think  
10 we need them to be there to make a presentation.

11 But let me try to write --

12 MEMBER REMPE: Let me interrupt for a  
13 minute, and I think Larry is going to have to weigh-in  
14 because the agenda is already published in The Federal  
15 Register.

16 If we were to do this in P&P, that puts  
17 this on Friday morning, just to be thinking  
18 processwise here and the mechanics of it. I don't  
19 think we can have -- or maybe we can; I'm asking  
20 Larry. Is it acceptable for Dave to give a  
21 presentation during that allocated time for this  
22 topic, Larry, instead of the staff, and go ahead with,  
23 if the Committee at that point decides to do a letter,  
24 is it acceptable to go ahead and go right into letter  
25 writing?

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1 MR. BURKHART: I think it's perfectly  
2 acceptable for the Chair of the Subcommittee to orient  
3 the full Committee on the topic, and then, move into  
4 report writing, if that is decided.

5 MEMBER REMPE: Okay. I just wanted to  
6 make sure, because this is a little different than  
7 what happened with the MELLA+ thing. But that's  
8 great. I just wanted to make sure we weren't going to  
9 go forward with something that wasn't allowed. Thank  
10 you.

11 CHAIRMAN PETTI: The other thing is what  
12 Mike raised. You know, we did talk about this with  
13 the Commissioners, and it would be nice to kind of  
14 close the loop on that. Again, a shorter letter, that  
15 both sides recognize this issue; thought it was  
16 timely, and we did.

17 So, it's on the agenda. I will,  
18 basically, I guess, during that slot, talk about the  
19 major points in the letter. And we can have a  
20 discussion there, and then, make a decision, as the  
21 full Committee, whether we want a document for the  
22 record or, in a summary sense, in the meeting summary.  
23 Is that reasonable?

24 MEMBER SUNSERI: Sounds good to me.

25 MEMBER DIMITRIJEVIC: Absolutely.

1 Absolutely.

2 CHAIRMAN PETTI: Okay. I will work on  
3 this next week, and I'll actually try to get some of  
4 the stuff to folks before the meeting, so they're not  
5 seeing for the first time at full Committee.

6 Okay. Then, with that, I guess if there  
7 are no other comments, we're ready to adjourn the  
8 meeting.

9 And I thank everyone and, again, thank the  
10 staff. This was a Herculean effort. We really do  
11 appreciate it.

12 So, everybody have a good evening. Thank  
13 you.

14 (Whereupon, at 4:28 p.m., the meeting was  
15 adjourned.)

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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

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## 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

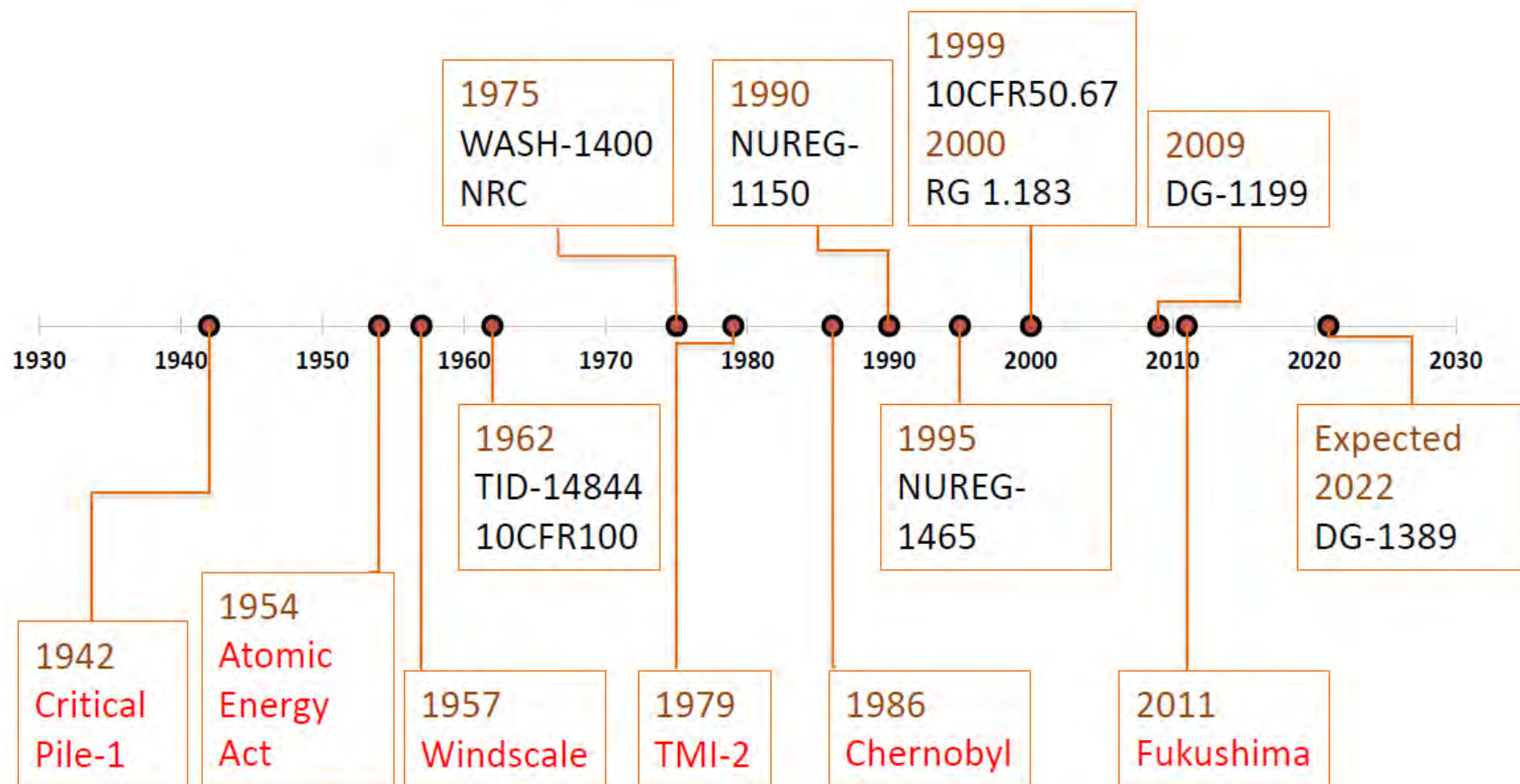
- Determining source terms is a critical component in the NRC's licensing process
- NRC team presenting today:
  - Mark Blumberg – NRR/DRA
  - Michelle Hart – NRR/DANU
  - Jason Schaperow – NRR/DANU
  - Bill Reckley – NRR/DANU
  - Tim Drzewiecki – NRR/DANU
  - Hossein Esmaili – RES/DSA

# 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 efforts 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 (Csl) – 95 Organic 0.15		Elem. – 91 Aerosol –5 Organic 4



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# 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<sup>1</sup>
- On going research efforts is underway to update the SAND2011-0128 accident source term to accommodate higher burnup and increased enrichments for LOCA releases.

<sup>1</sup> 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)”

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# 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

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## 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

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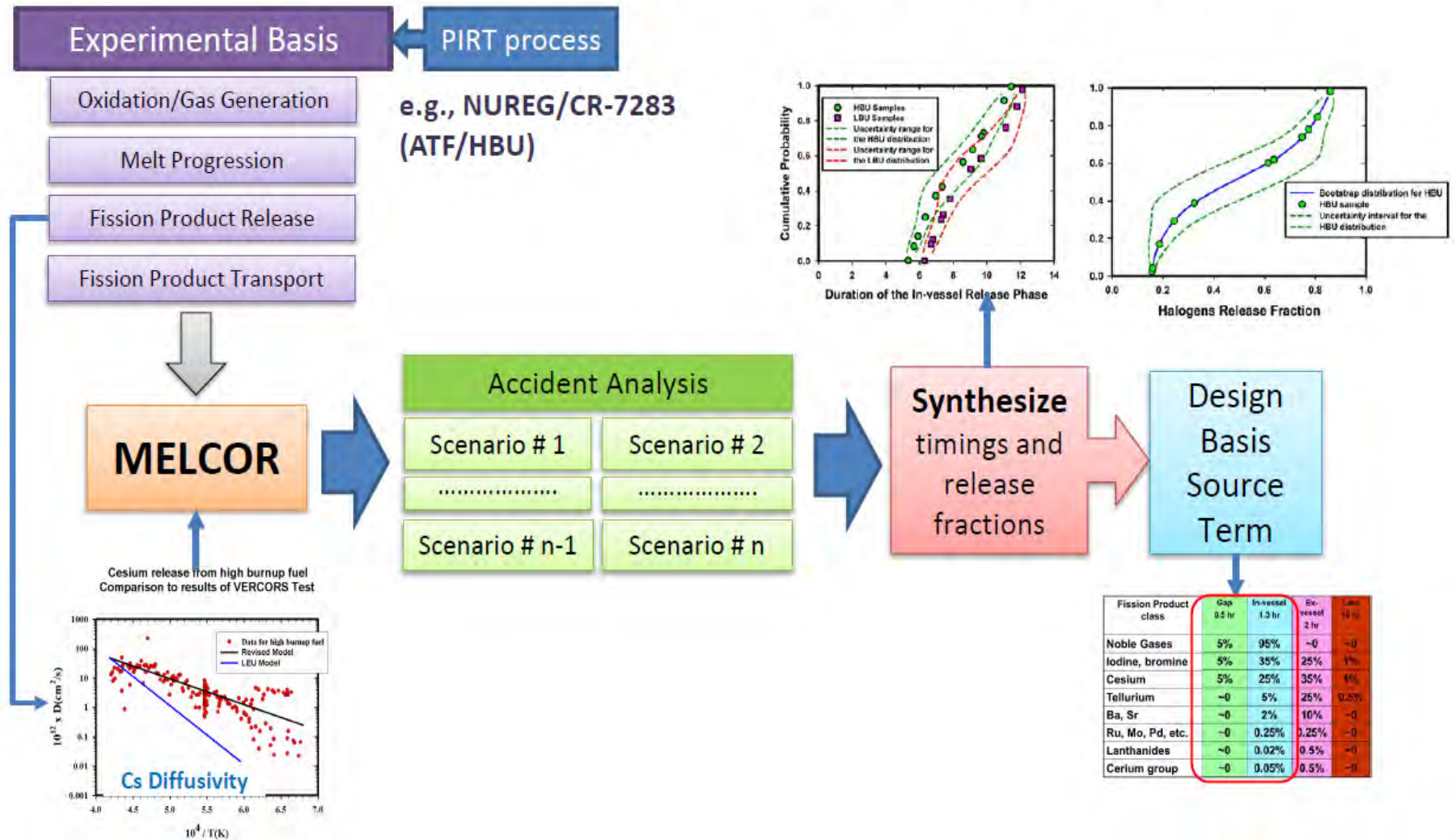
## Outline

- 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

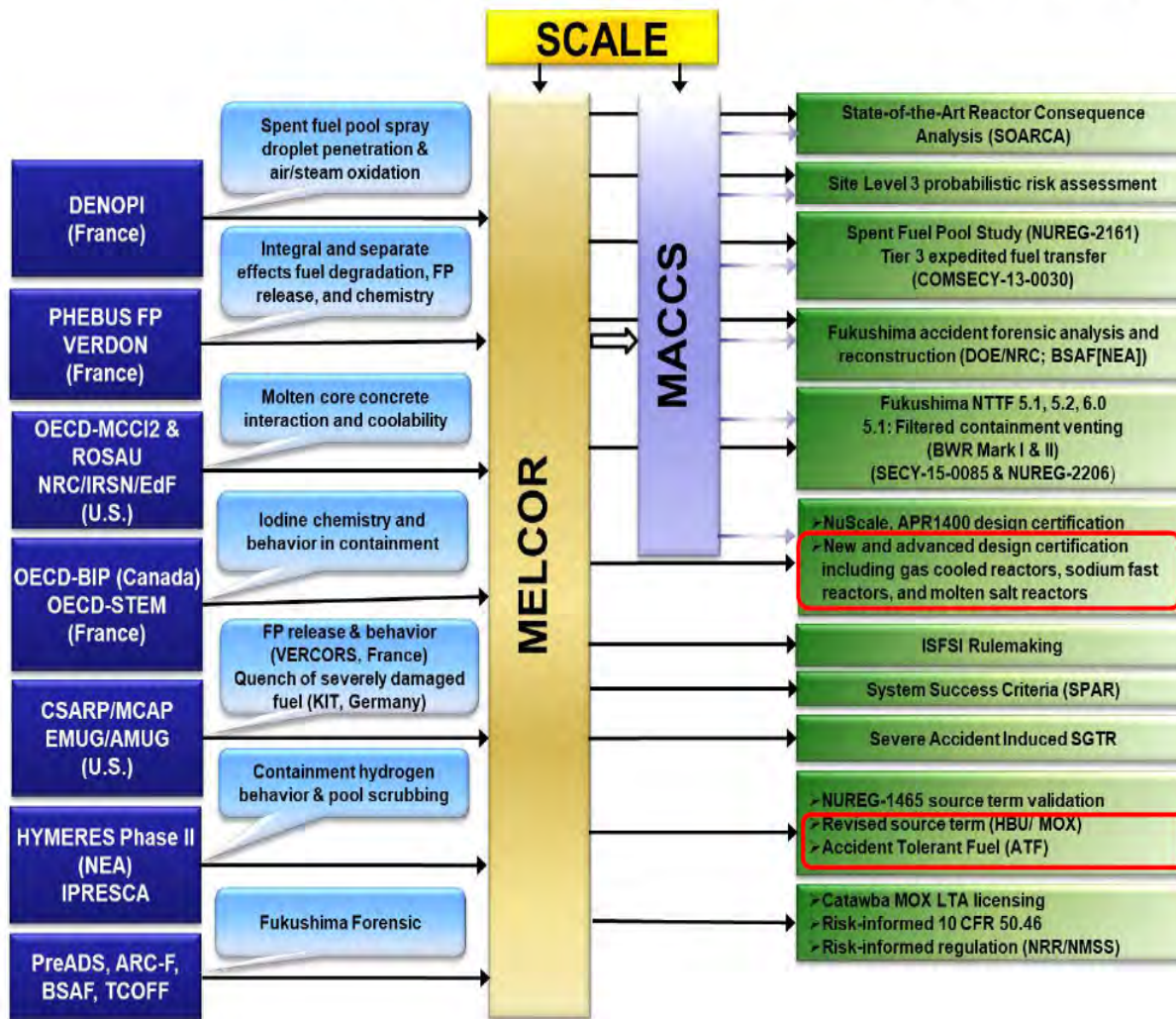
# Introduction



# Source Term Development Process



# 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?

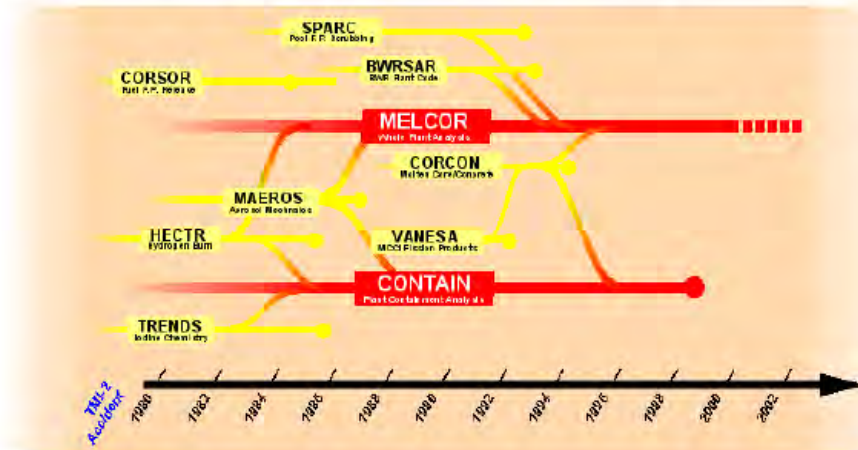
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

# MELCOR Overview



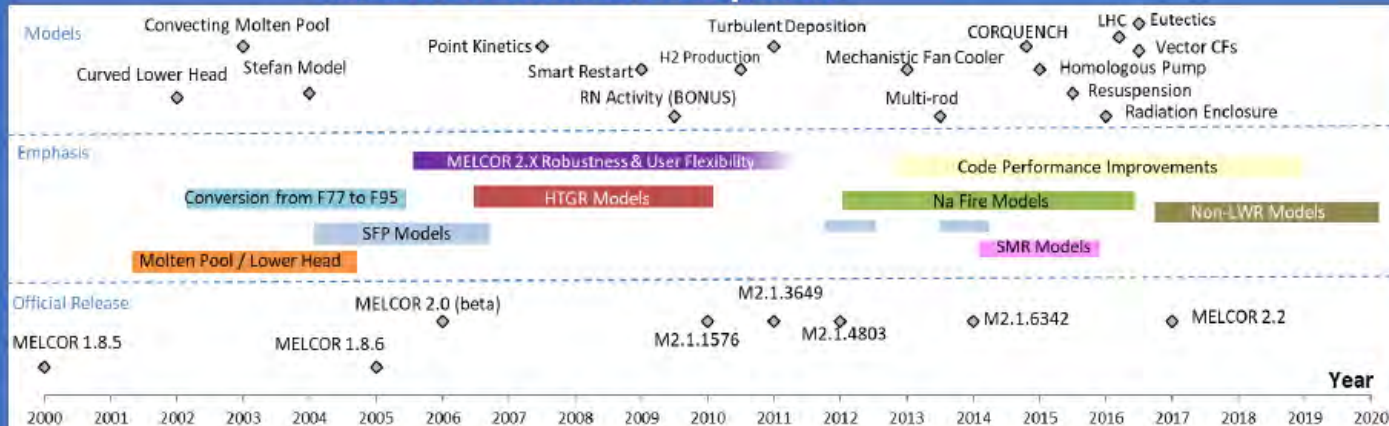
# MELCOR History

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

## MELCOR Code Development



# MELCOR Development

## Fully integrated, engineering-level code

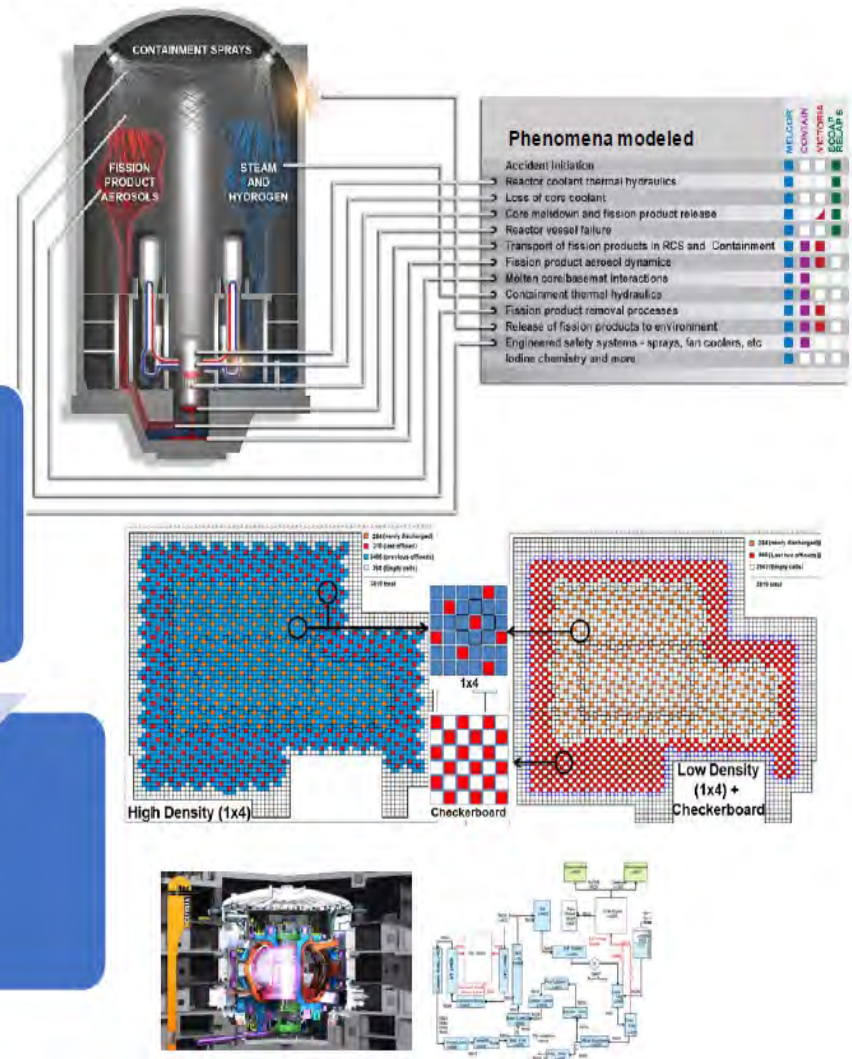
- Thermal-hydraulic response of reactor coolant system, reactor cavity, reactor enclosures, and auxiliary buildings
- Core heat-up, degradation and relocation
- Core-concrete interaction
- Flammable gas production, transport and combustion
- Fission product release and transport behavior

## 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

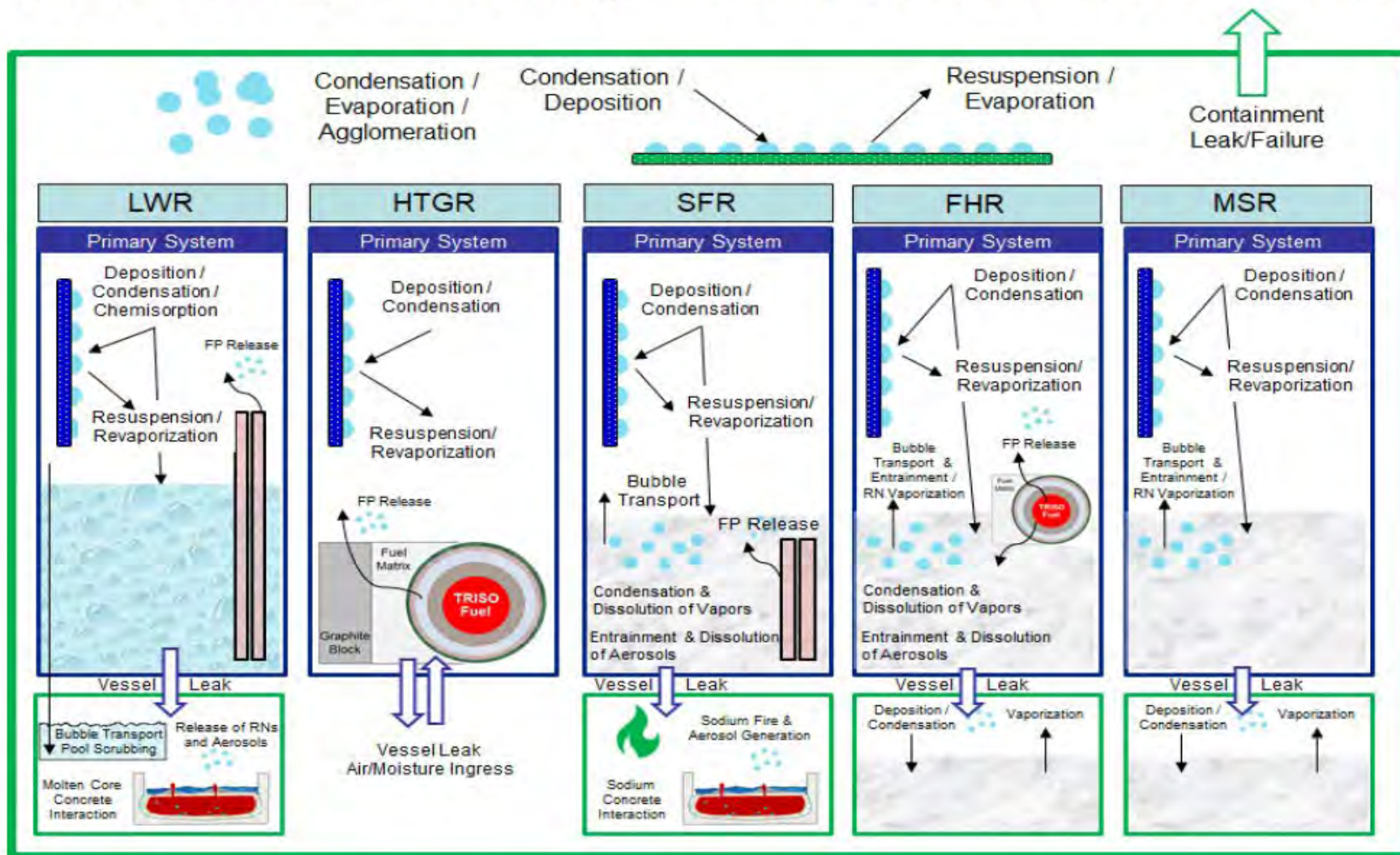
## Traditional application

- Models constructed by user from basic components (control volumes, flow paths and heat structures)
- Demonstrated adaptability to new reactor designs – HPR, HTGR, SMR, MSR





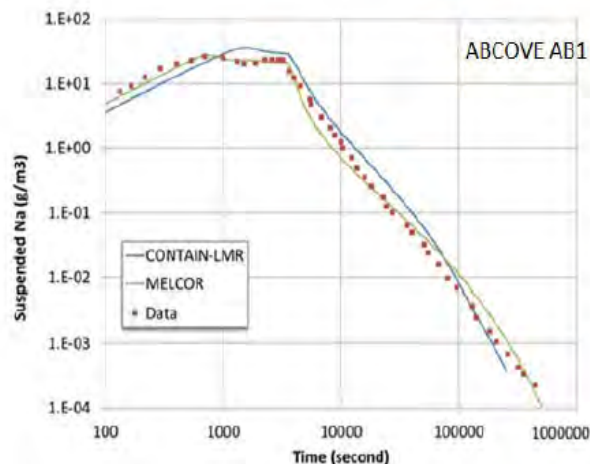
# MELCOR Flexibility - Common Phenomena



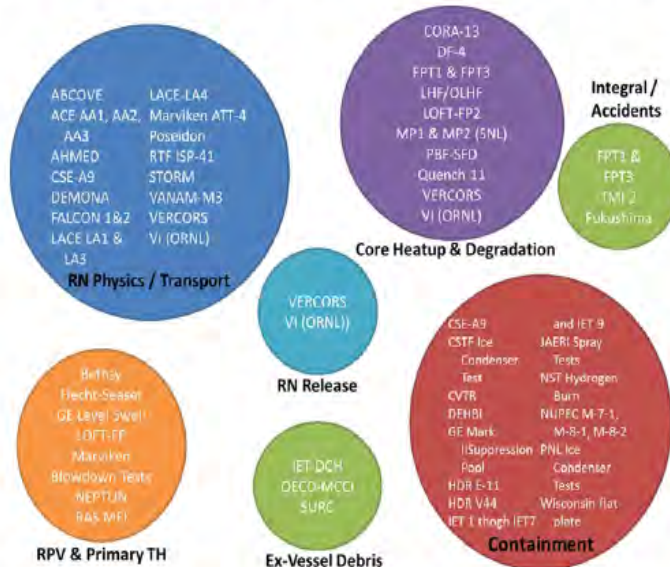
# MELCOR Verification & Validation Basis



## Primer & User Guide Reference Manual Assessment Problems



Specific to non-LWR application



## 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)



Sodium Fires  
(Completed)



Molten Salt  
(planned)



Sodium Reactors  
(planned)



HTGR  
(planned)



# MELCOR State-of-the-Practice Modeling

## Timeline for Evolution of MELCOR Modeling Practices

Circa 1985

Circa 1990

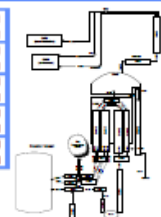
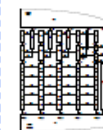
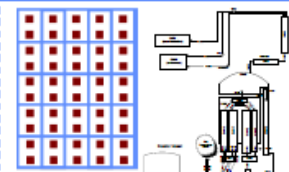
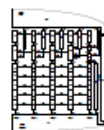
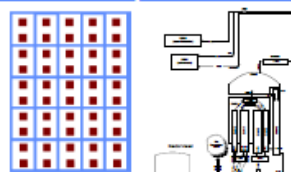
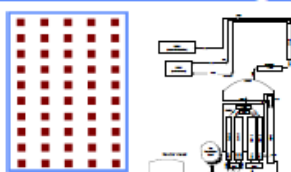
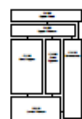
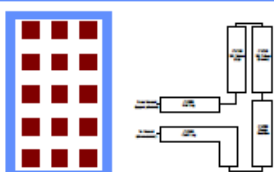
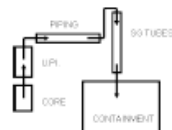
Circa 1995

Circa 1998

Present

Future

### Modeling Techniques



### Example Regulatory Applications

- 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

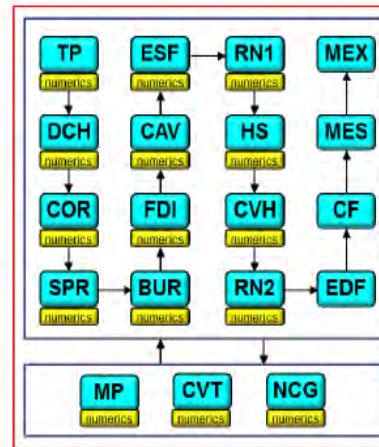
Generalized numerical  
solution engine

Hydrodynamics

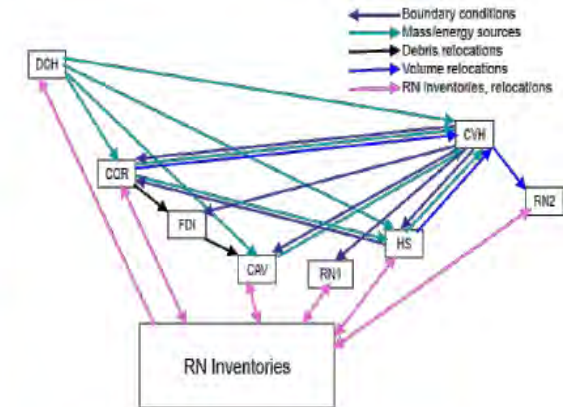
In-vessel damage  
progression

Ex-vessel damage  
progression

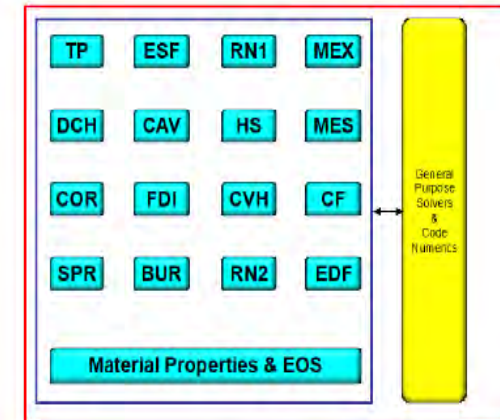
Fission product release  
and transport



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** & **Numerics**



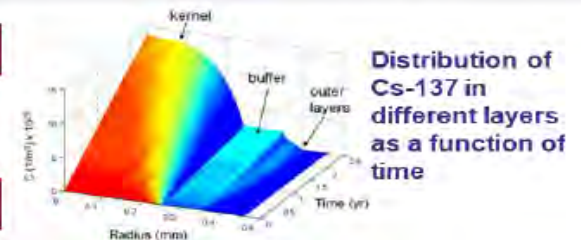
# MELCOR Data Requirements

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)



$$\frac{\partial C}{\partial t} = \frac{1}{r^m} \frac{\partial}{\partial r} \left( r^m \frac{\partial C}{\partial r} \right) - \lambda C + S$$

Diagram showing the equation with 'D' in a box under the diffusion term and 'S' in a box under the source term. Arrows point from these boxes to 'Experiments/Analysis' and 'SCALE' respectively.



# International Collaboration

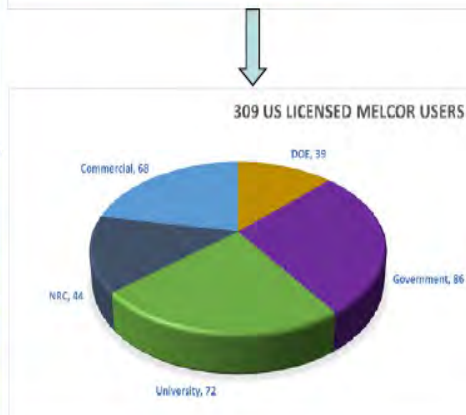
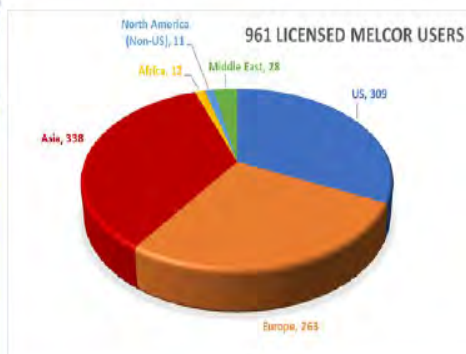


# User Groups & Technical Meetings

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

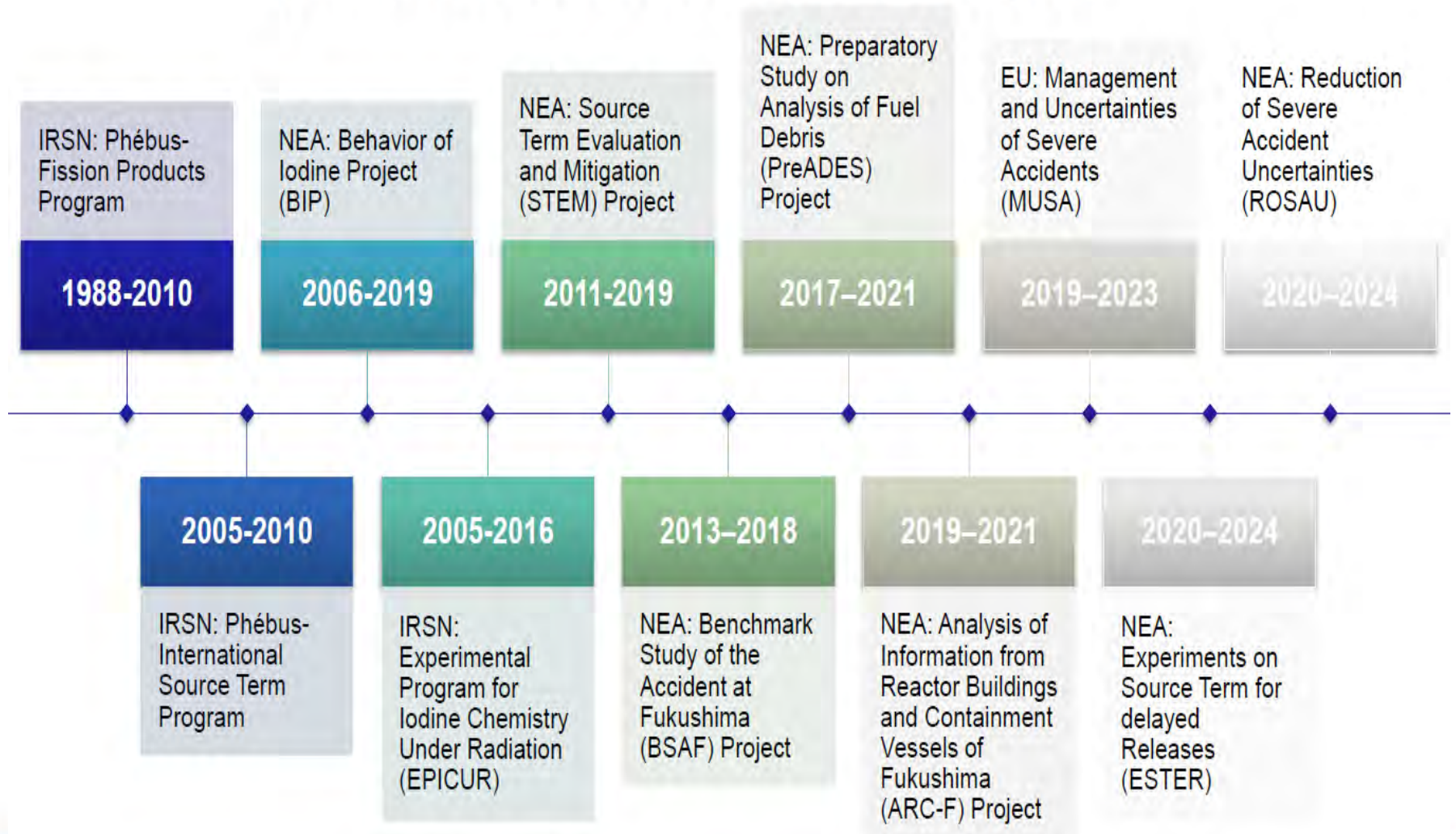


~1000 Code Users  
 Worldwide



ACRS meeting on Integration of Source  
 Term Activities in Support of Advanced  
 Reactor Initiatives, 02/17/2022

# International Severe Accident Projects





# 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



Review of Accident Tolerant Fuel Concepts with Implications to Severe Accident Progression and Radiological Releases

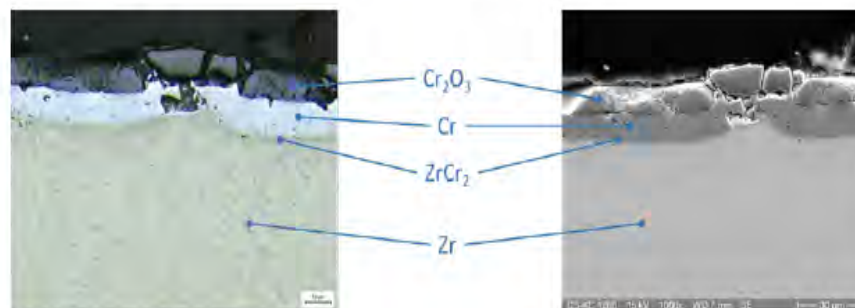
**NUREG/CR-7282**



Phenomena Identification Ranking Tables for Accident Tolerant Fuel Designs Applicable to Severe Accident Conditions

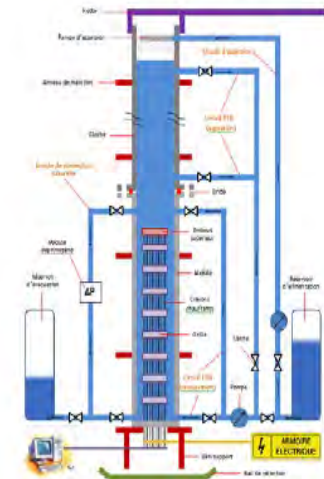
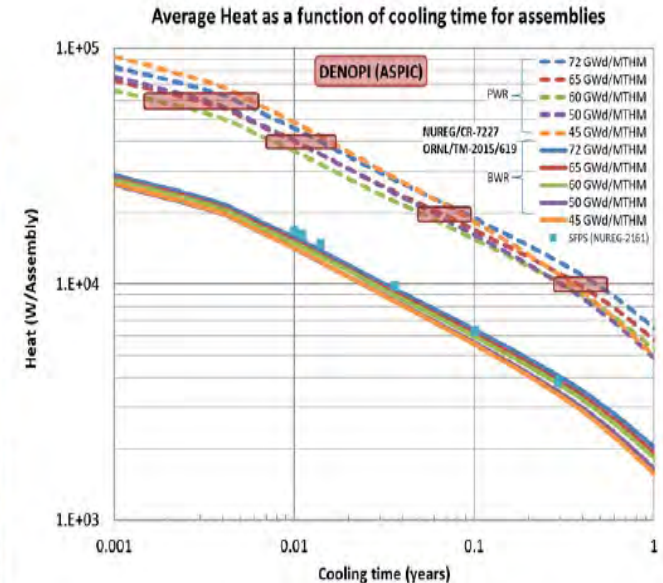
**NUREG/CR-7283**

Office of Nuclear Regulatory Research



# 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

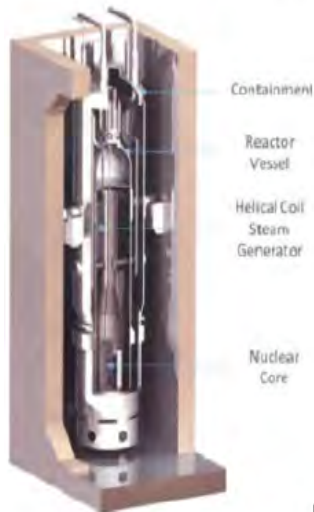


# MELCOR Applications



# Design Certification

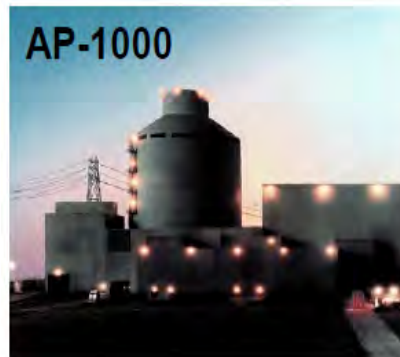
- Severe accident response and source term
- Containment response to design basis accident



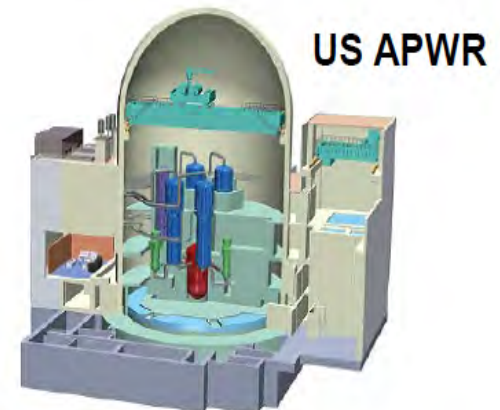
NuScale



APR-1400



AP-1000



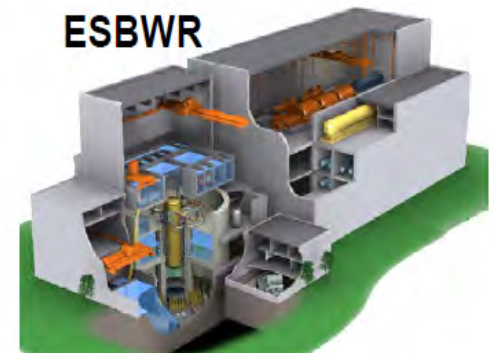
US APWR



US EPR

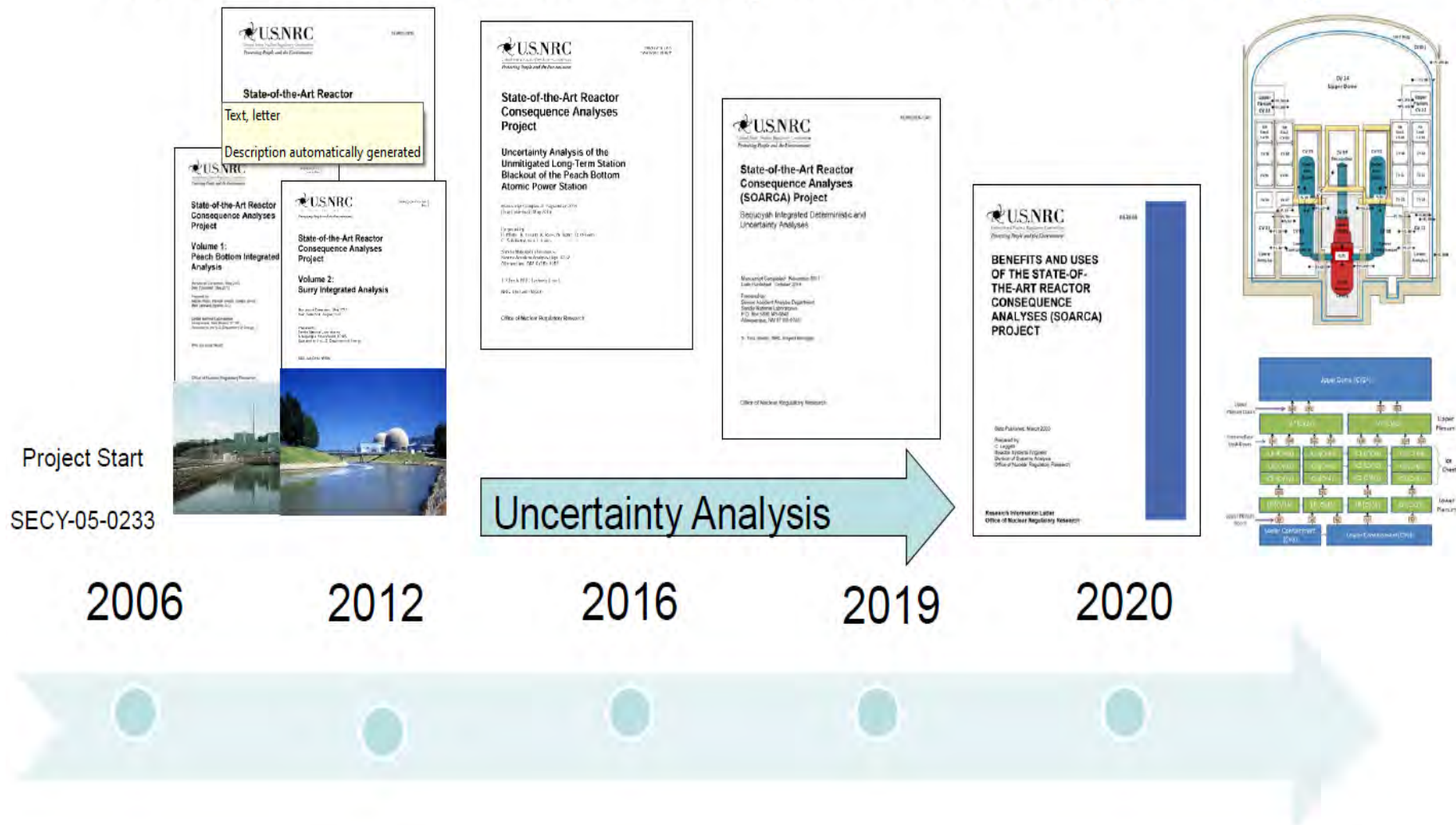


BWRX-300

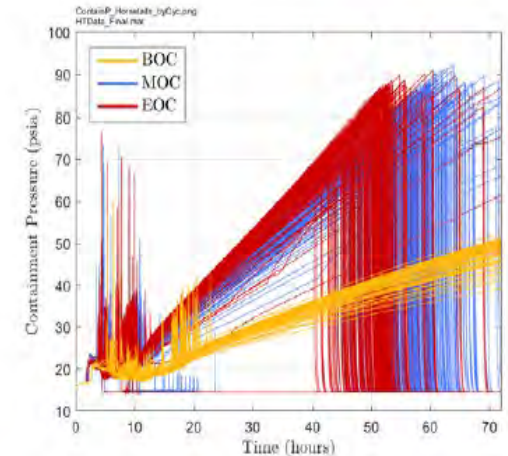
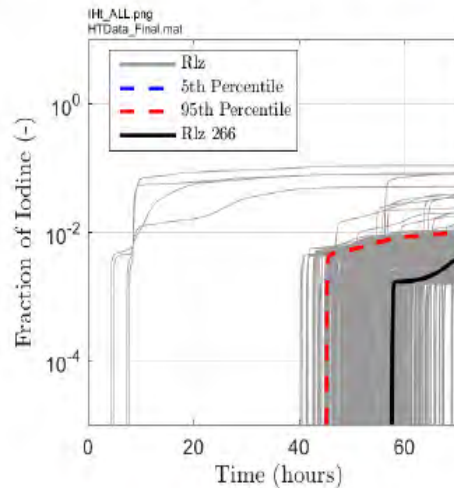
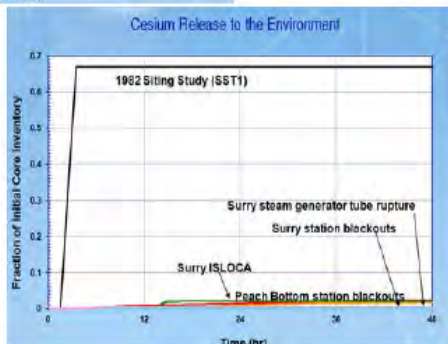
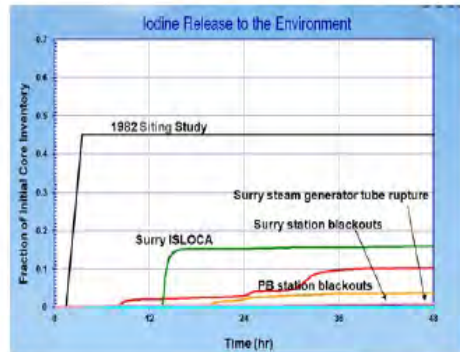


ESBWR

# State-of-the-Art Reactor Consequence Analysis (SOARCA)



# State-of-the-Art Reactor Consequence Analysis (SOARCA)



Uncertainty Analysis

2006

2012

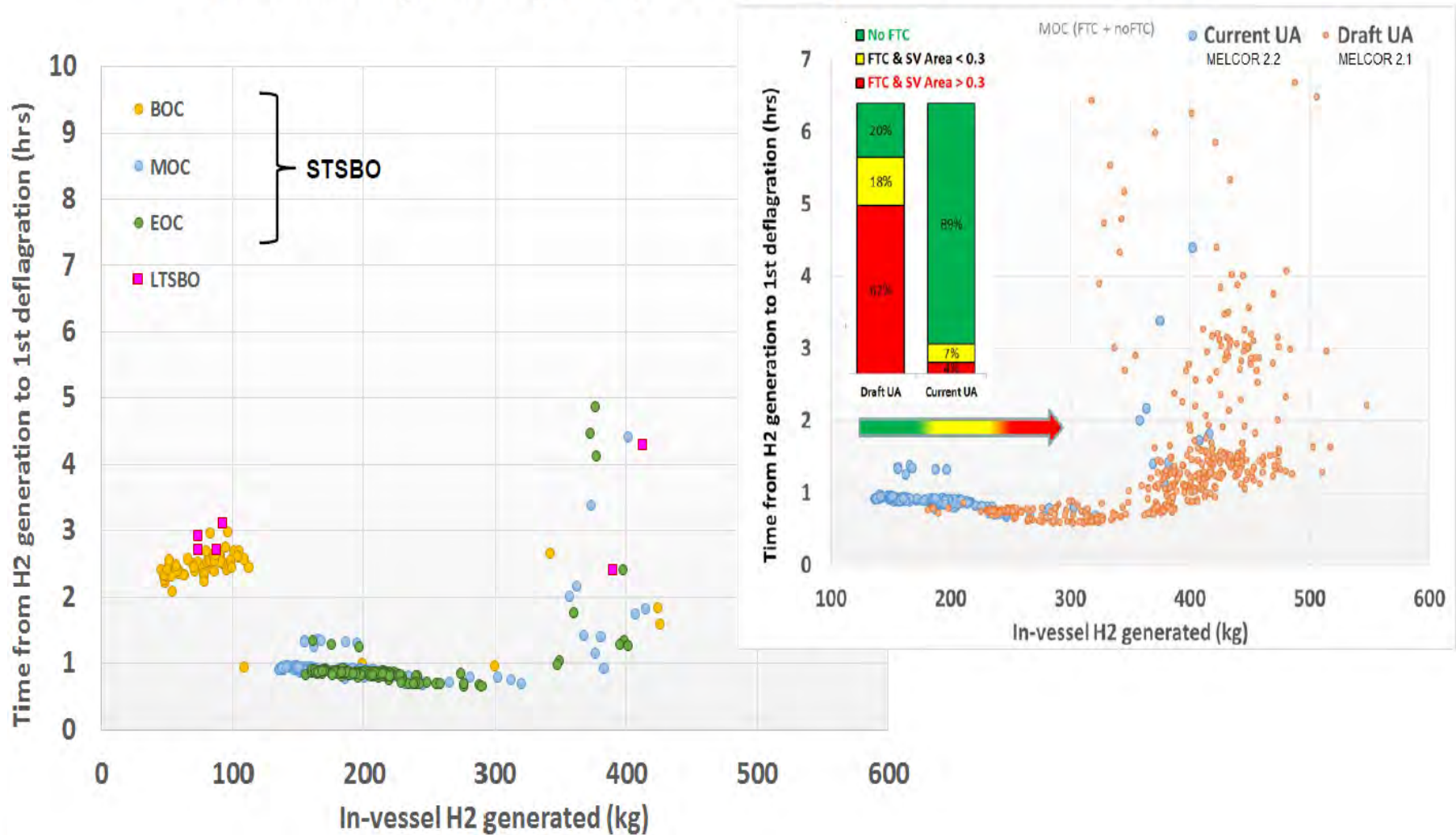
2016

2019

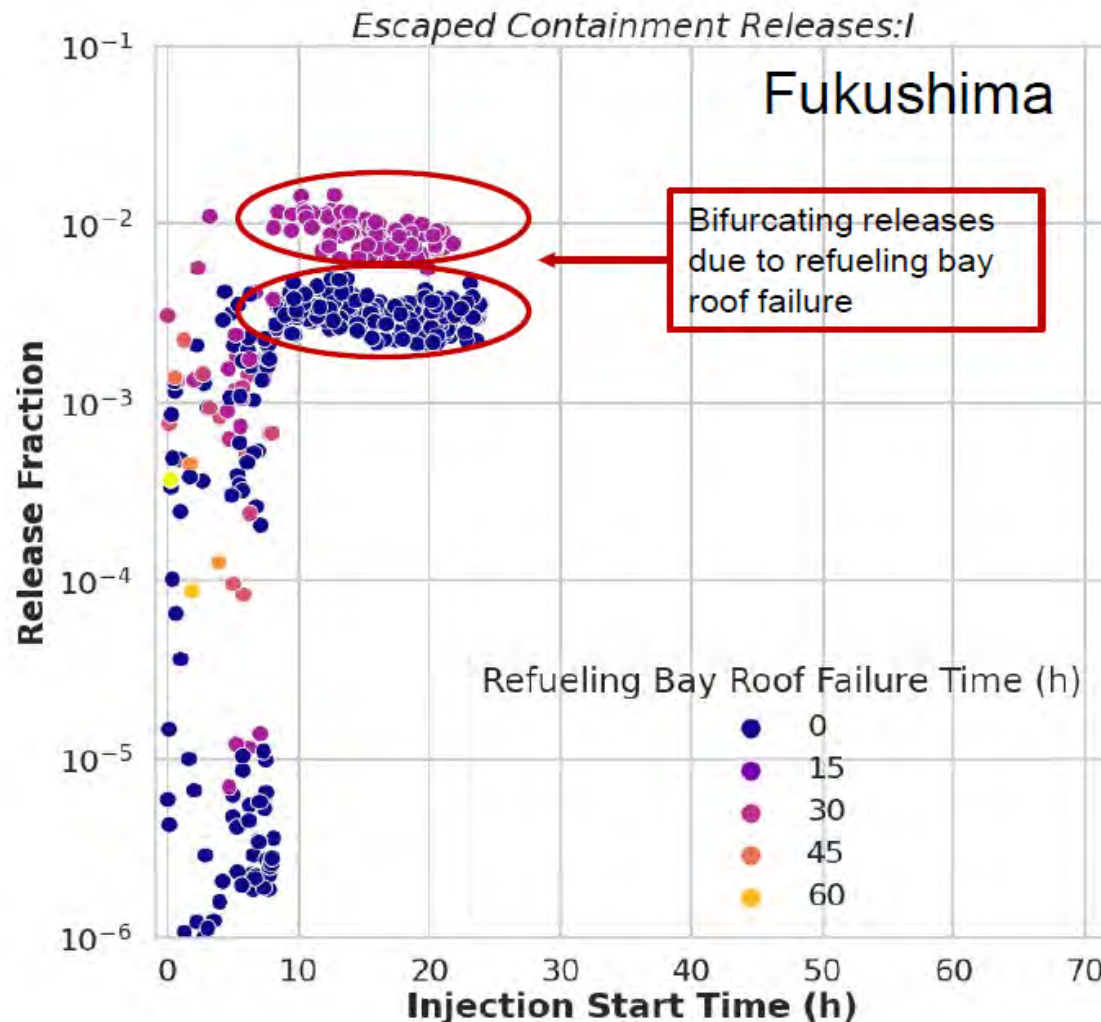
2020



# Uncertainty Analysis (SOARCA) NUREG/CR-7245

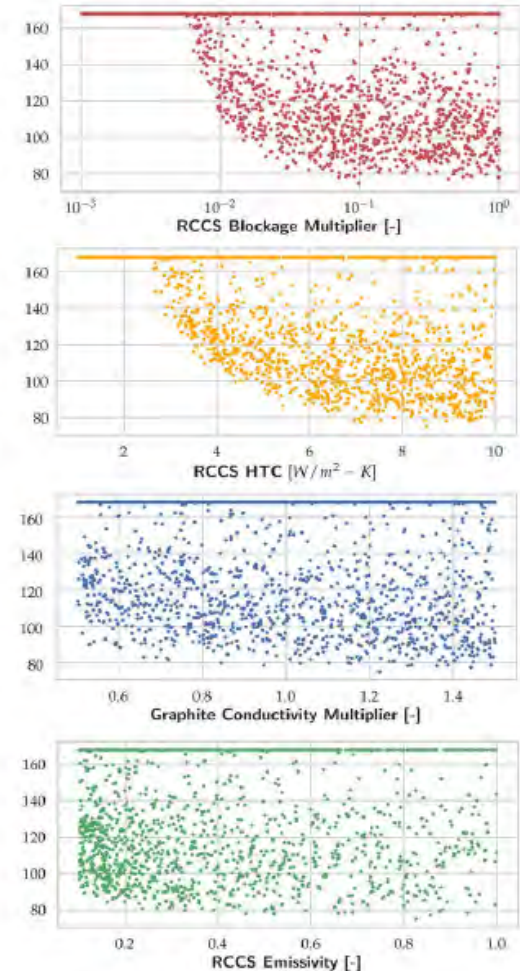


# Uncertainty Analysis Applications



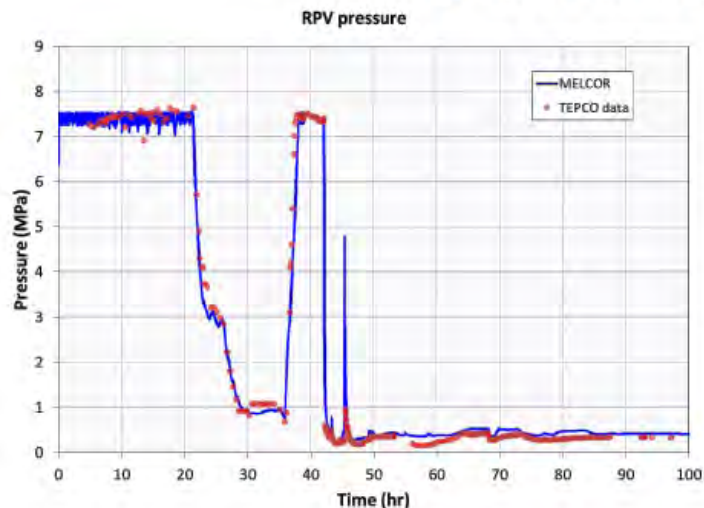
## Non-LWR

Time to Peak Fuel Temperature (h)

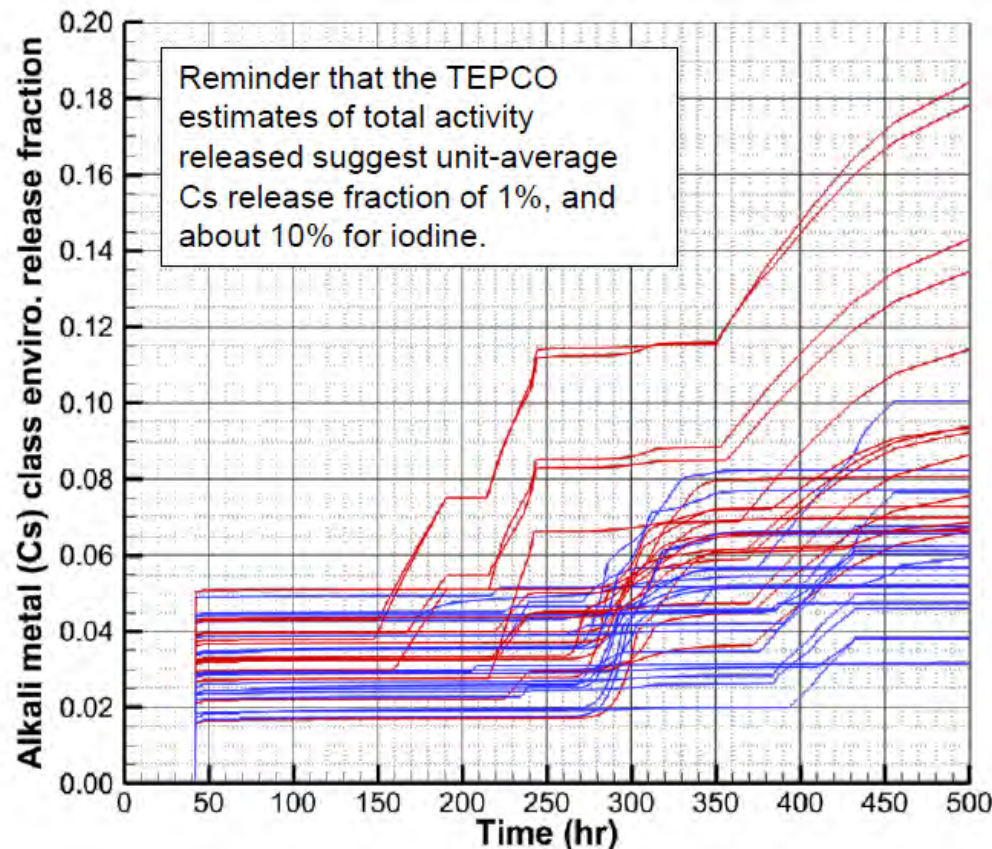
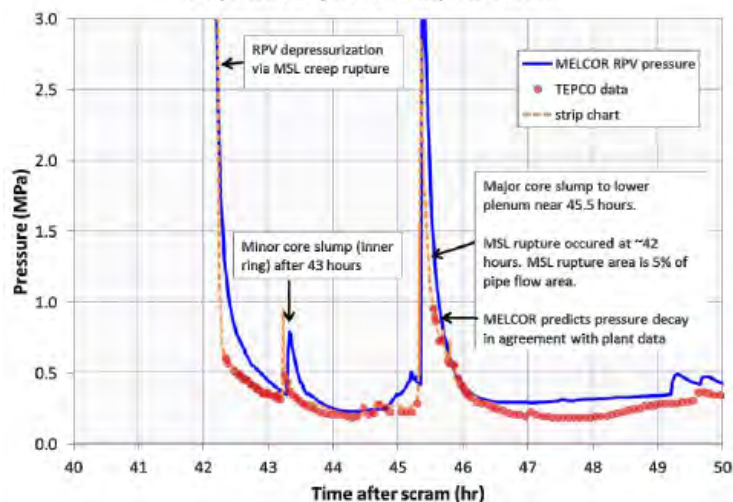




# Fukushima Forensics (Unit 3)



RPV pressure response after depressurization



# MELCOR Spent Fuel Pool Modeling



Laminar Hydraulic Analysis of  
a Commercial Pressurized  
Water Reactor Fuel Assembly

NUREG-1744  
REV. 1, 1988



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

NUREG-1744  
REV. 1, 1988

Office of Nuclear Regulatory Research

Reactor Safety NRC/  
CERD, N121512  
Mar 2010

**S**tatus Report on Spent Fuel  
Pools under Loss-of-Cooling  
and Loss-of-Coolant Accident  
Conditions

Final Report

Nuclear Safety  
NRC / CERD, N121512  
April 2010



**P**henomena Identification  
and Ranking Table

RBD Priorities for Loss-of-Cooling  
and Loss-of-Coolant Accidents in  
Spent Nuclear Fuel Pools



Convective Heat Transfer Surfaces:

Ring 1

Clad, Canister / Water  
Rods, Rack

Ring 2

Clad, Canister /  
Water Rods, Racks

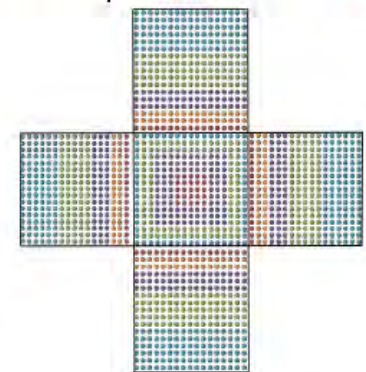
Radiative Heat Transfer Flow Path:

Fuel → Clad → Canister / Water Rods → Rack → Canister / Water Rods → Clad → Fuel

Ring 1

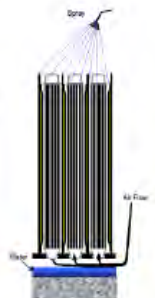
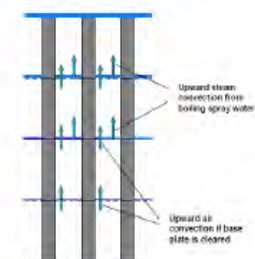
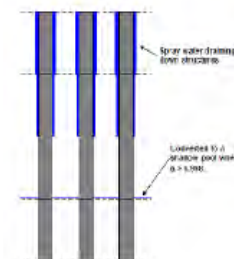
Racks

Ring 2



Multiple fuel rod  
components in the  
center assembly  
(Ring 1) and four  
peripheral assemblies  
(Ring 2)

Integral Spray Model



ACRS meeting on Integration of Source  
Term Activities in Support of Advanced  
Reactor Initiatives, 02/17/2022

27

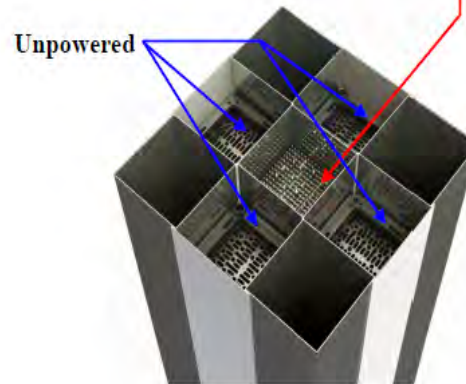
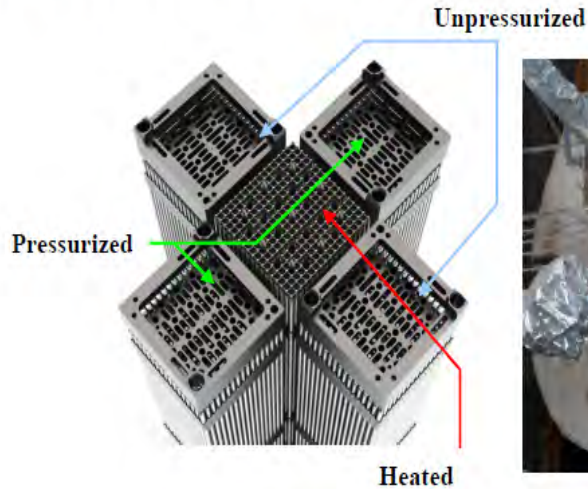
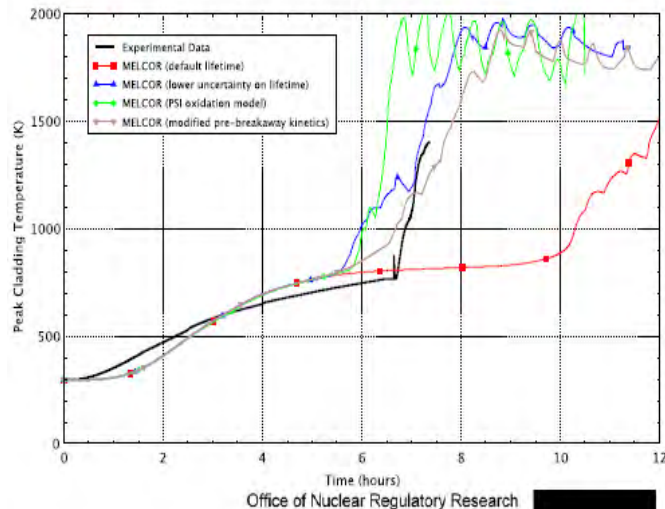


# MELCOR SFP Modeling Basis



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



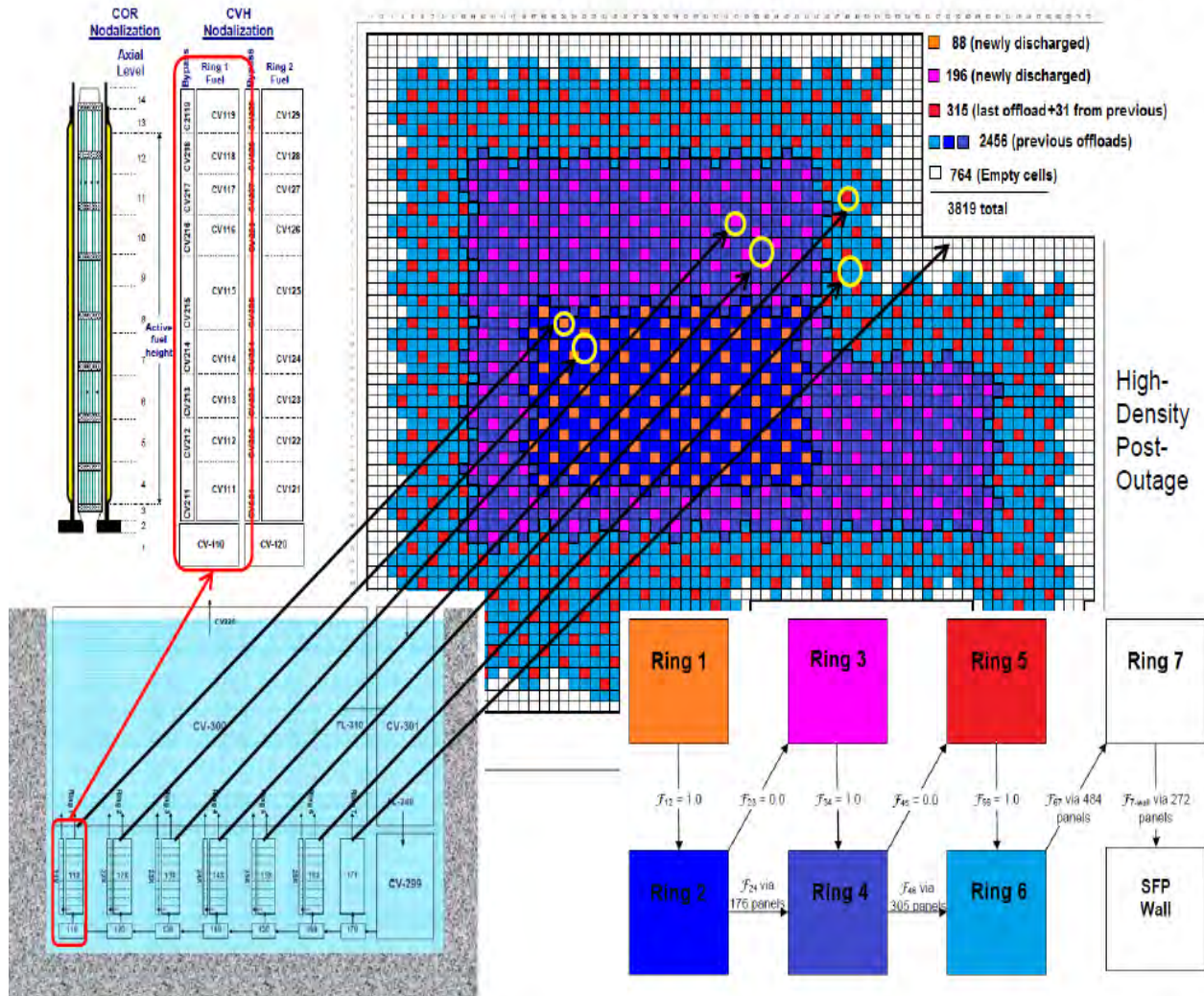
Before



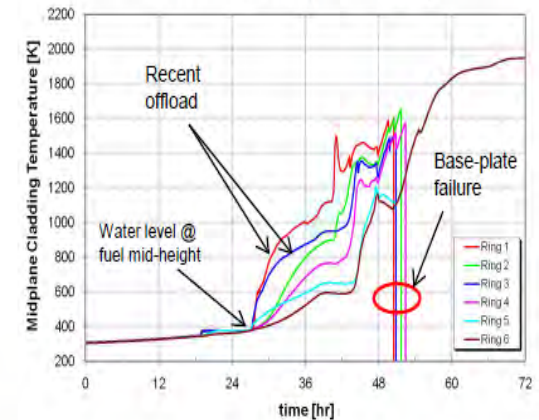
After



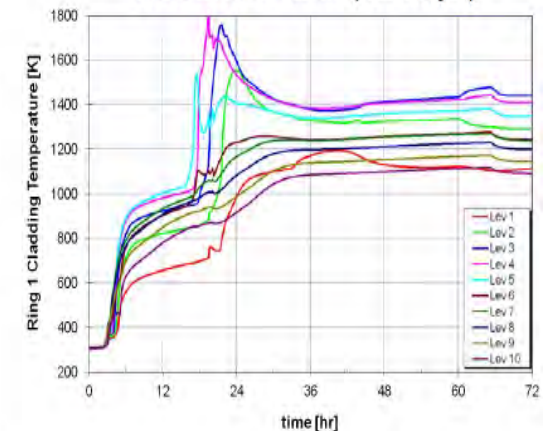
# MELCOR SFP Model (NUREG-2161)



Small Leak (13 days)



Moderate Leak (13 days)

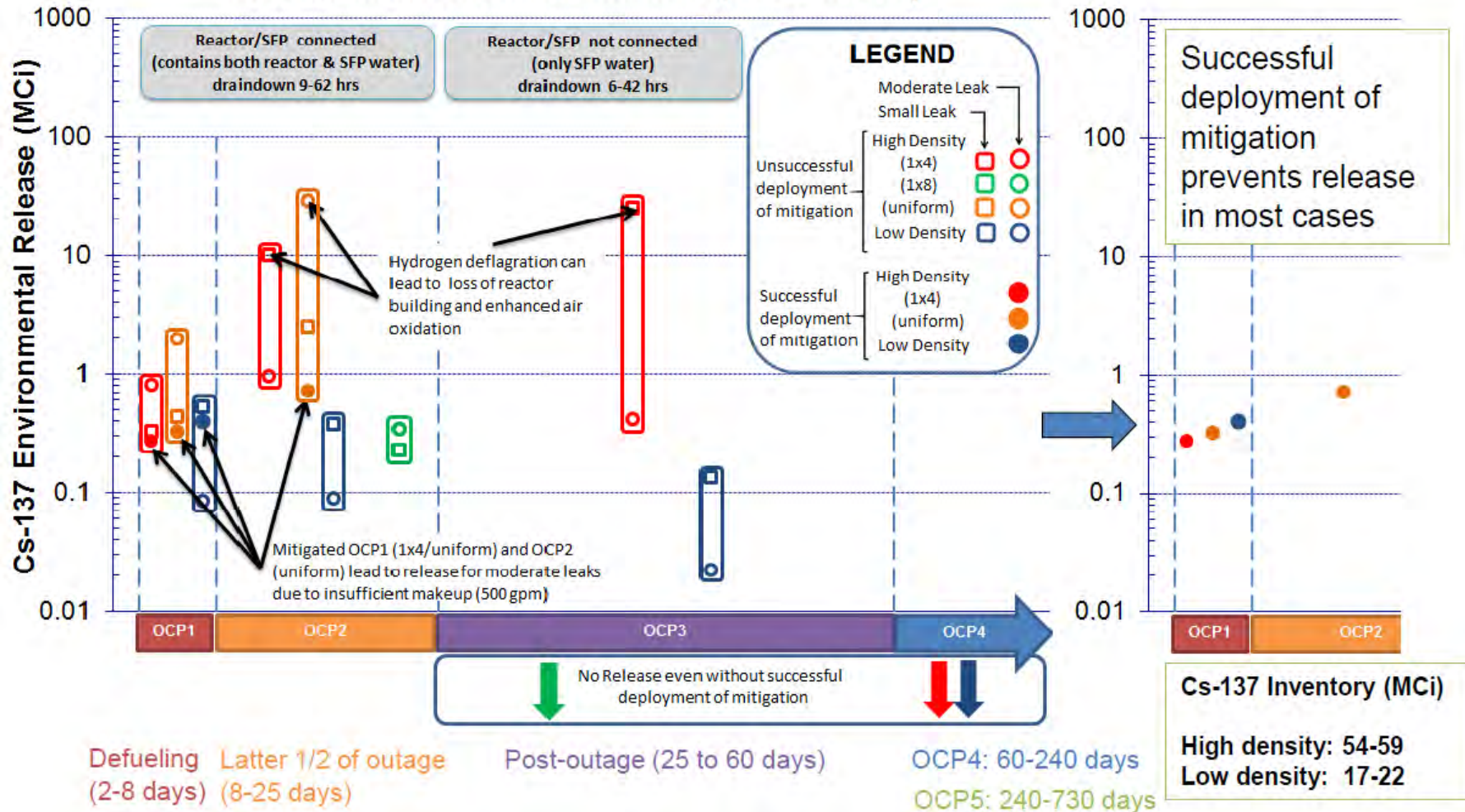


**BREAK**



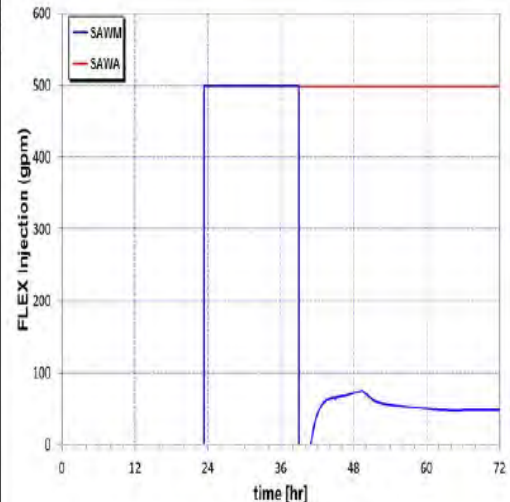
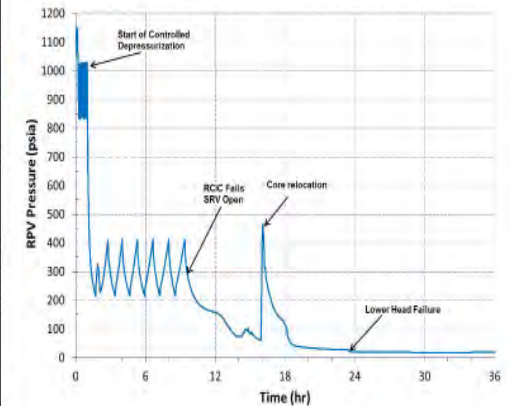
# MELCOR Results (NUREG-2161)

Cases that lead to release (OCP1/2/3)



# Containment Protection and Release Reduction (NUREG-2206)

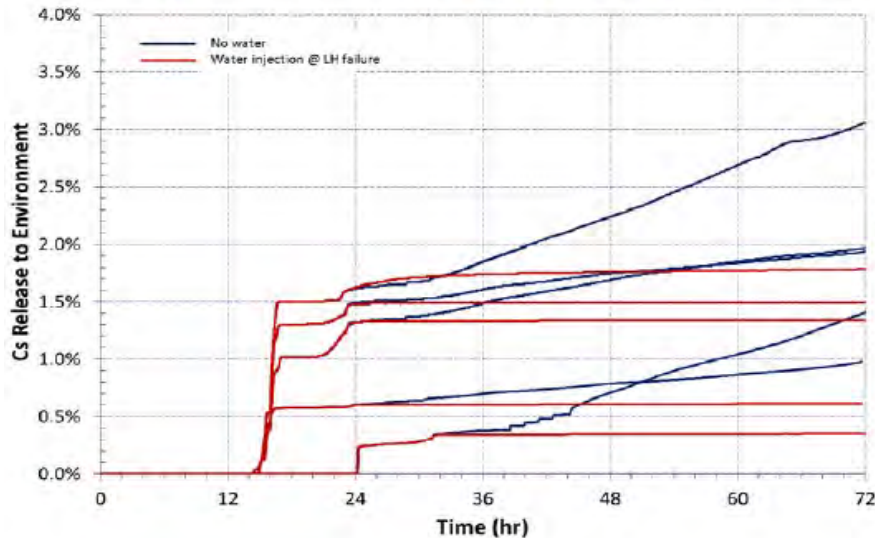
		Pre Core Damage						Post Core Damage					
		RPV Pressure control	RCIC Operation			Anticipatory Venting	Flex Operation		SRV Operation	Venting			
		Availability (hr)	RCIC Availability (hr)	RCIC Suction	Failure Temp (F)	Open SRV after RCIC fails	Setpoint (psig)	Injection @ LH failure	WW Level Control Injection @ 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	
4Ai(1)	8	72	16	SP	230	N	15	RPV	throttle	Y	WW	PCPL	
4Ai(1)	9	72	16	SP	230	Y	15	RPV	throttle	Y	WW	PCPL	
4Ai(1)	12	72	16	SP	230	N	15	RPV	throttle	Y	WW	PSP	
4Ai(1)	13	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL	
4Ai(1)	14	72	16	CST	230	N	15	RPV	0	Y	WW	PCPL	
4Ai(1)	15	72	16	CST	230	N	15	RPV	throttle	Y	WW	PCPL	
4Ai(1)	18	72	16	CST	230	N	15	RPV	500	Y	WW/DW	PCPL	
4Ai(1)	16	72	16	CST	230	N	15	RPV	500	Y	WW/DW	PCPL	
3B	21	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	24	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
	24dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	22	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	22dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	23	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	25	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	26	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	27	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	28	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
	28dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	32	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	30	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
	30dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	29	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
	29dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	31	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
	31dw	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3A	41	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL	
3B	43	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3A	42	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL	
3B	44	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(1)	47	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL	
4Ai(2)	48	72	16	SP	230	N	15	RPV	0	Y	WW	PCPL	
3B	45	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	46	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	49	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
4Ai(2)	50	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	51	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	52	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	
3B	53	72	16	SP	230	N	15	DW	0	Y	WW	PCPL	



ACRS meeting on Integration of Source Term Activities in Support of Advanced Reactor Initiatives, 02/17/2022

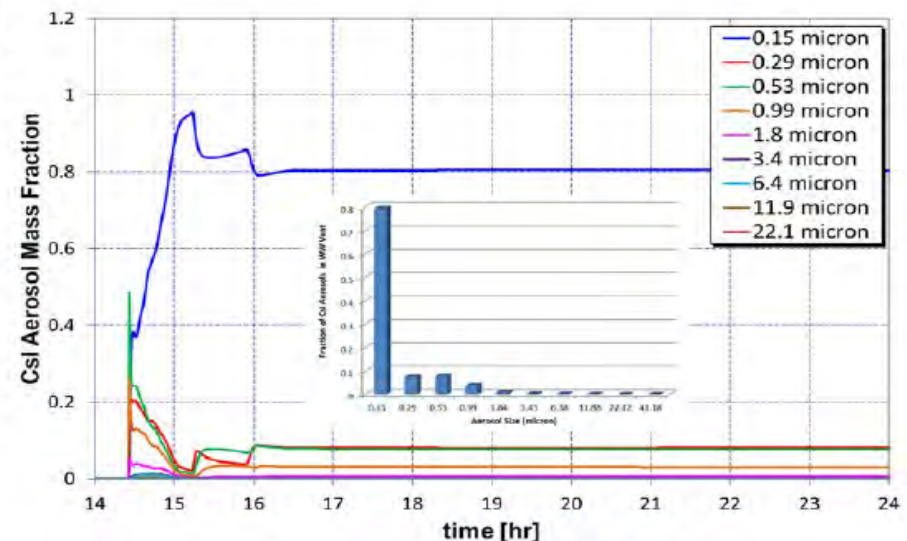


# Containment Protection and Release Reduction (NUREG-2206) - Mark I Results

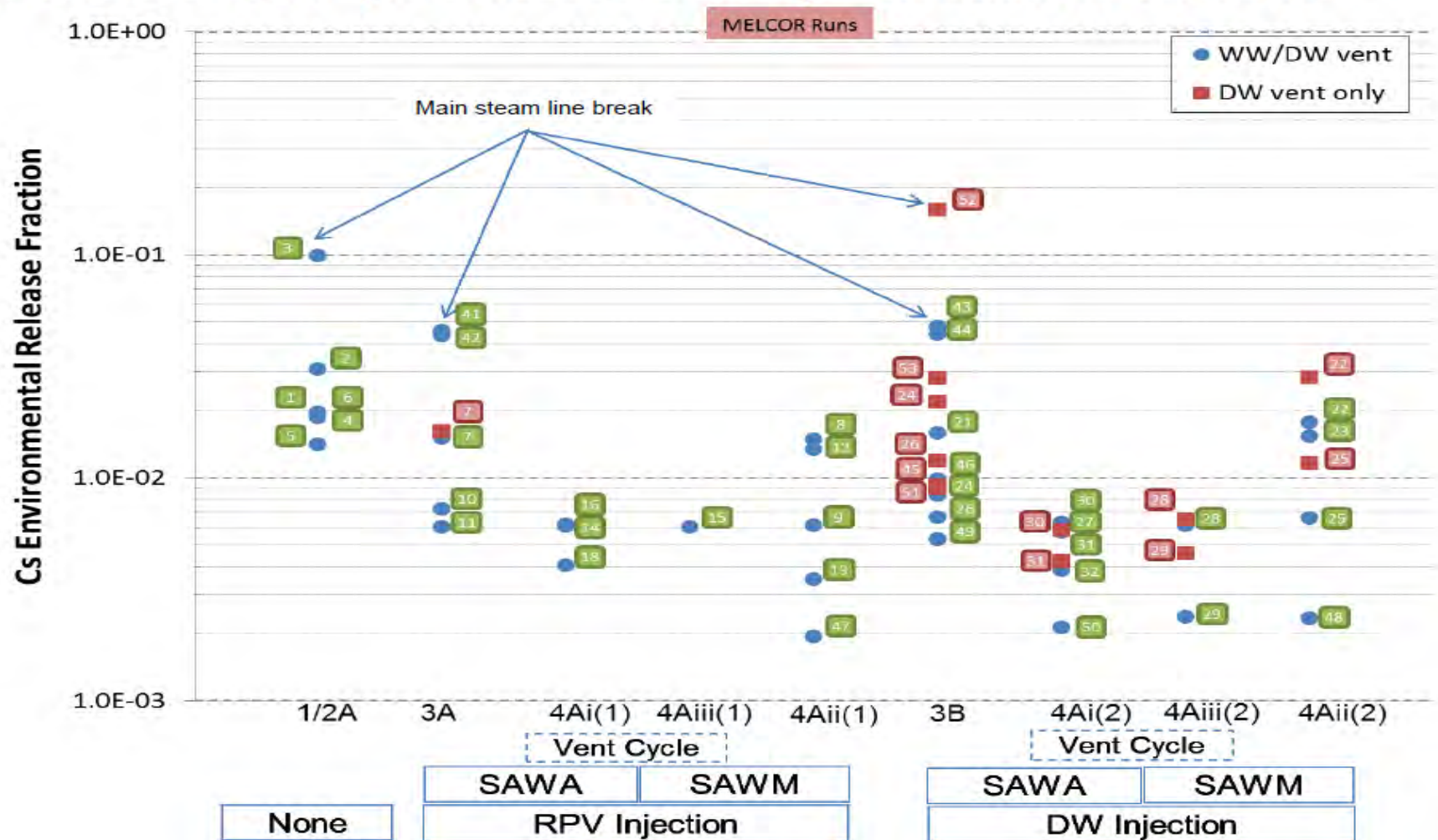


Water addition at lower head failure has the benefit of mitigating further release, but does not affect the release at the time of venting

Particle size distribution dominated by very small aerosols at the time of venting



# Containment Protection and Release Reduction (NUREG-2206) - Mark I Results



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# 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

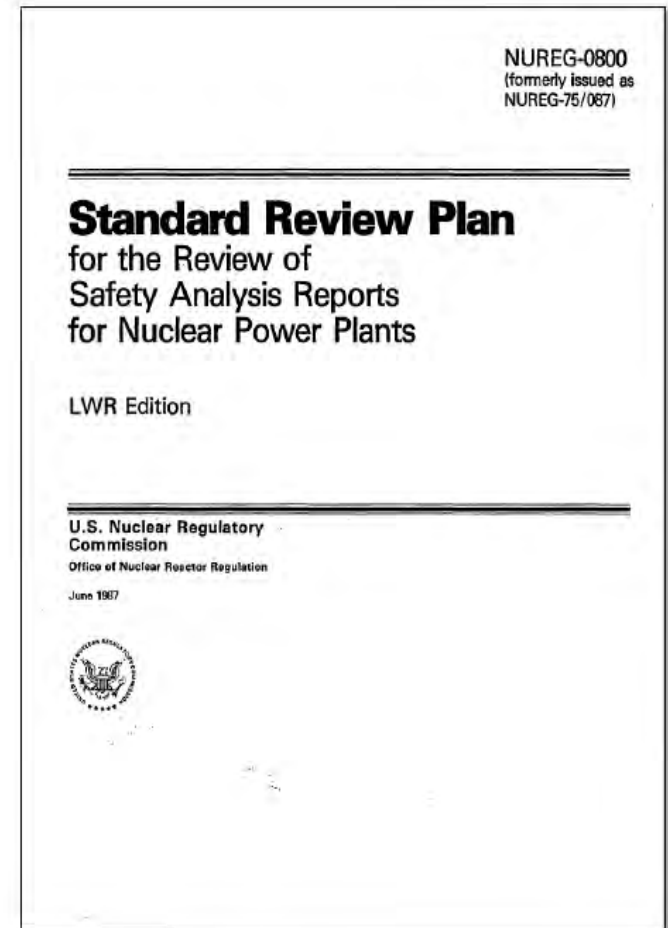
# MELCOR application to new reactors



# Standard Review Plan

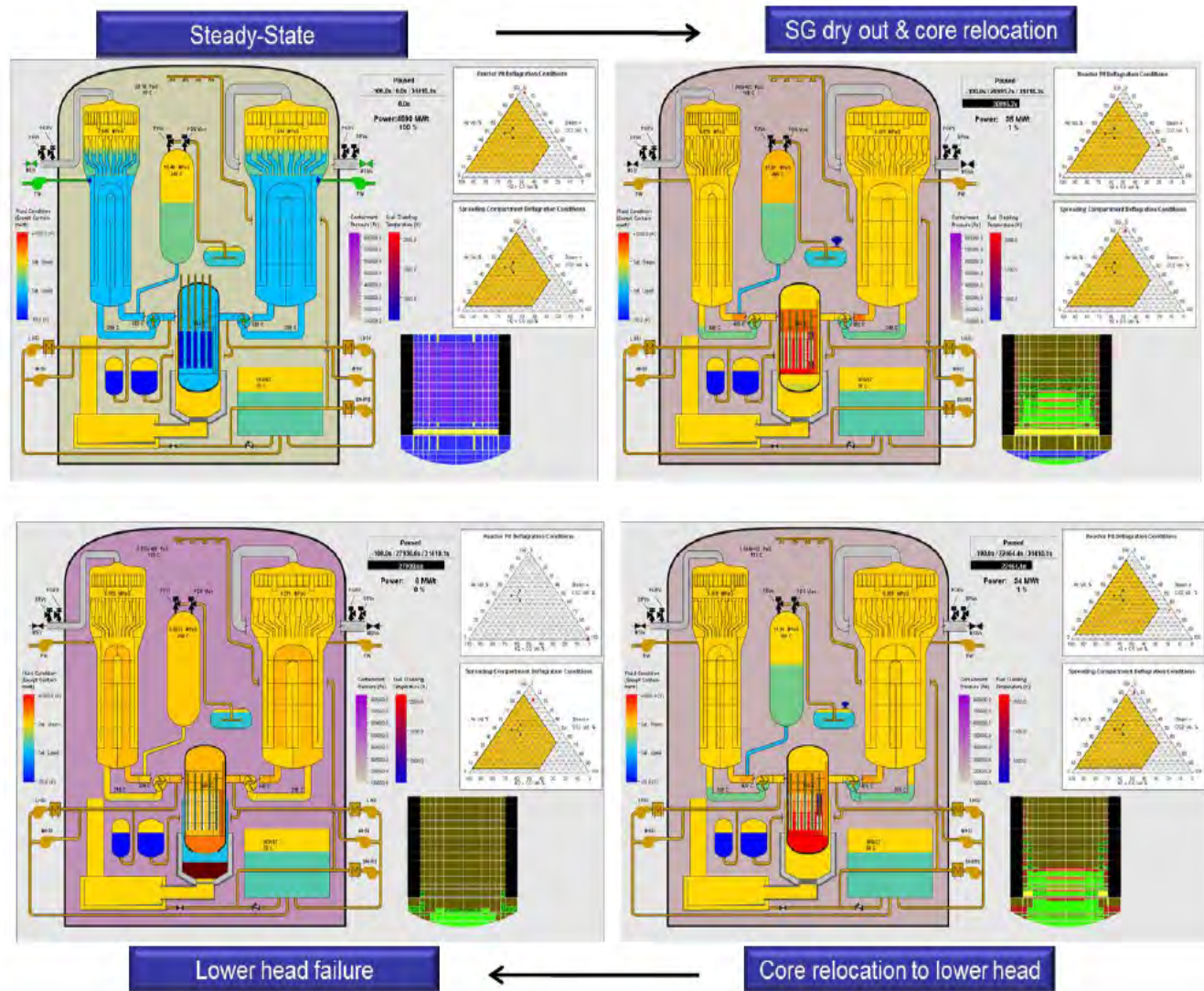
## Staff independent analysis

- Independent assessment of plant response and source term
- Scenarios from the PRA
- Engage with the applicant to resolve differences with the applicant's analysis



# Large LWRs

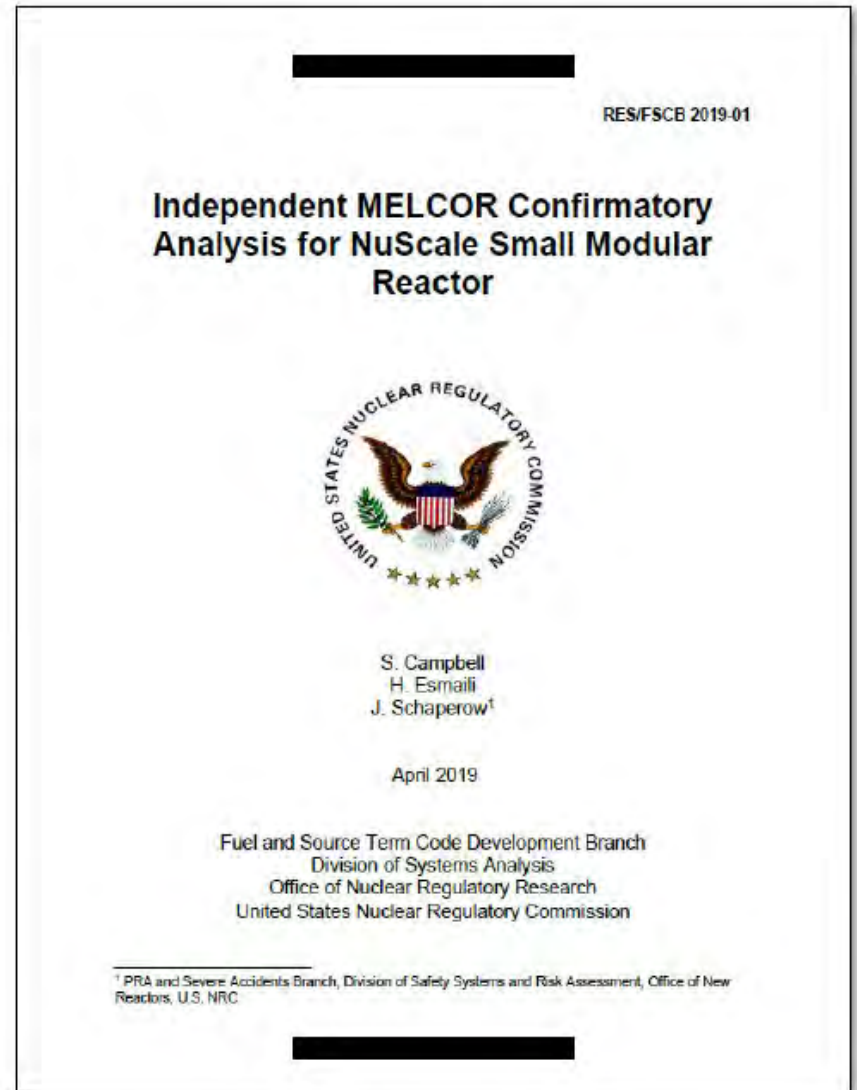
- ABWR
- AP-600
- System 80+
- AP-1000
- EPR
- APWR
- ESBWR
- APR-1400



ACRS meeting on Integration of Source Term Activities in Support of Advanced Reactor Initiatives, 02/17/2022

# SMRs

- NuScale
- mPower
- Westinghouse SMR
- BWRX-300





# 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



# SCALE/MELCOR non-LWR source term demonstration project

# Outline

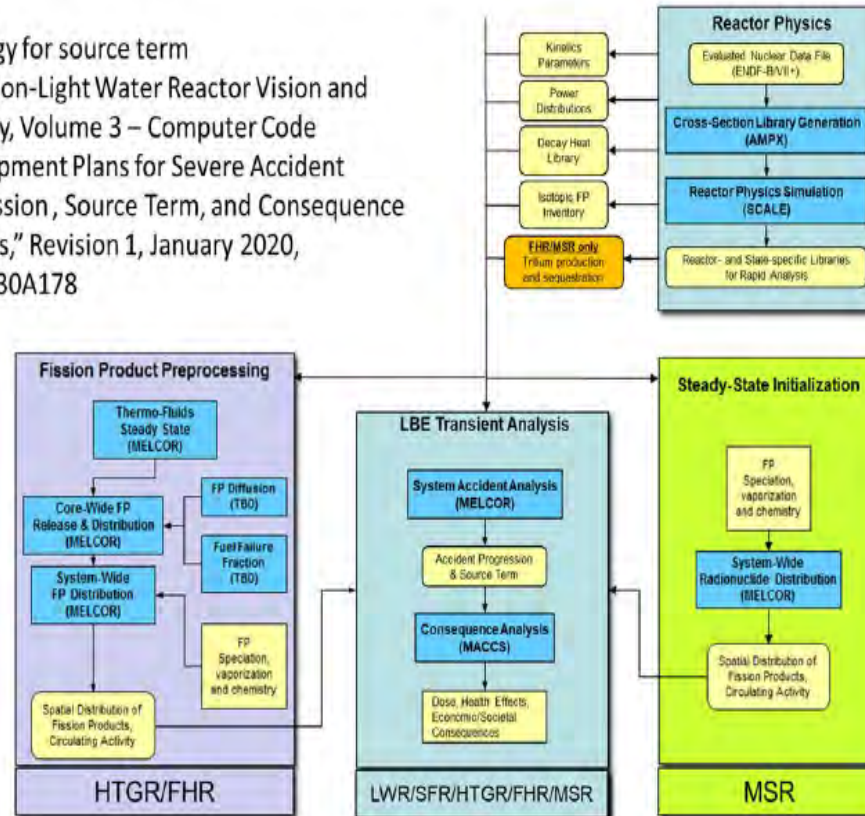
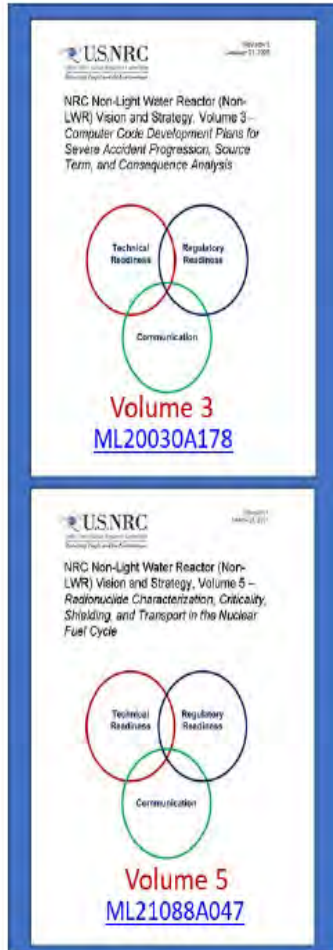
- NRC strategy for non-LWR source term analysis
- Project objectives
- Public workshops
- Sample results
  - Heat pipe reactor (HPR)
  - High-temperature gas-cooled reactor (HTGR)
  - Pebble-bed molten-salt-cooled reactor (FHR)
- Summary

# 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



ACRS meeting on Integration of Source Term Activities in Support of Advanced Reactor Initiatives, 02/17/2022

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# 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

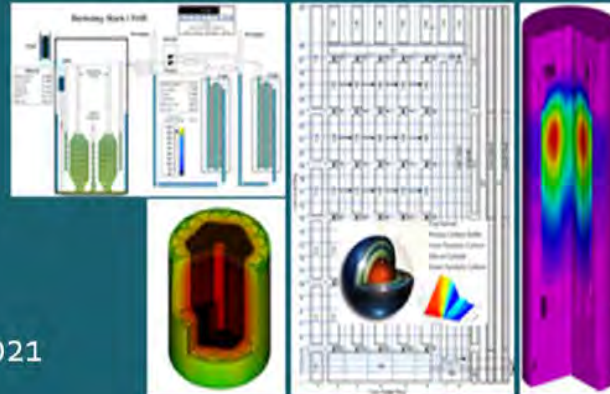
# Public Workshops

## 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

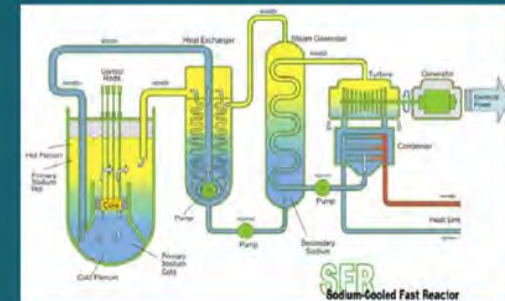
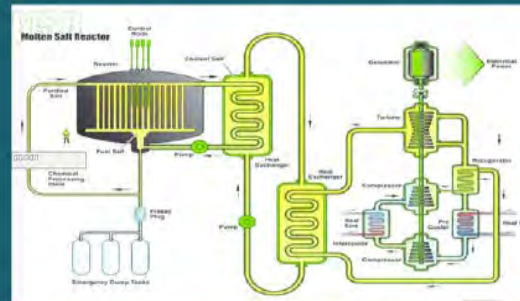


For More  
Information



## Coming in 2022

Molten-salt fueled reactor  
Sodium-cooled fast reactor

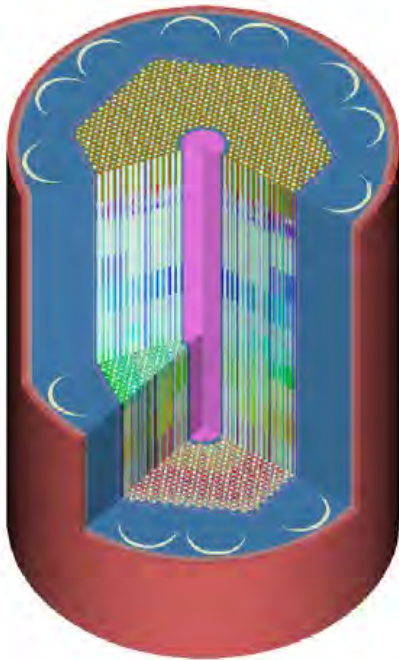


<https://www.nrc.gov/reactors/new-reactors/advanced/details.html#non-lwr-ana-code-dev>

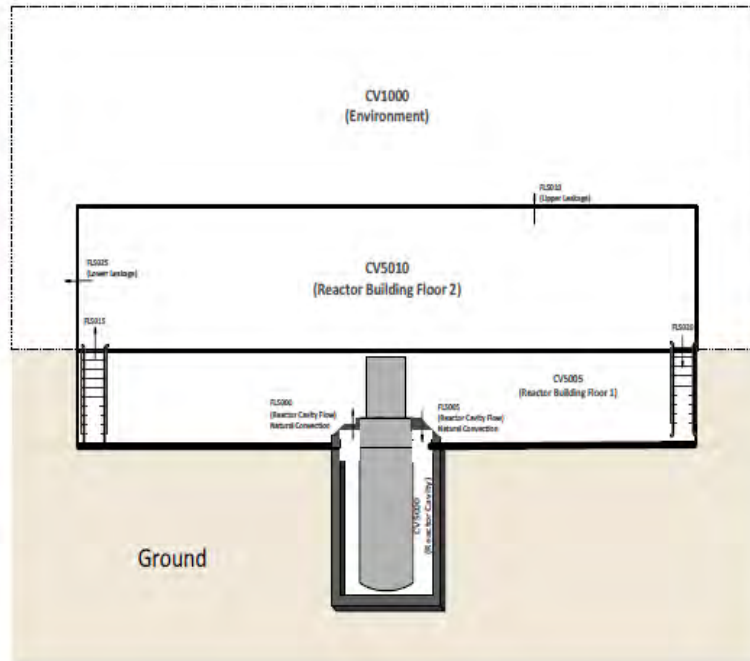
# Sample Results

calculations by ORNL and SNL

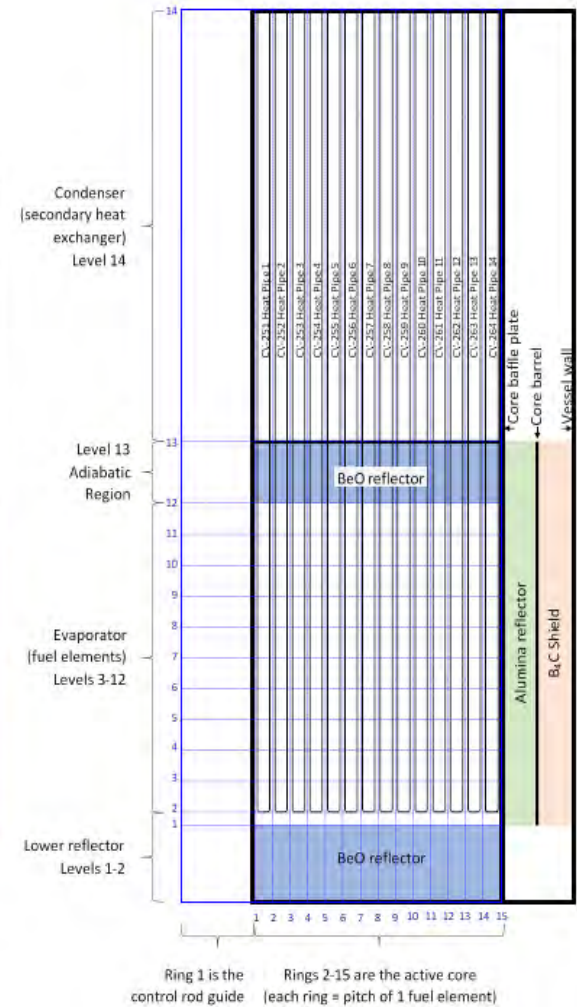
# 



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

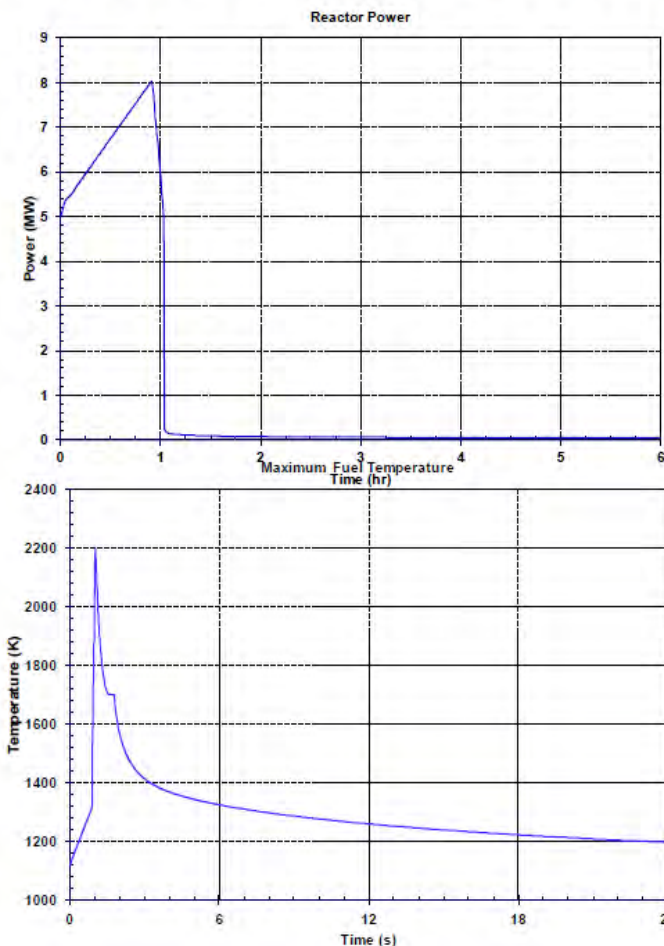
- Negative fuel temperature reactivity feedback limits the rate of power increase

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)



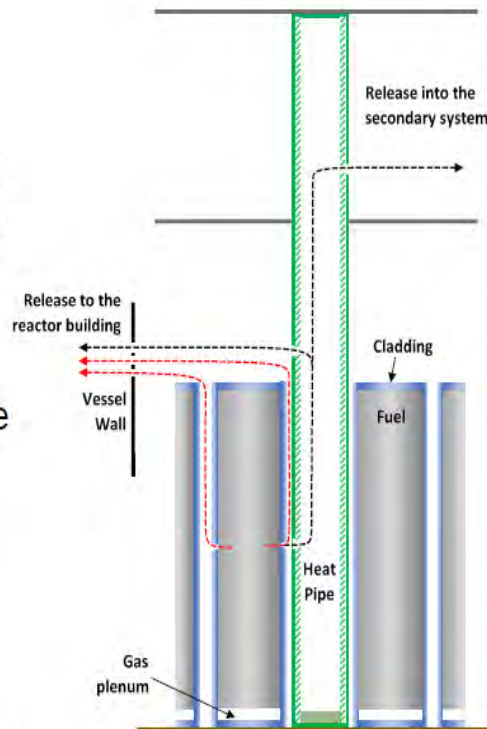
# HPR – reactivity addition accident with delayed scram

## Cladding failure at 1650 K resulting in fission product release

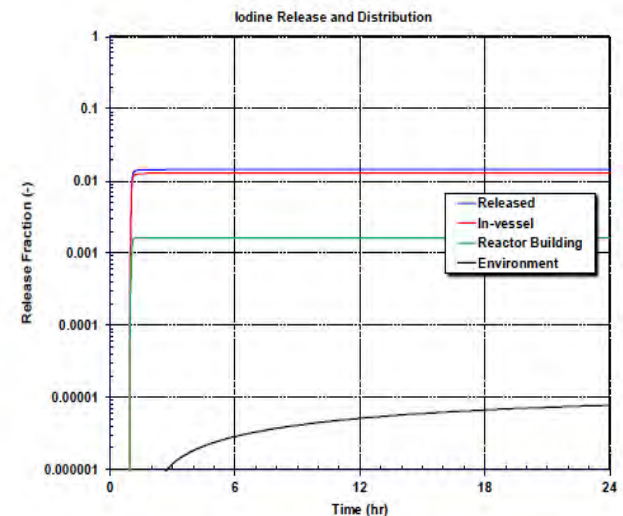
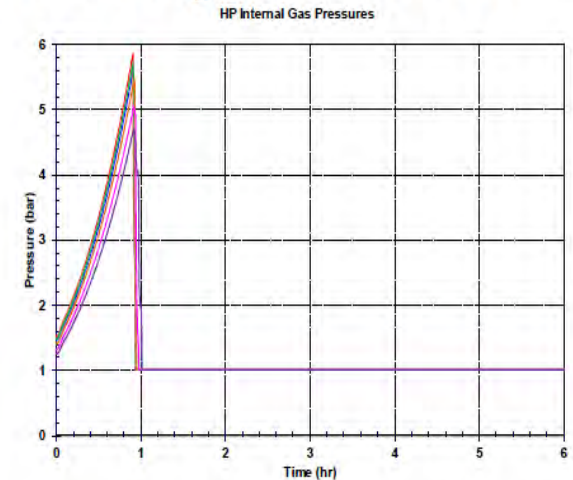
- Heat Pipes (HP) that exceeded the boiling limit rapidly heat to cladding failure (1650 K)
- ~20% of the 1134 HPs and fuel elements failed
- HP depressurization on failure drive release from the vessel

## 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<sup>2</sup>
- Building leakage is 1.8 in<sup>2</sup>





[illegible]



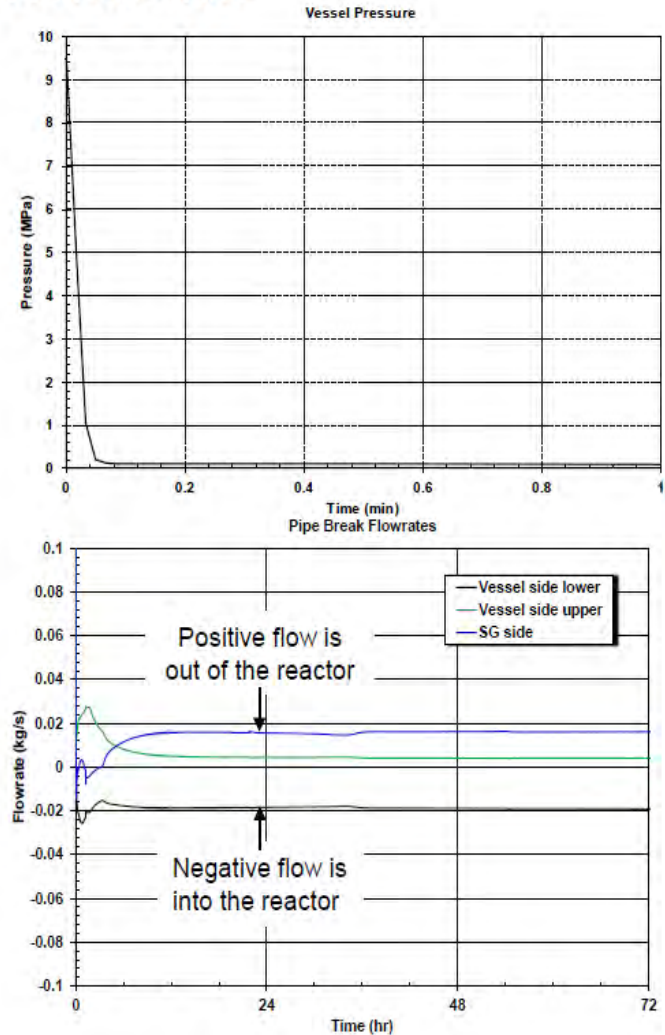
# HTGR – loss of coolant accident

## 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



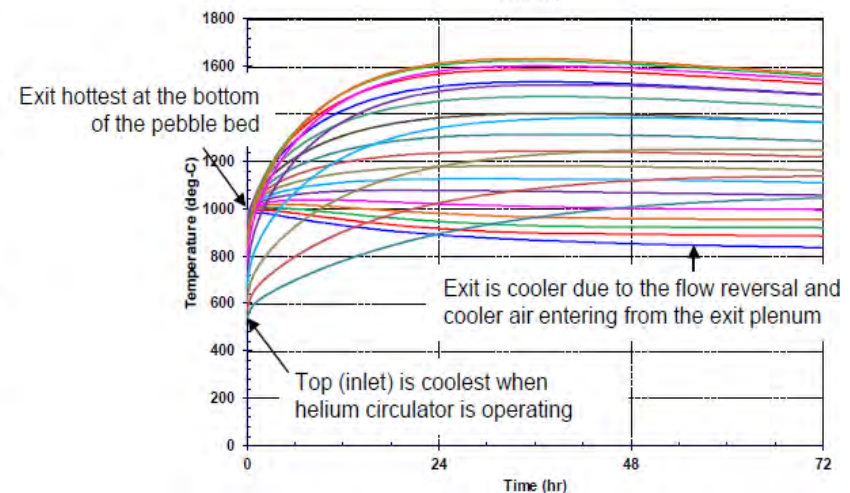
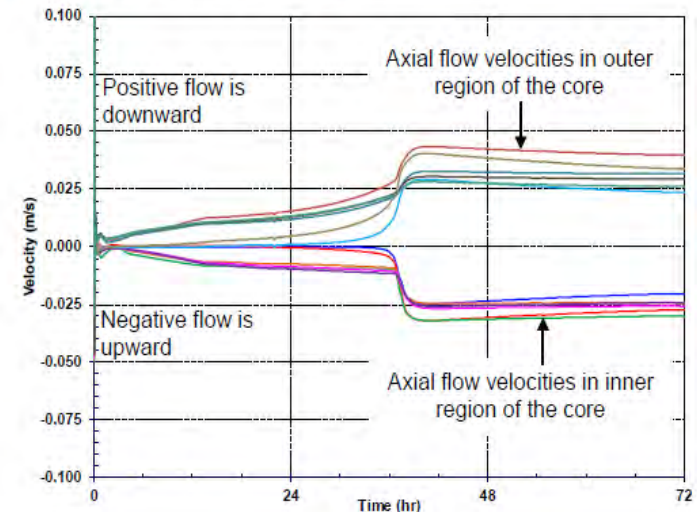
# HTGR – loss of coolant accident

## 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)

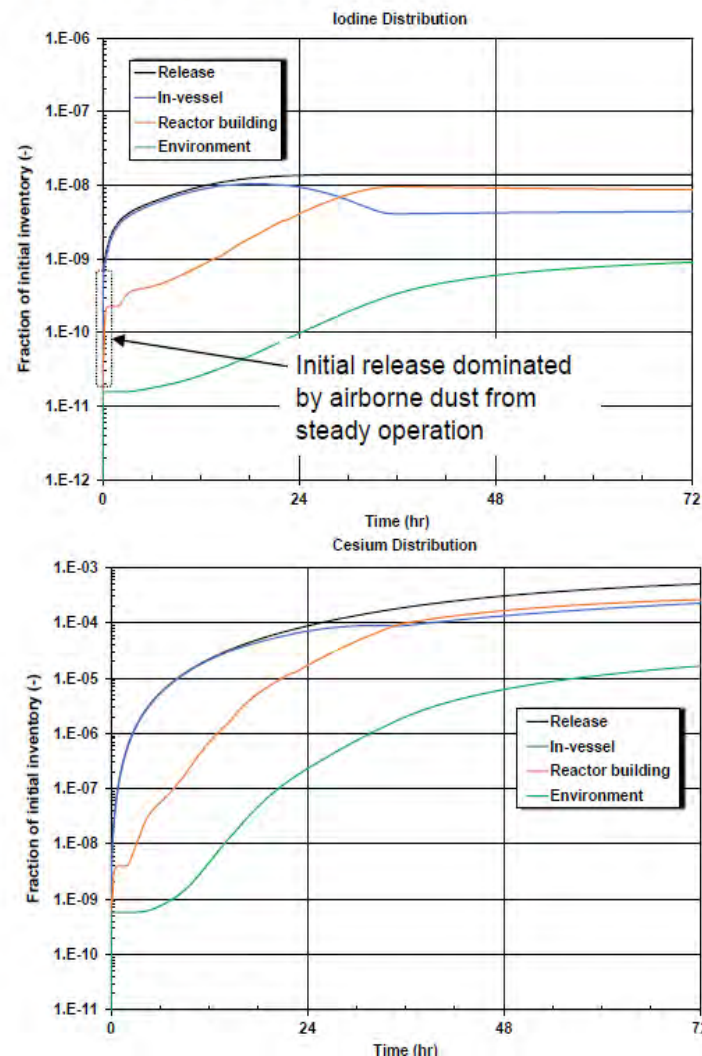




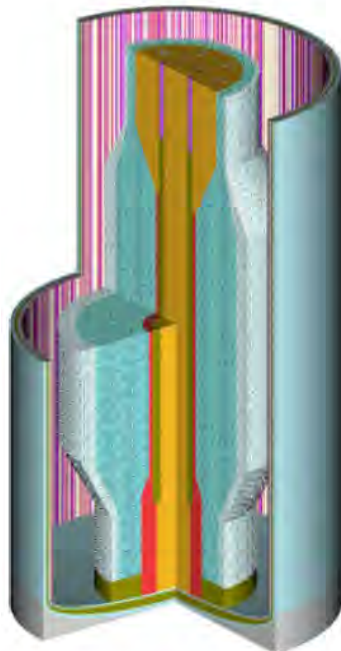
# HTGR – loss of coolant accident

## The impact of the low TRISO failure fraction leads to small releases

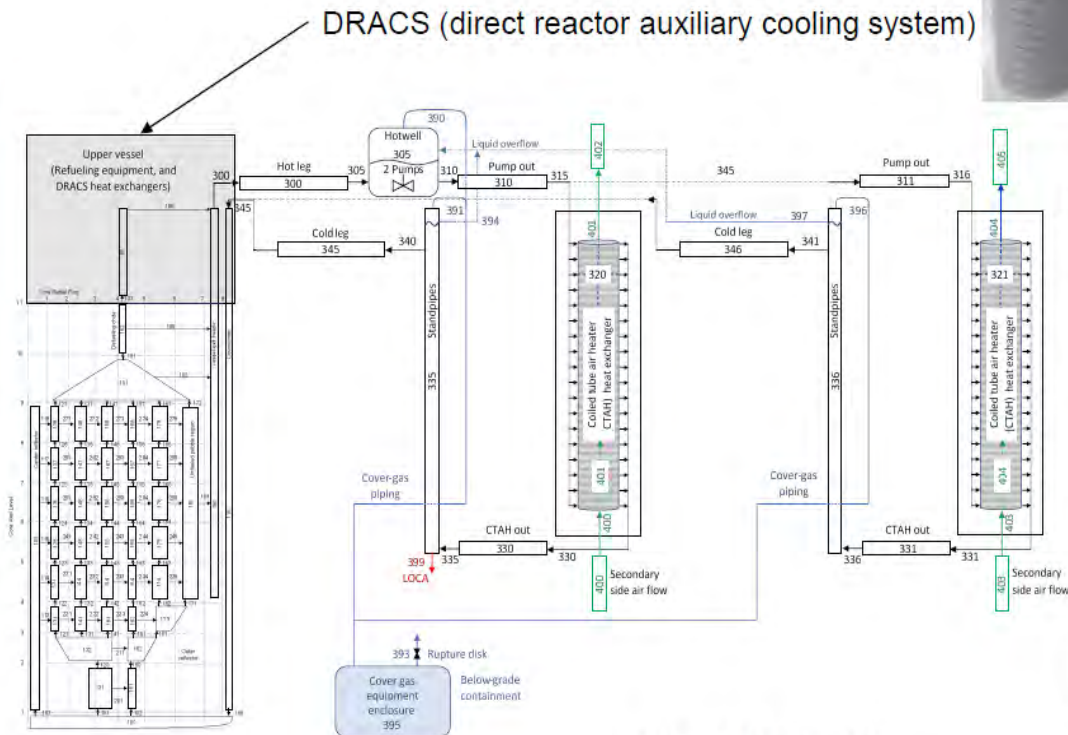
- Iodine diffusivity assumed to be same as krypton
- Assumes most iodine reacts with cesium
- Larger cesium release due its the higher diffusivity
- Ag release to the environment is  $1.2 \times 10^{-3}$  (highest diffusivity)



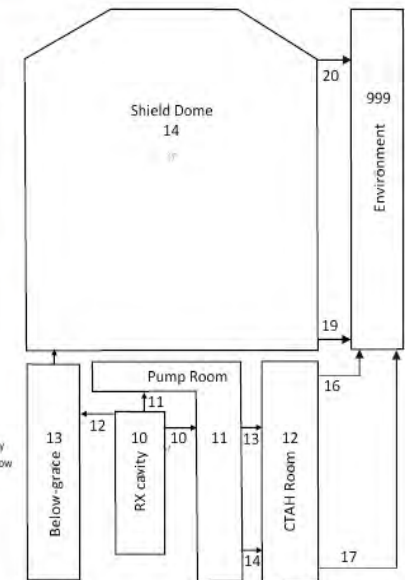
# FHR model (UCB Mark 1)



**SCALE Model**



**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

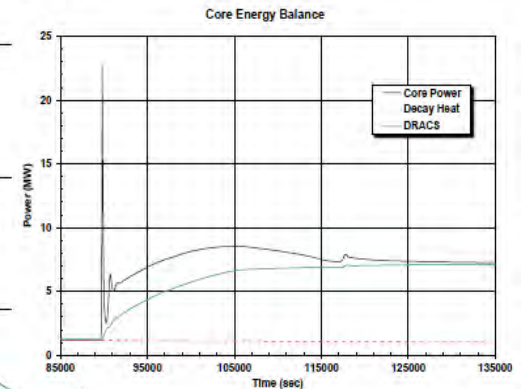
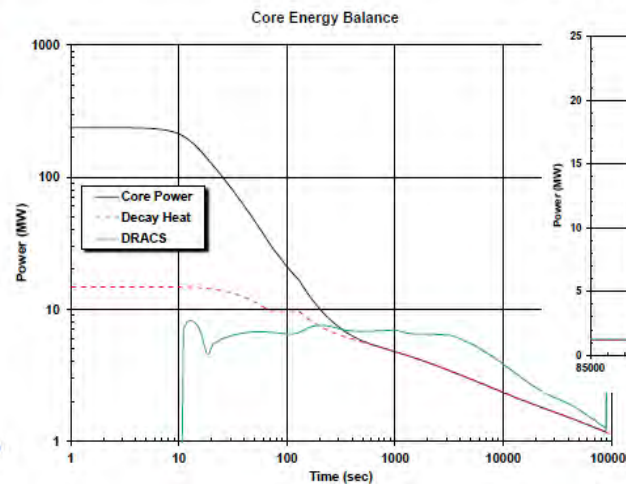
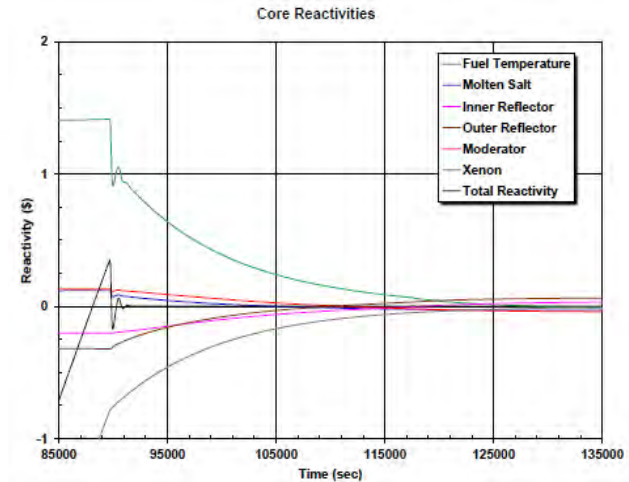
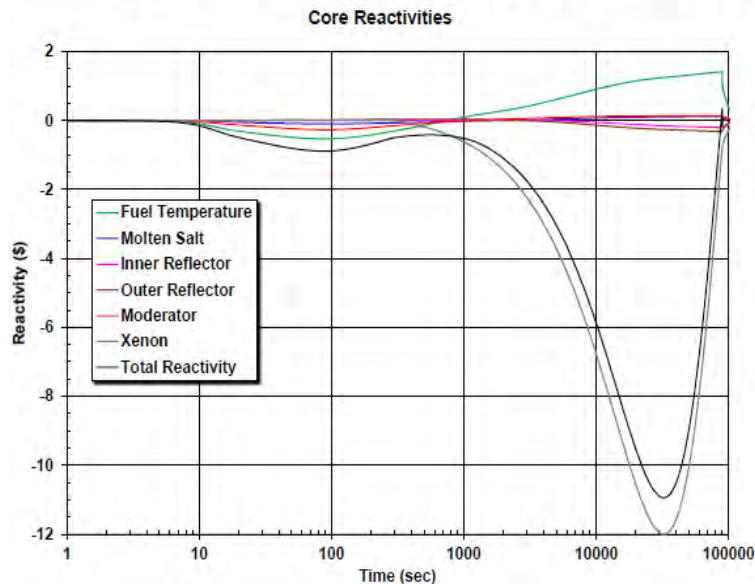
- Guided by ORNL calculations
- Xenon reactivity feedback model being implemented into MELCOR

# 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





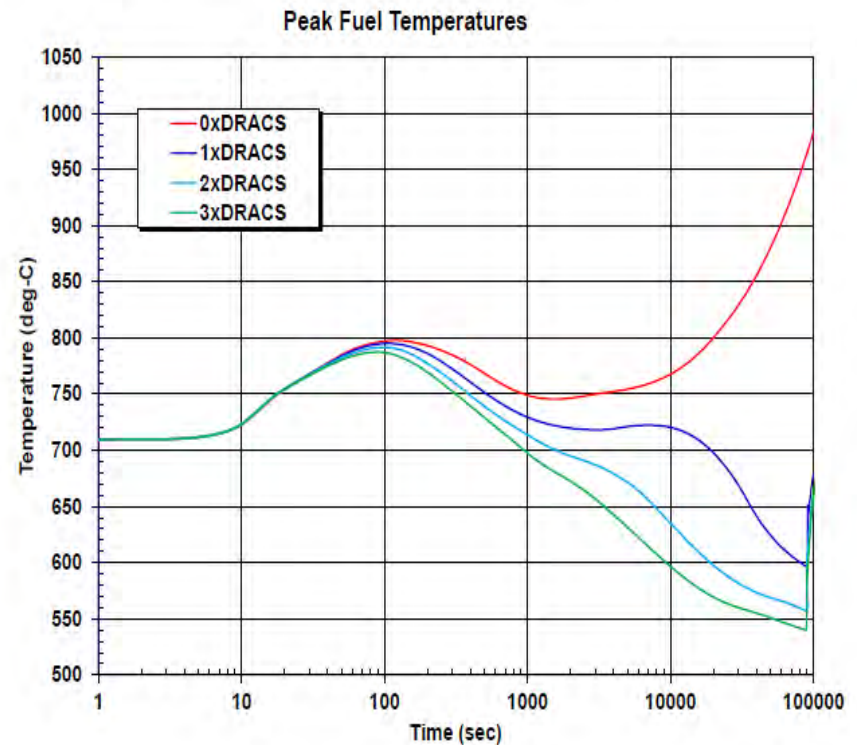
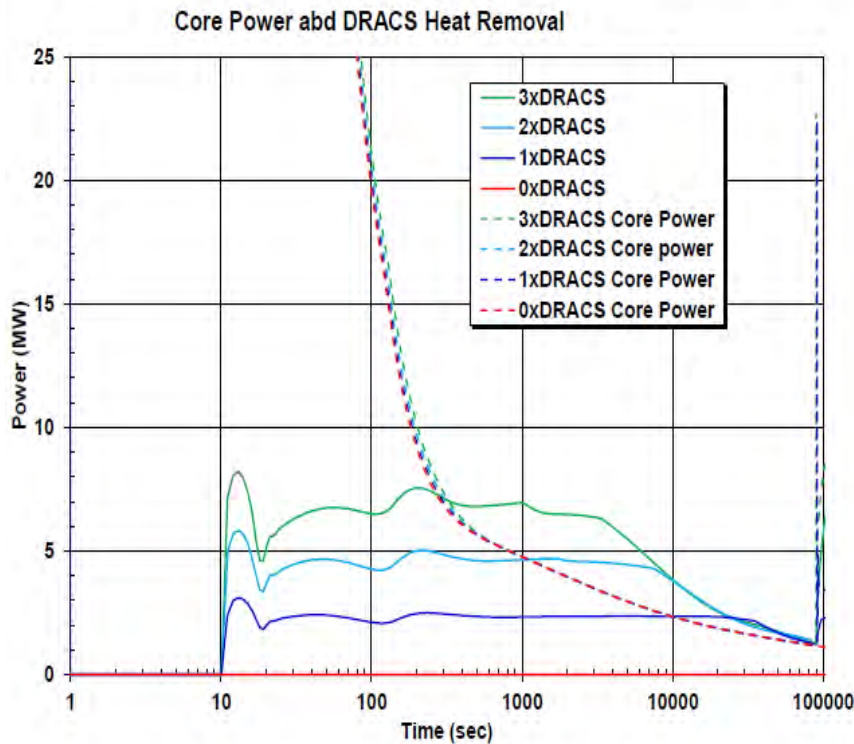
# FHR – ATWS with variable DRACS

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 ( $T_{\text{sat}} \sim 1350$  °C)



# 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](http://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](http://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	LPZ	low-population zone
ATWS	anticipated transient without scram	MSR	molten salt reactor
DBA	design basis accident	NEA	Nuclear Energy Agency
DW	drywell	OCP	operating cycle phase
EAB	exclusion area boundary	ORNL	Oak Ridge National Laboratory
EU	European Union	PBMR	pebble bed modular reactor
DRACS	direct reactor auxiliary cooling system	PIRT	phenomena identification and ranking table
FHR	fluoride salt-cooled high-temperature reactor	PRA	probabilistic risk assessment
HALEU	high-assay low-enriched uranium (fuel)	RPV	reactor pressure vessel
HBU	high burnup (fuel)	SAWA	severe accident water addition
HPR	heat pipe reactor	SAWM	severe accident water management
HTGR	high temperature gas-cooled reactor	SFP	spend fuel pool
INL	Idaho National Laboratory	SFR	sodium-cooled fast reactor
IRSN	Institut de radioprotection et de sûreté nucléaire (France)	SMR	small modular reactor
		SNL	Sandia National Laboratory
		SOARCA	State-of-the-Art Reactor Consequence Analysis

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# **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

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# 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

# 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

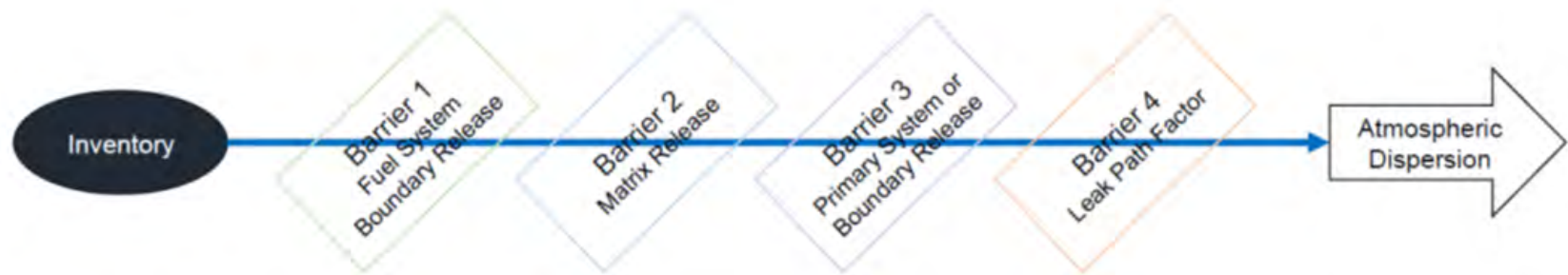
- Technology inclusive, what to do to develop accident source terms, not specific on how to do it
- No specific methods or phenomenological models
- Do not provide technology-related source terms or releases

# 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



$$I(RN_j) * F(S_i, RN_j, t) * MR(S_i, RN_j, t) * PSR(S_i, RN_j, t) * LPF(S_i, RN_j, t) = ST(S_i, RN_j, t)$$

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

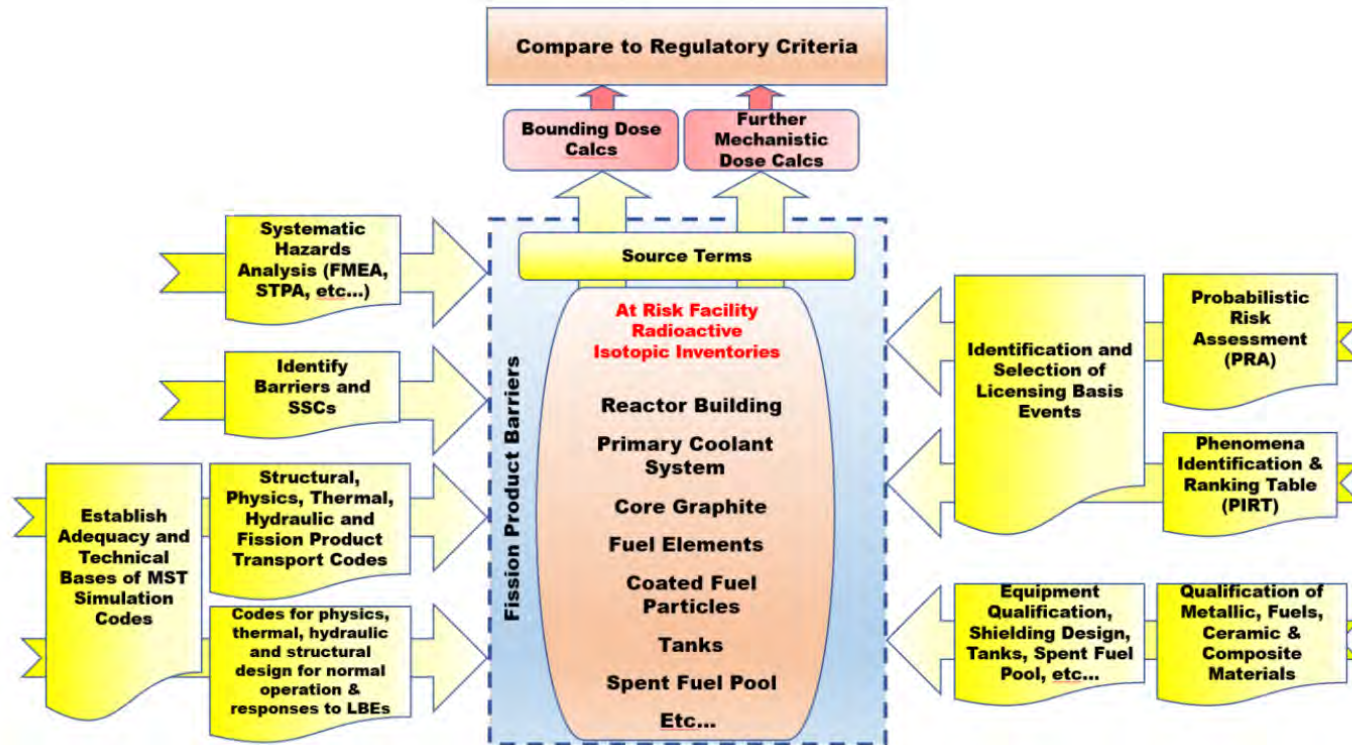


Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).



# 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](#)

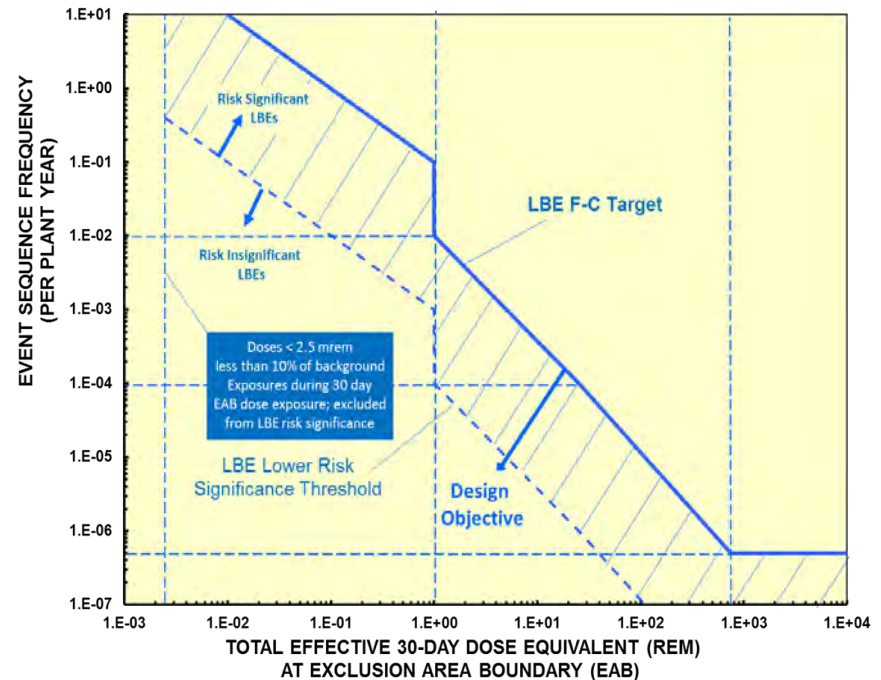
- Primarily qualitative means to identify the dominant considerations that affect a release mitigation strategy
- Classifies release mitigation strategies based on a range of barriers, physical attenuation processes, and system performance under sample accident scenarios
- Did NOT develop quantitative 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

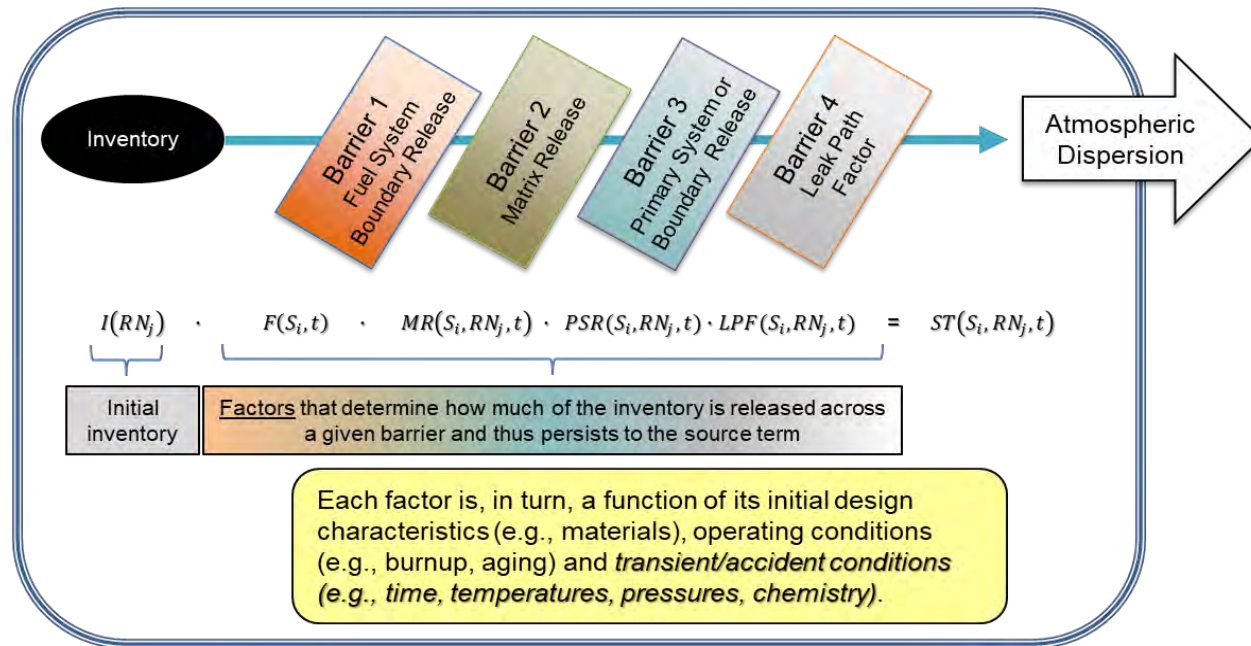
# Licensing Modernization

- Risk-informed approach to selection and analysis of licensing basis events
- Combined with assessment of cumulative risks
- Key roles for PRA and MST



# 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

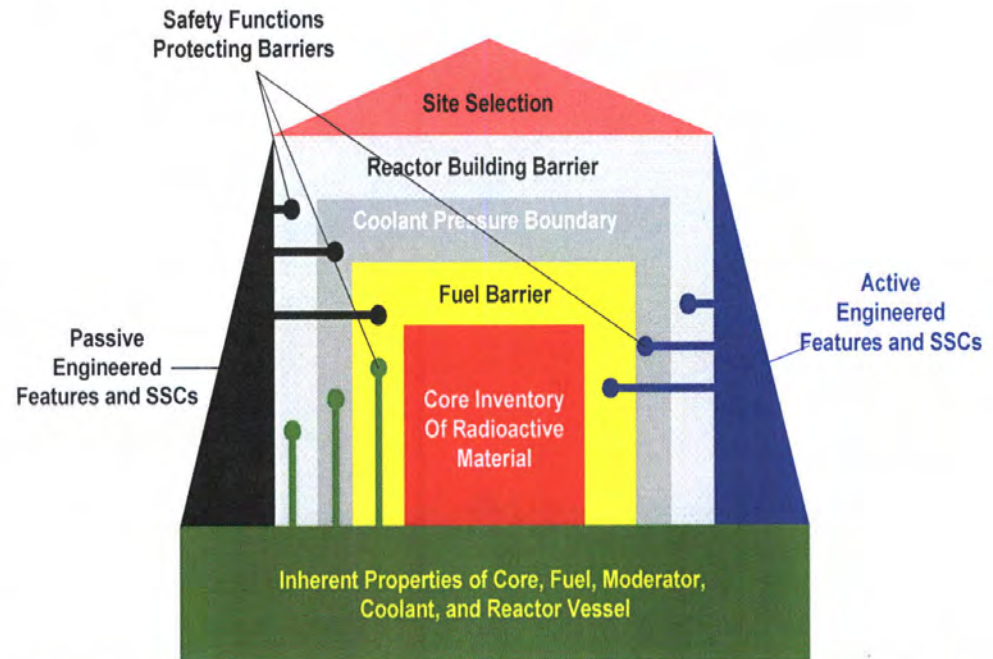
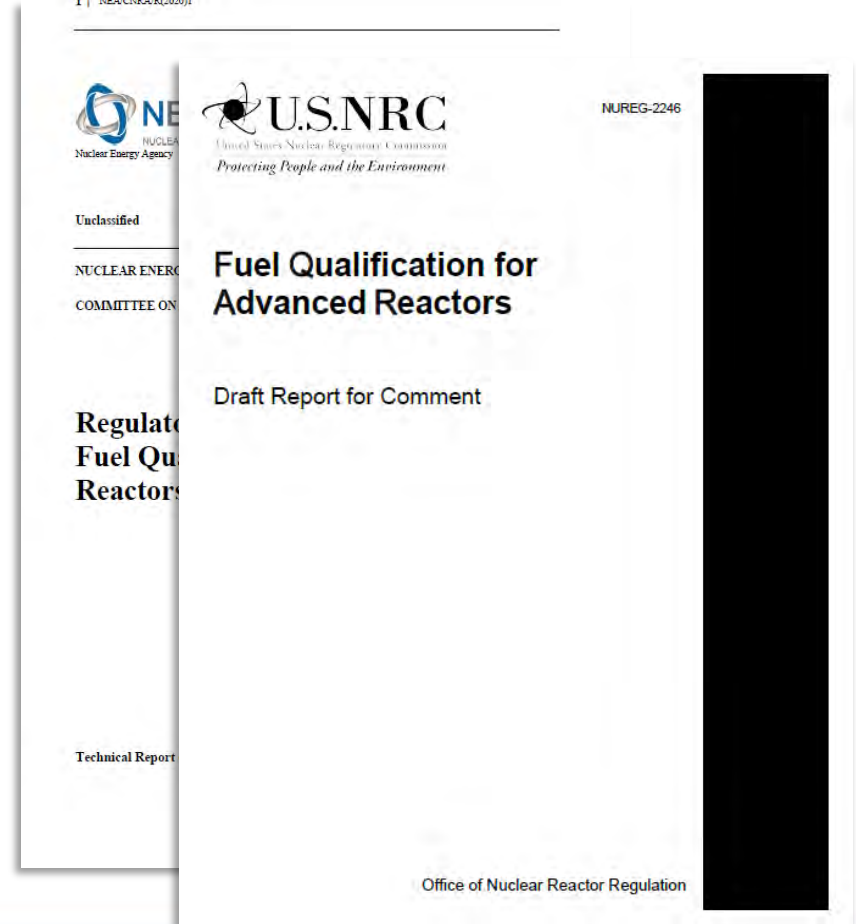


Figure 3-6. Elements of safety design approach incorporated into *Plant Capability Defense-in-Depth*.

# Assessment Frameworks

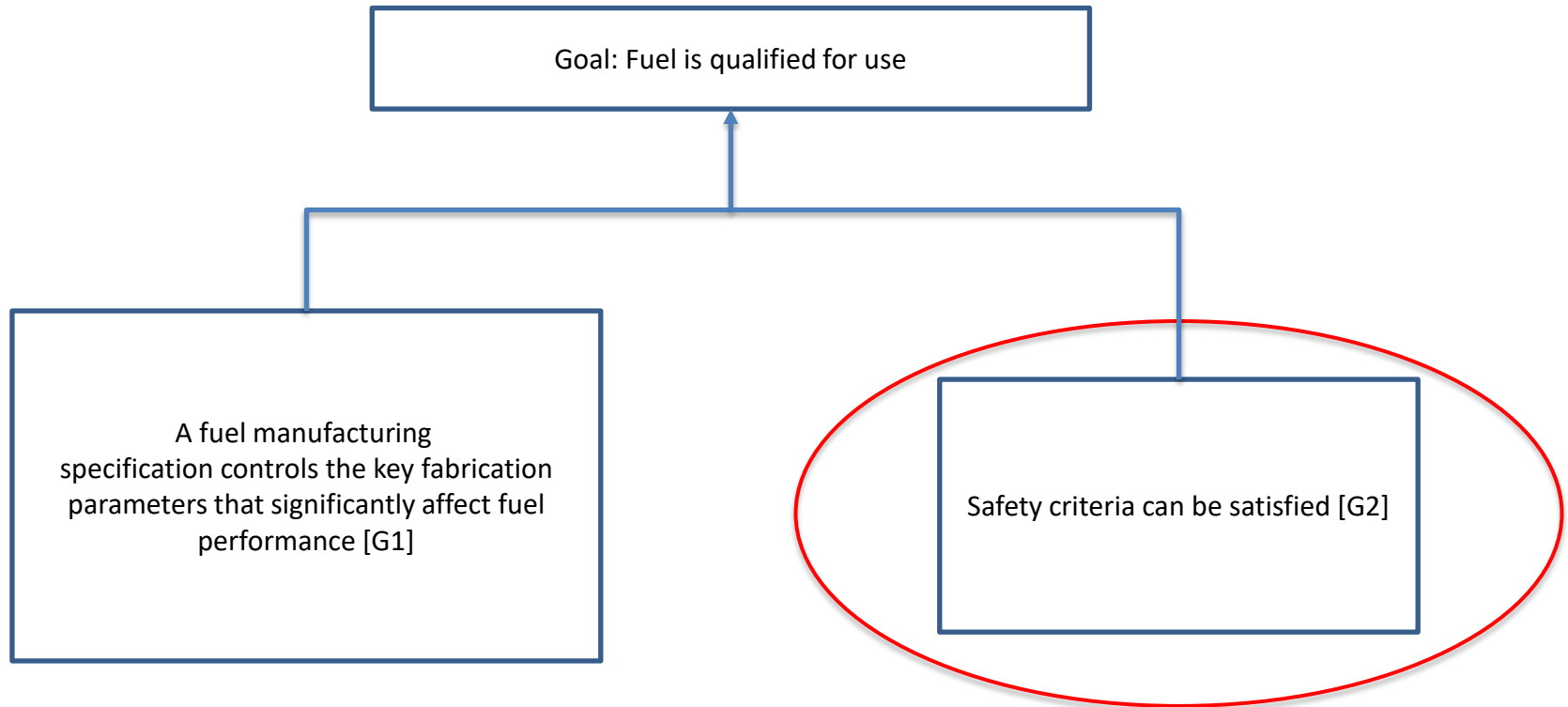
## Fuel Qualification (FQ)

- Top-down approach to identify criteria (goals) to support a finding that “fuel is qualified”



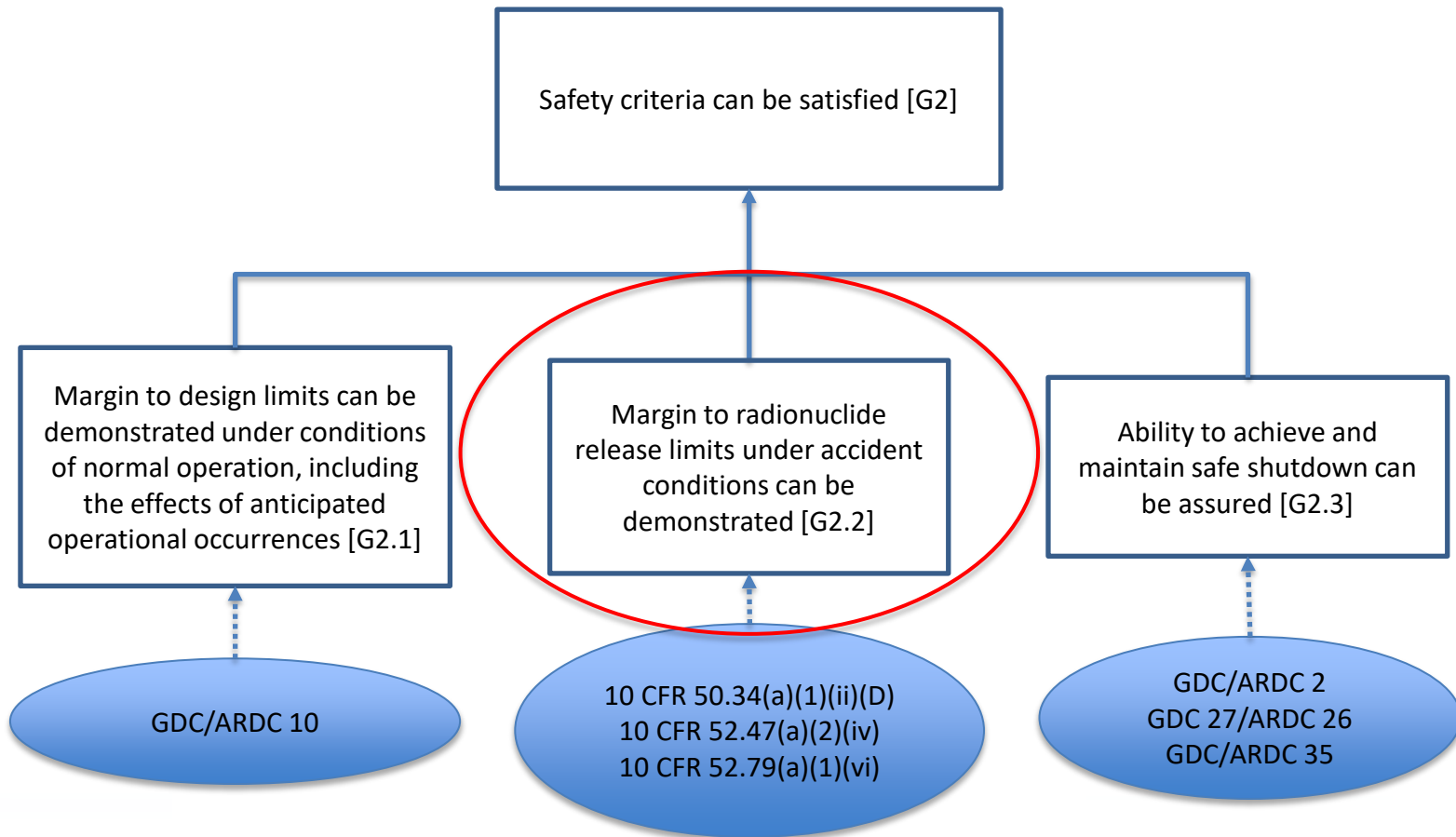


# FQ Assessment Framework

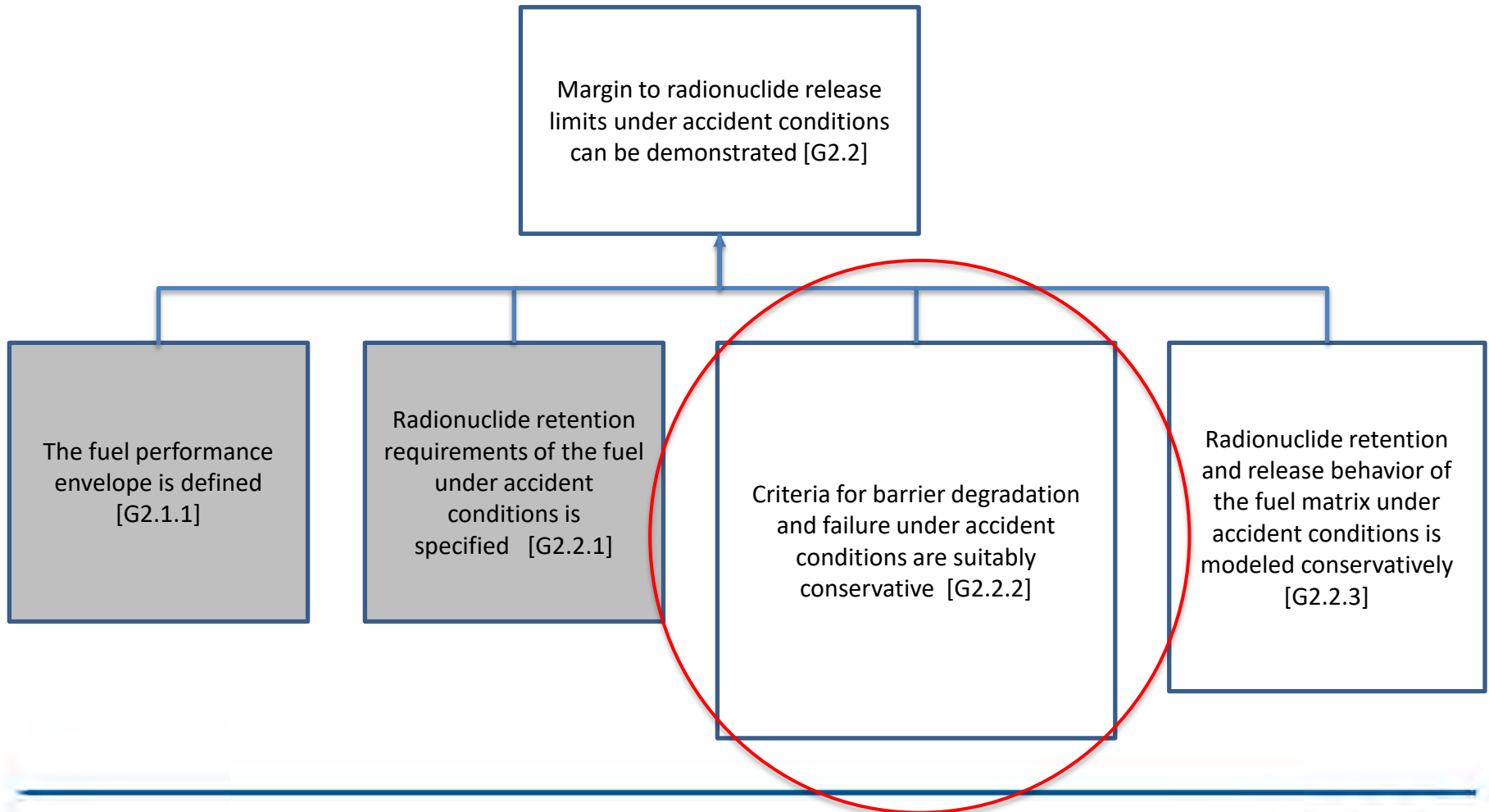




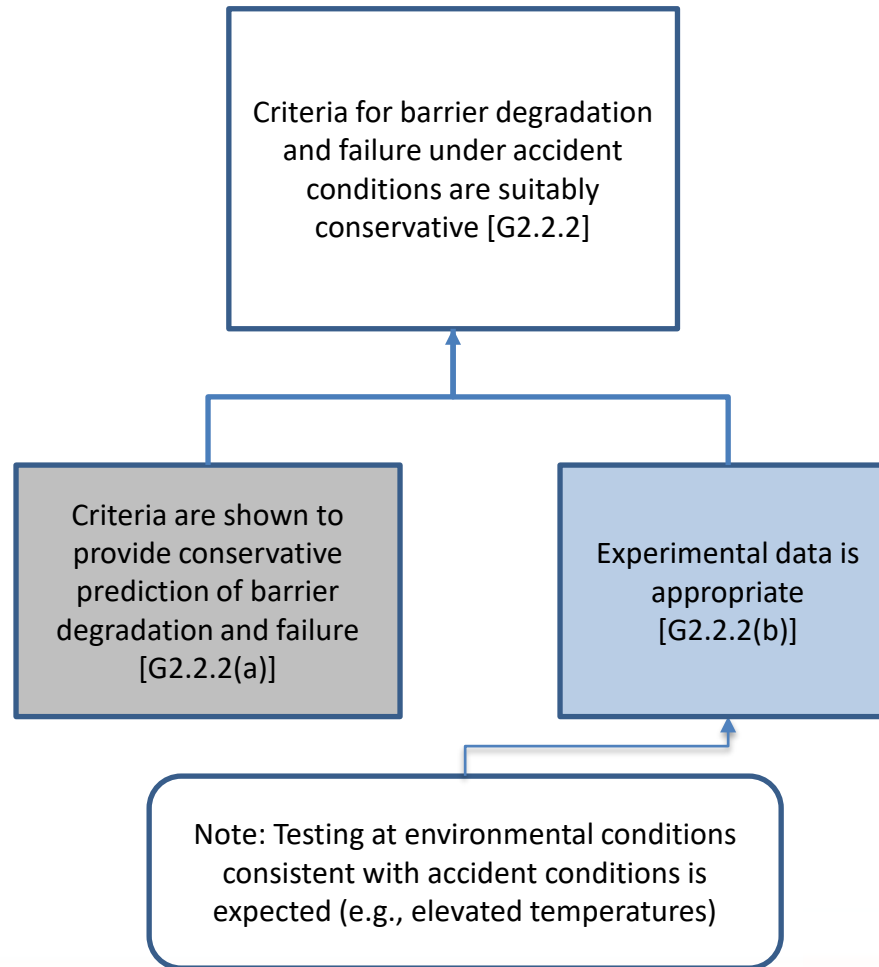
# G2: Safety Criteria



# G2.2: Radionuclide Release Limits



## G2.2.2 Criteria for Barrier Degradation



# 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	
G2.2.1	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)

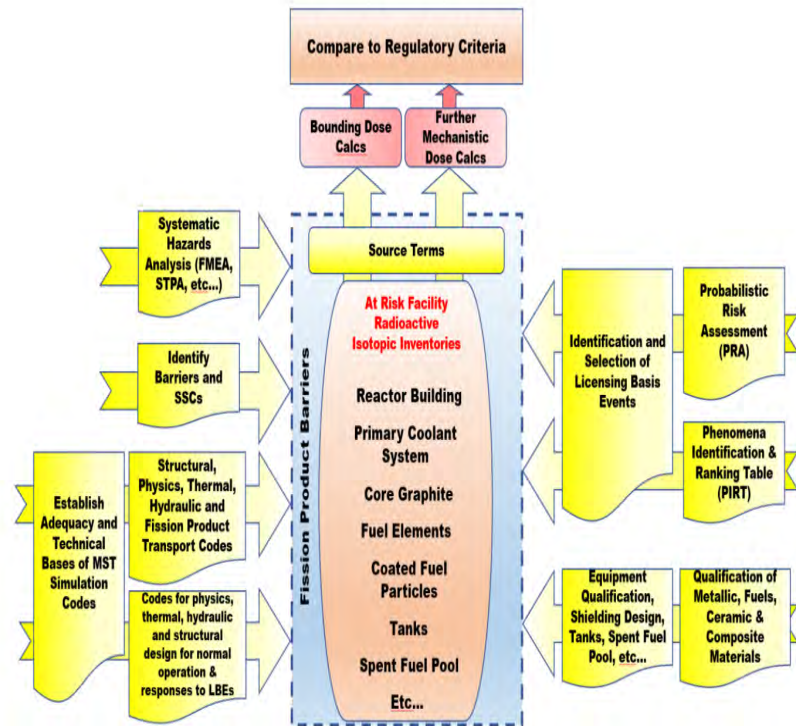
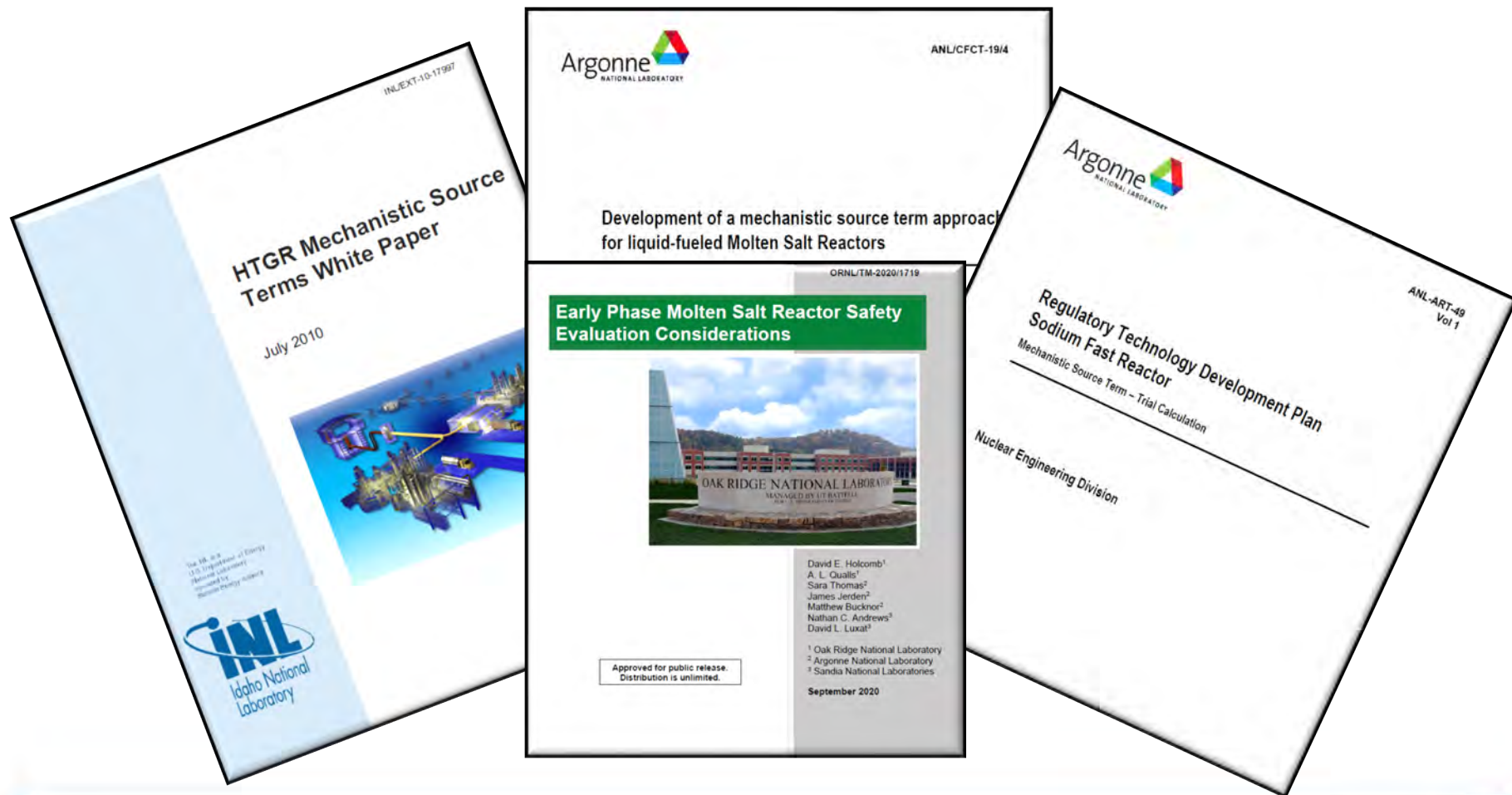


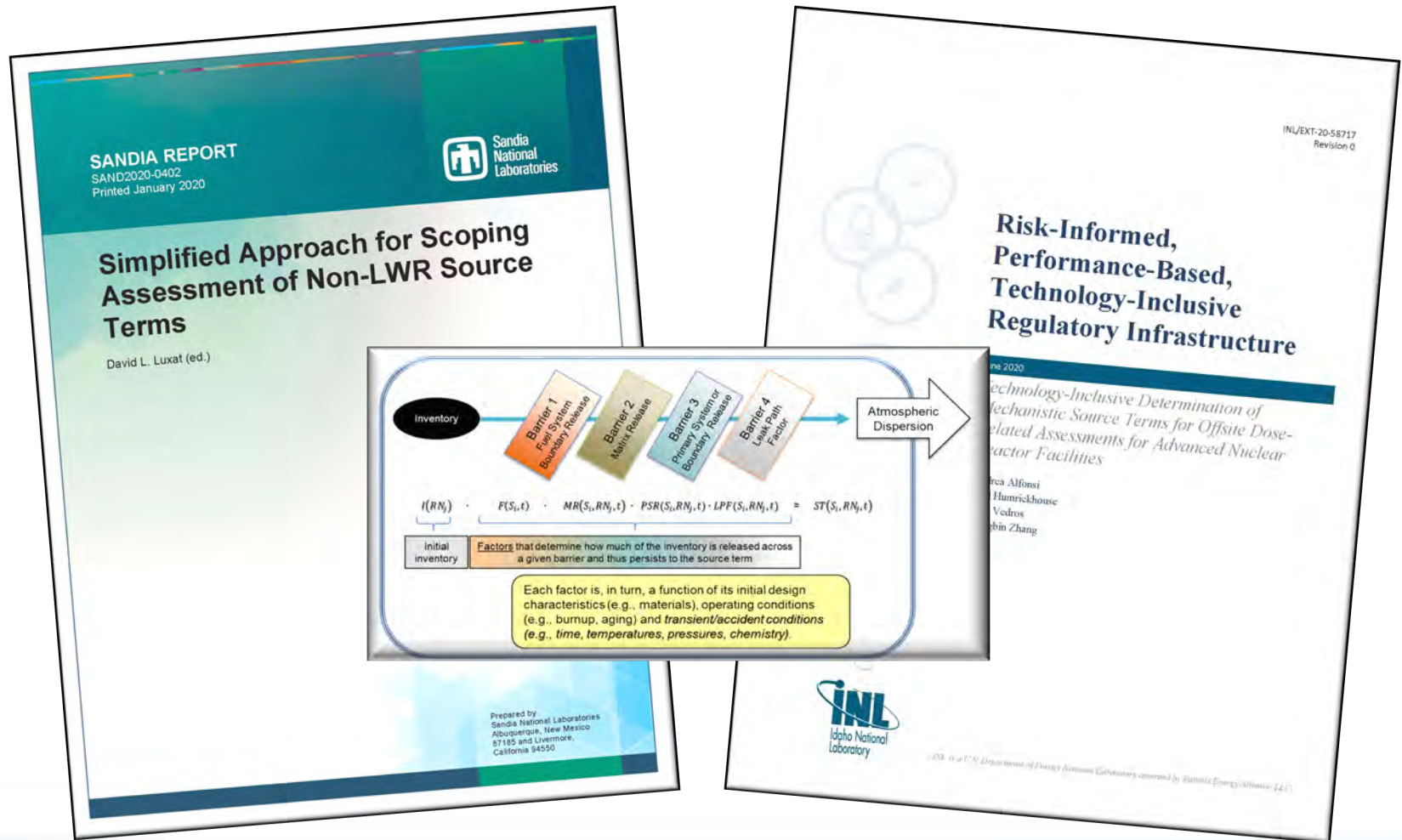
Figure 3-1 Technology-inclusive source terms determination methodology components (modified from Ref. [18]).



# DOE/National Laboratories



# NRC Activities



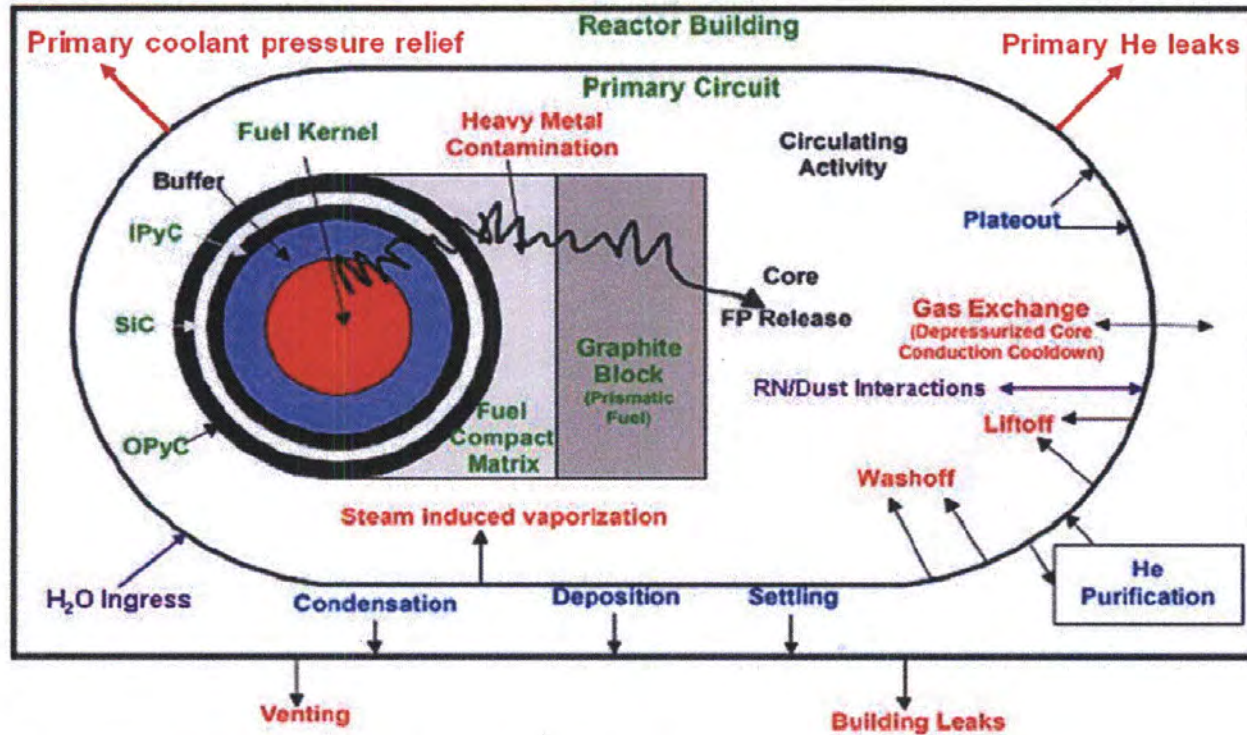
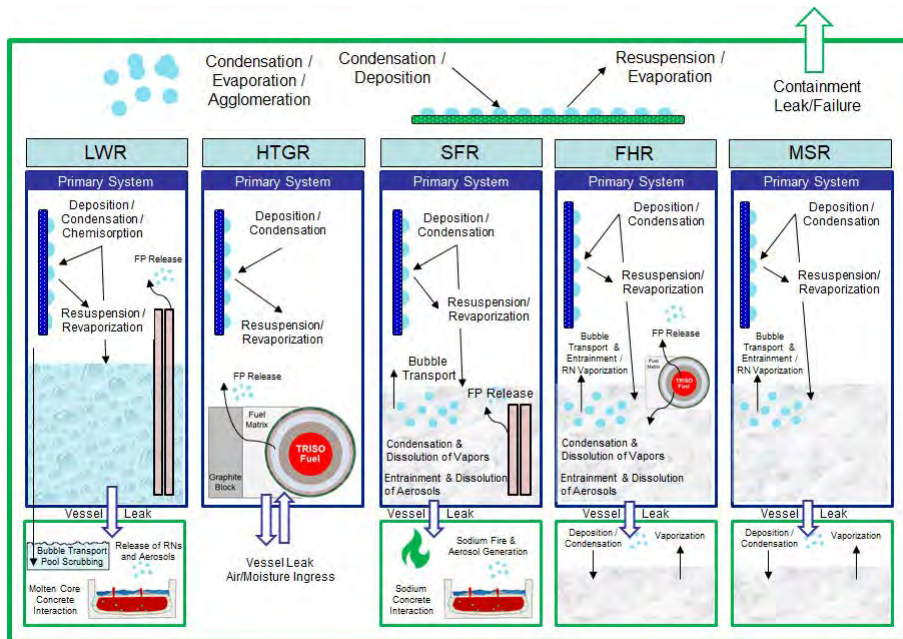


Figure 2-4. HTGR radionuclide retention system.



# Model Development



## Primer & User Guide Reference Manual Assessment Problems

# Applications & Pre-App Interactions



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# 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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# Adjourn

Full Name	User Action	Timestamp
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Walton, Shandeth	Joined before	2/17/2022, 9:00:15 AM
Burkhart, Larry	Joined before	2/17/2022, 9:00:15 AM
Dashiell, Thomas	Joined before	2/17/2022, 9:00:15 AM
Dave Petti (Guest)	Joined before	2/17/2022, 9:00:15 AM
Costa, Arlon	Joined	2/17/2022, 9:04:17 AM
Nourbakhsh, Hossein	Joined	2/17/2022, 9:05:11 AM
Smith, Micheal	Joined	2/17/2022, 9:07:45 AM
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